Reliability Assessment

of the Vermont Yankee Nuclear Facility

Provided by Nuclear Safety Associates

to the

State of Vermont Department of Public Service

12/22/08

REDACTED PUBLIC VERSION

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Executive Summary

Entergy Nuclear Vermont Yankee (ENVY) provides about one-third of the energy for the State of Vermont. Since 2002, ENVY has increased its plant capacity from 1563 Mwt to 1912 Mwt. ENVY, on January 27, 2006, applied to the NRC for a 20-year extension of its operating license. Historically, ENVY has been a reliable source of power for Vermont. However, in recent years the station has experienced several operational events which have raised concerns about the reliability of the station.

The State of Vermont General Assembly passed Legislative Act 189 (S.364) which called for an independent assessment of the ENVY station's current and future reliability. Subsequently, during the period of August 13 through December 16, 2008, a team of nuclear professionals with exceptionally deep and diverse nuclear experiences from Nuclear Safety Associates (NSA) conducted an assessment of Entergy's Vermont Yankee nuclear power station. This assessment was conducted in accordance with the requirements of Legislative Act 189 (S.364) and the scope of work was approved by the State of Vermont Department of Public Service in consultation with the Public Oversight panel.

Act 189 called for a thorough, independent, and public assessment of the reliability of the systems, structures, and components of the ENVY facility and of its management and organizational effectiveness to examine the reliability of the nuclear station. Further details on the purpose, goals and the assessment process are included in the Introduction Section of this report.

As part of this assessment, NSA team members conducted reviews and assessed ENVY performance in comparison with NSA experiences and expectations for high performing nuclear plants. The criteria were applied to the evaluation of ENVY's systems, structures, components, station processes, and management and organizational effectiveness. Overall assessment of performance to these criteria was based on the collective professional judgment of the NSA team members, taking into account a range of qualitative and quantitative factors.

Overall and Principal Conclusions

The overall and principal conclusions are those high level, over-arching or cross-cutting issues that potentially support or challenge reliable operation of ENVY. Assessment findings of a minor nature, or limited to one area, are contained within the respective sections in the body of the Reliability Assessment Report.

Overall Conclusion

ENVY is operated reliably.

Entergy, the 2nd largest nuclear power generating company in the US, purchased the Vermont Yankee Nuclear Power Station in 2002. Following purchase of the station, Entergy made significant investments to improve the reliability of the station. NSA noted that station personnel were effectively trained and qualified to industry standards. Under Entergy direction, ENVY is moving to a fleet standard organization with consistent procedures and standards. Overall, many station managerial and technical areas meet or exceed industry standards for performance. The station is operated and maintained in a reliable manner.

In addition, ENVY can be a reliable station beyond its current operating license, provided that the areas identified in the following principal conclusions are effectively addressed. Management action, oversight and follow-through are needed to ensure that these issues are addressed and resolved if ENVY is to improve its performance to top industry levels.

Principal Conclusions

The following issues are, or may be, watch areas or challenges to plant reliability.

1. Procedure quality issues

NSA review of procedures determined that, while procedures were technically correct, the current formatting did not readily support Human Performance (HU) tool usage, such as place keeping and data collection on each page. The formatting also was not up to current industry standards relative to linkage to other procedures. The existing format also lacks specific guidance at times, with 'if desired; when necessary' statements, leaving it open to interpretation and judgment by workers. As a result, there have been plant events related to procedure quality or procedure use and adherence.

Previously, ENVY had a stable workforce. However, in recent times there has been an influx of new employees, especially in the Operations Department and the Maintenance Department Electrical and Instrument and Controls sections. These newer individuals will be more dependent upon detailed procedure guidance.

In recognition of these procedure shortcomings, ENVY recently developed an action plan to improve station procedures. The plan is currently focused on developing a process to identify which procedures to upgrade on a priority basis; considering: condition reports, frequency of use, complexity, significance and other criteria. The General Manager Plant Operations stated that he intends that this new plan will supersede the procedure efforts that were previously ongoing in the Maintenance Department.

Once the full scope of procedure upgrades is identified, a detailed schedule will need to be developed to determine which procedures will be completed in order of priority. A detailed change management plan should also be developed to help manage the overall process and ensure its completion, especially in light of previous procedure projects being aborted. In recognition of the need for better procedures and the potential costs and complexity of this project, this is considered a challenge to future reliability.

2. Human Performance Issues.

Safety and Human Performance is one of 4 site focus areas identified in the ENVY 'Good to Great' program. ENVY has had issues with Human Performance in the past and conducted training on Human Performance expectations for self checking, peer checking, procedure adherence and pre-job walk downs after an Emergency Diesel Generator (EDG) Trip in February, 2007.

However, Human Performance does not meet expectations at ENVY because the organization continues to have issues. Some examples are:

- Procedure use and compliance - A common cause analysis was performed for Condition Report (CR) CR 2008-02152 to evaluate the causes for recent human performance events. One of the contributing causes identified was procedure use practices, and ENVY committed to develop a procedure use and adherence improvement plan which is currently in progress. This is an ongoing Human Performance issue.
- High OSHA Recordables - Benchmark data shows that ENVY is in the bottom quartile with respect to OSHA Recordables when compared to the sister plants. The site has recognized the need to make improvements in this area but contractor injuries remain high and ENVY had another OSHA Recordable during the October 2008 outage.
- Foreign Material Exclusion (FME) and Housekeeping practices - The Housekeeping and FME Programs at ENVY do not meet industry standards. Plants with good FME programs have high standards for housekeeping and FME controls. ENVY's performance, especially during the outage when it is most critical, was less than adequate. The low number of Condition Reports (CRs) identifying poor worker practices associated with FME indicates an inadequate threshold for identifying FME and Housekeeping issues.

In addition, as part of the Corrective Action Process, the Condition Report Review Group (CRG) can require a Human Performance Error Review (HPER) as part of the CR analysis process; however, these are infrequently performed. The high number of Human Performance events suggests that HPERs should be performed frequently.

Human Performance can impact plant reliability in numerous ways. Failure to follow procedures, FME which causes equipment problems or fuel failures, or an individual making a mistake which causes a loss of generation are all Human Performance issues. Therefore, Human Performance is a challenge to future reliability.

3. System and Technical Focus Issues

Condensers

ENVY recognizes that they have issues involving the plant's condensers and management has discussed long term options to resolve the problems. Chemistry index continues on an adverse trend. This is due partially from the increased flow as a result of the Extended Power Uprate (EPU) carrying materials from the condenser tubes. The demands on the condensate and feedwater system demineralizers have increased. All demineralizers are required to be in service at 100% power.

Therefore, when one needs to be backwashed or maintenance needs to be performed, the demineralizers have to be bypassed which causes a conductivity spike. ENVY has budgeted for future condenser improvements; however, the current condition of the condenser, coupled with the increased flow from EPU, is posing both a reliability challenge and affecting plant chemistry. The options to re-tube or replace the condenser with erosion resistant materials to mitigate these effects or to increase demineralizer capabilities are on hold until the decision is made regarding the plant license extension. It is the opinion of the NSA Team that this is a challenge to both near term and long term reliability.

Cooling Tower (CT)

ENVY developed a repair/upgrade plan for the Cooling Towers (CT) after the cell collapse in 2007. The plan covers 3 years for the non safety-related portion, with the first step completed in Spring 2008. During the Fall 2008 outage, the safety-related cell was inspected and repairs made. During the inspection, degraded columns were found which were not scheduled for replacement, and columns which were scheduled for replacement were more degraded than anticipated. On-line visual inspections did not identify these issues. Inspections and repairs to be performed during the spring of 2009 on non-safety related cells could also reveal items not found during on-line visual inspections.

A re-evaluation of on-line inspection methods and the repair plan/schedule for safety and non-safety related CT cells should be performed to ensure long term reliability especially considering that the current plan is based on system conditions that are now known to be degraded beyond their initial assumptions. It is the opinion of the NSA team that the Cooling Tower is a challenge to future plant reliability.

Spare Main Transformer

Large Power transformers such as ENVY's Main Transformer manufactured by ABB are not off-the-shelf or in stock items from a manufacturer. There is an industry-wide challenge regarding the ability to obtain replacement transformers in a timely manner. To acquire one could potentially take several years for delivery. The current designated spare for the Main Transformer at ENVY is the previous Main Transformer manufactured by Peebles, which was removed from service prior to the uprate of the plant. At the time of removal, the transformer was experiencing gassing issues (typically caused by a condition involved with the degradation of the transformer windings). ENVY decided to replace the transformer in anticipation of the Extended Power Uprate initiative. The spare transformer is capable of being utilized if the currently installed transformer (ABB) fails, but only at 80% of the current rated output of the plant. Additionally, the gassing issue in the Peebles transformer has not been addressed. This is a potential reliability issue. ENVY is in the process of addressing this issue as part of the Transformer and Switchyard system long-term plan. A corrective action (LO-VTYLO-2007-00136) was entered into the Corrective Action Program (CAP) to address the preventive maintenance and monitoring activities for maintaining the Peebles transformer in a 'ready' for installation condition. However, the actual plan for maintenance and monitoring activities needs to be developed and actions must be completed, to address the gassing issue. In addition, LO-VTYLO-2007-00136 does not address the Peebles transformer relative to its capability to provide 100% power.

The NSA team considers the transformer to be a watch area. A more comprehensive plan to include potential upgrades to the spare or access to a 100% load capacity spare should be included.

4. Delays in Adopting Industry Equipment Reliability (ER) Best Practices

The current ENVY culture is based on being a ‘single-plant company’ for most of the plant’s operating history. Being exposed to numerous process changes as part of a large fleet like Entergy is a cultural change for many people. The Systems and Component/Programs Engineering Managers and Supervisors are very experienced but appear to not fully appreciate the value of fleet Equipment Reliability process standardization, which may be contributing to the slow movement to the new processes.

- Equipment Reliability Index

  The Nuclear Power Industry has worked together to define and standardize the Equipment Reliability (ER) processes and definitions. As part of this standardization effort, the Nuclear Industry’s Equipment Reliability Working Group, comprised of most US Nuclear Generators, has created the ‘ER Index’ (ERI) made up of an aggregate of 19 standard performance metrics. ENVY has only recently begun to utilize the industry standard ERI as part of the Monthly Management Review Meeting.

  When compared to the US Nuclear Industry, the ENVY ERI index is in the bottom quartile. Major contributing factors for this poor performance are described in Section 1.3.

- System and Component Engineering Staffing and Expertise

  Attracting and retaining qualified, experienced personnel in the Systems & Component/Program Engineering Groups has been challenging at ENVY. There are vacant positions within the System Engineering group at ENVY resulting in some System Engineers having responsibilities for as many as 6 systems as compared to the industry average of 2 to 4 systems per System Engineer. Within the Component/Programs Engineering group, there has been a 40% turnover (eight engineers left, transferred or retired from the group) over the past year.

- System Health and Performance Monitoring

  The ENVY site manages system health in a matrix approach where the System Engineer, Component Engineer and Program Engineers have responsibility for specific processes, but no distinct single point accountability exists for ensuring system health. This matrix approach is currently effective at ENVY, where system, component and program engineers have high levels of experience. However, as the workforce turnover and near-term retirements impact the experience levels at ENVY, it will be more difficult to ensure reliability with this matrix approach. Industry top performers have implemented the System Manager concept, which emphasizes single point accountability for overall system health.

A detailed Change Management plan needs to be developed and executed to ensure timely transition to ER industry standards. The delay in adopting industry ER best practices and transition to fleet-wide ER processes could result in challenges to future plant reliability.

5. Ineffective Use of the Change Management.

ENVY has had major organization and operation changes as part of its transition to the Entergy fleet model, including staffing changes, organization realignment, and adoption of new processes, standards, and procedures. The steps taken to plan and implement the changes are governed by a Fleet Change Management process detailed in the Nuclear Management Manual EN-PL-155. Entergy and ENVY have not used the Change Management process effectively to implement many of the major change initiatives.

The Change Management process is intended to guide the user in developing a comprehensive plan to identify why the change is being made, what needs to be accomplished, who is responsible for implementation of actions, performance indicators to help track and measure transition progress, and dates for completion. Following are examples of major changes that do not have comprehensive Change Management plans.

- An integrated 'Fleet Transition Plan' to drive full integration of ENVY into the fleet could not be found. The 'Good to Great' plan, which is intended to set direction and help drive the station to top decile performance, is not a step-by-step plan. The plan is mostly a communication tool to keep employees informed and foster support for the 'Vision' to improve.
- The lack of Corporate direction for common procedures has hampered the procedure upgrade process at ENVY. Subsequently, ENVY has started and stopped procedure upgrade projects and has now developed another procedure upgrade effort for working level procedures. Detailed Change Management plans for these efforts have not been created.
- A major technology change from EMPAC to Indus (the computerized maintenance management system) was implemented without a comprehensive Change Management plan which has slowed the transition and use of the new program.
- ENVY recognizes the importance of the upcoming proposed transition to Vermont Electric Power Company (VELCO) for operating and maintaining the high voltage switch yard. A Change Management plan should be created for the transition.

Transition from a stand alone single unit plant to being part of a large fleet requires major changes in organization and processes. Detailed Change Management plans are needed to ensure timely and effective implementation of the changes. ENVY's transition has been ineffective or slower than necessary because of inadequate use of Change Management. The NSA team opinion is that the Change Management process is a watch area.

6. Shortcomings in contractor oversight.

Lack of contractor oversight of Cooling Tower (CT) work was identified as a cause of a cell collapse in 2007. In July 2008 problems were found with sagging of the distribution piping and leakage on one slip joint, in an area repaired during Spring 2008 work. Connections made during this work were not installed correctly. Similar issues with connections were found in two other cells. Again, lack of contractor oversight contributed to this issue.

Contractor Oversight is provided from two organizations at ENVY - Maintenance and Engineering Project Management. Outage work on the Cooling Towers is controlled by the Engineering Project Management group in accordance with their group procedures, and during non-outage by Maintenance. Maintenance does not have a program, procedure or specific training for the individuals responsible for Project Management.

Oversight issues identified during events include lack of drawings, inadequate Work Orders, insufficient oversight, and over reliance on the skill of the vendor craft. In addition, there was excessive delegation and lack of 'trust but verify' behavior by ENVY.

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Contactor oversight does not meet industry standards and has already caused reliability issues for ENVY and therefore, contract oversight is a watch area.

Introduction

Purpose, Goals and Assessment Approach

Entergy Nuclear Vermont Yankee provides about one-third of the energy for the state of Vermont and now is applying for a 20-year license extension beyond its original license expiration. The purpose of this assessment is to comply with the State of Vermont Legislative Act 189. The Act called for an independent assessment of the reliability of the systems, structures, and components of the ENVY facility; and management and organizational effectiveness, to examine the comprehensive reliability of the nuclear station to assist the state in determining if it should be authorized to operate beyond the current license expiration date of March 21, 2012.

Historically, ENVY has been a reliable source of power for Vermont. However, in recent years the station has experienced several operational events (e.g. partial collapse of a cooling tower, main transformer fire, and problems with the Reactor Building Bridge Crane while handling spent fuel), which raised concerns about the reliability of the station.

Therefore, the goals and objectives for this assessment, as stated in Vermont Legislative Act 189 were to:

1. Assess the conformance of the facility to its design and licensing bases, for operating at up to 120 percent of its originally intended power production level, including appropriate reviews at the plant's site and its corporate offices;
2. Identify all relevant deviations, exemptions, or waivers, or any combination of these from any regulatory requirements applicable to Vermont Yankee and from any regulatory requirements applicable to new nuclear reactors, and verify whether adequate operating margins are retained despite the cumulative effect of any deviations, exemptions, or waivers for the present licensed power level for the proposed period of license extension;
3. Assess the facility's operational performance, and the facility's reliability for continued power production, giving risk perspectives where appropriate;
4. Evaluate the effectiveness of licensee self-assessments, corrective actions, and improvement plans; and
5. Determine the cause or causes of any significant operational shortcomings identified, and draw conclusions on overall performance.

Specific responses to these 5 goals and objectives can be found in Appendix A

In preparation for a Reliability Assessment of the Vermont Yankee Nuclear Facility, the NSA evaluation team members reviewed Legislative Act 189, the scope of work approved by the Vermont Department of Public Service, and the requirements matrix created by the Vermont Department of Public Service and in consultation with the Public Oversight Panel (Refer to Appendix D). From this review it was determined that a limited scope vertical system evaluation approach would be applied to the list of safety-related and balance-of-plant systems that were provided in the requested scope of work by the State of Vermont. In addition to the six systems listed in Act 189, seven technical focus areas were evaluated.

To assess the management areas, horizontal and vertical evaluations were conducted for key site organizations and their associated plant processes in the Operations, Maintenance, Engineering/Equipment Reliability, and Work Control areas. The scope included an assessment of the following elements to identify issues which could potentially impact plant reliability:

- Organization and staffing
- Experience levels and training
- Procedures
- Observation of field activities
- Human performance
- Overall departmental performance

For the management assessment, NSA team members used a variety of assessment tools, including:

- Interviews with Corporate and Station personnel.
- Review of selected station procedures and documents.
- Evaluation of processes and programs associated with procedures.
- Review of Performance Indicators and any associated analysis details.
- Review of Condition Reports and effectiveness of improvement plans/corrective actions.
- Review of station Self Assessments.
- Review of regulatory reports pertaining to performance and performance deficiencies.
- Attendance at various management organizational meetings.
- Plant and control room tours.
- Observation of operator rounds, on-line maintenance, and outage work activities.
- Evaluation of the methods and effectiveness of organizational communications.
- Attendance at various classroom and simulator training sessions.

In addition, NSA conducted an industry benchmarking study to compare ENVY staffing, performance, and Equipment Reliability Index (ERI) to other U.S. nuclear plants.

Any organizational performance trends or concerns were identified and evaluated. The intent of these evaluations was to identify potential organizational deficiencies or significant risks or challenges to plant reliability

1.0 Management and Organizational Performance Assessment

1.1 Corporate Elements

Introduction

ENVY was purchased by Entergy in 2002 and has been part of the fleet since that time. In general, across the nuclear utility industry, fleet ownership has proven to be positive when plants are purchased by a large nuclear operating company like Entergy; particularly single unit plants operating on their own. Nuclear fleets offer best practices, capital funding, technical support, operating experience, management talent, and transparency which benefit all plants in the fleet.

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Entergy Corporation Fleet Structure

The following describes the Entergy Corporate Organizational structure and guiding documents. The impact on ENVY is discussed where appropriate.

Entergy Nuclear Operating System (ENOS) – Defines Entergy Nuclear through the common policies, processes and procedures that the Entergy Nuclear fleet follows to do business. ENOS includes statement of Vision, Mission, Strategy, Values and Operating Philosophy. ENVY supports the high level common policies, processes and procedures established as part of ENOS.

Standard Organizational Structure – Maintains a common, standardized organizational structure for headquarters, single unit plants <900 MWe; single unit > 900 MWe; and dual unit sites. This structure is approved and controlled through a common fleet procedure EN-HR-134 Standard Organization Change Process.

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GOES Matrix – GOES – Governance, Oversight, Execution and Support Matrix is a controlled listing of functions and responsibilities performed within Entergy Nuclear. This matrix establishes the organization responsible for each role for every activity performed.

GOES defines the Corporation's role in managing plants within the Entergy Fleet. Although ENVY is part of this process, it has not made as much progress as would be expected after almost seven years as part of the fleet. This may be due to ENVY operating as an independent stand-alone plant for many years, but Corporate has not provided the Governance and Oversight necessary to assure that ENVY participates fully in the process.

Management Review Meetings (MRM) – A fleet standard comprehensive plant status presentation on key indicators and information presented by the site team to the Senior VP and representatives from corporate organizations and other site representatives. The meeting's purpose is for the site team to present the current performance, challenges and actions being taken to improve performance. The meetings occur 6 to 8 times per year at each station, and help to create a transparent atmosphere.

Fleet Performance and Planning Review Meeting – Entergy Nuclear Directive EN DIR-OM-002 describes the structure for the Fleet Performance and Planning Review meetings to be held three times a year. As part of the alignment effort, it was concluded that a fleet forum was needed to consolidate various regional review meetings, and to allow leadership to come together regularly under a structured method to discuss the fleet performance, strategic and financial planning, and human resource planning. In other words, do at the fleet level what should be done at the fleet level, to allow the regions to do what should be done at the regional level.

Entergy Nuclear Functional Area Summary Report – Entergy Nuclear Directive EN DIR-OM-003 establishes the standard, format, process, and distribution for the Functional Area Summary Reports and the Functional Area Performance Summary. This report is issued monthly and quarterly to provide management with the status of Performance Indicators, Peer Group Activities and Peer Group engagement in the functional areas identified in the Directive.

Regional Scorecard Report – A fleet-wide report produced monthly which contains a standard set of key performance indicators for each station. This score card is reviewed by Senior Vice Presidents.

Performance Indicator Data Base – A data base available on the Entergy Nuclear website that contains all indicators used by Entergy Nuclear. It contains PIs and goals established by the Senior VP Nuclear Operations; PIs for MRM's and those PIs determined by Peer Groups.

Entergy Corporation Operations Support

The Operational Support organization provides governance, execution and support of nuclear functions. For groups like Operations, Maintenance, Radiation Protection, and others, the governance is provided by a fleet manager that drives standard processes via the peer group process. Execution is the responsibility of the on-site managers. For other groups, including Training, Licensing, Material, Purchasing and Contracts, Security, etc., the managers located at each station report to a corporate Director responsible for both governance and execution.

Entergy has recently implemented the fleet manager concept; a corporate functional area manager to provide guidance and assistance for fleet standard processes and enhancements. The positions were created to provide improved leadership for the peer groups and their initiatives. Peer Groups can be an effective way to develop best practices for nuclear fleets. Entergy's progress in this area has been slow and it has impacted process improvements at ENVY, such as procedure upgrades and human performance improvements. It is too soon to determine if the fleet manager concept will improve Peer Group activities, but this area should be monitored to assure continuous progress.

Equipment Reliability

The Equipment Reliability Improvement Process for Vermont Yankee is standard across the Entergy fleet. A fleet-wide Self Assessment of Equipment Reliability was performed in 2007. Approximately 400 corrective actions were developed and assigned, with approximately 50 assigned specifically to ENVY. Follow-up assessments are currently in progress, with follow-up at ENVY scheduled for early 2009.

An Entergy Nuclear Equipment Reliability Excellence Plan has been developed that establishes the foundation and primary initiatives for improving Equipment Reliability, both across the fleet and at ENVY. The Excellence Plan projects the Entergy Nuclear vision of Zero Tolerance for Unanticipated Equipment Failures, as discussed in EN-PL-161.

Elements of the Entergy Equipment Reliability Improvement Process

Unit Reliability Team (URT) is where key equipment issues for the site are prioritized and management ensures that resources are available to resolve the specific issues. This is defined and controlled by fleet standard procedure EN-DC-336.

The URT was initially developed at ANO and has been recognized as an industry good practice. In 2007 and the first half of 2008, the Unit Reliability Team was fully implemented at each EN Site.

Component Classification is part of the preventive maintenance strategy where high, low and non-critical components are identified. This is defined and controlled by EN-DC-153.

EN-DC-153 was revised in July 2008 to align EN classification criteria with current industry standards. Initial classification has been completed at each EN Site, and targeted re-classifications are in progress.

Single Point Vulnerability Studies identify a component whose failure results in a reactor trip, turbine trip, or loss of generation capacity. Once identified, these components go through a process to ensure that the risk of component failure is analyzed, and that the optimum strategy is applied to prevent failure. This process is defined and controlled by EN-DC-175. SPV Studies have been completed at each EN Site, and implementation of the mitigation strategies is in progress.

PM Optimization is a living process where PM Templates are used in conjunction with maintenance history, industry experience, vendor recommendations, maintenance feedback, and other inputs to arrive at the optimum PM Strategy for a specific component. This is defined and controlled by EN-DC-324, EN-WM-100, and EN-DC-335.

Vermont Yankee completed the initial PM Optimization in 1995 and 1996 with PMs established. PMs were re-reviewed to ensure that PMs were established for components classified as High Critical in May 2006 to August 2007.

System Health Reports are used as an indicator of the general effectiveness of the site programs and processes to maintain and improve system material condition and performance; it is a snap-shot-in-time of the health of the system. This is defined and controlled by EN-DC-143. In 2007, a web-based application was developed and implemented to provide standardized presentation of System Health Reports as described in EN-DC-143.

Predictive Monitoring is used to improve plant safety, equipment availability and reliability through early detection and analysis of equipment problems and degradation prior to equipment failure. Predictive monitoring provides input to just-in-time PM optimization, providing data necessary to perform the right maintenance at the proper frequency. This is defined and controlled by EN-DC-310. In 2007, Predictive Maintenance (PdM) gap analyses were completed for the EN fleet. Variations of existing technologies and methodologies between the sites were identified. Common vibration software and motor monitoring equipment have been purchased for the fleet. Standardized monitoring for major components is in progress.

2009 Fleet Equipment Reliability Improvement Focus Areas

The following list contains the focus areas for the Fleet Equipment Reliability Improvement initiative.

- Implementation of the Living PM Program
- Single-Point Vulnerability Mitigation
- Switchyard and Transformer Reliability Improvements
- Fuel Reliability
- Aging, Obsolescence, and Life Cycle Management
- Critical Spares
- Resolution of Important Equipment Issues
- Work Management
- Governance/Oversight for Equipment Reliability

Change Management

Entergy utilizes a Fleet Change Management process which is detailed in the Nuclear Management Manual EN-PL-155. The Change Management process provides guidance to the user on how to assess the impact of the change, who will be impacted, and what needs to be monitored to ensure that the change is effectively implemented.

NSA review shows varying degrees of detail in the Change Management Plans. In some cases, the NSA Team did not feel that the change plan was comprehensive or contained adequate detail to ensure effective or consistent implementation. Items lacking from some plans include:

- Detailed communication plan to inform targets and stakeholders
- Clear accountability/dates for key implementation milestones
- List of performance measurements that will indicate impact on the change
- Method to check for compliance after implementation
- Identification and mitigation strategy for risks (barriers)

In many cases change plans have not been created for new processes, such as standardizing procedures and the transition from EMPAC to Indus (the computerized maintenance management system), which has led to ineffective implementation of the change.

Conclusions

Current guidance from Entergy meets industry standards for a Fleet Organization; however, transition to the fleet concept has been slow and at times inconsistent at ENVY. Some examples are:

- ENVY did not start using the industry standard Equipment Reliability index until September of 2008.
- A standard Procedure Writer's Guide has not yet been created, and the transition to fleet processes and procedures has been slow.
- Peer Group progress was slow which led to Entergy creating Fleet Managers to provide improved leadership for the peer groups and their initiatives.

The application of change management should be improved and emphasized to ensure that new fleet-wide process changes underway at ENVY are implemented effectively, and that the workforce complies with the new process requirements.

1.2 Management Elements

1.2.1 Leadership, Setting Goals and Direction

Entergy Corporation has provided vision, mission and goals for the overall corporation. Corporate Business Plans are developed annually to set direction and establish goals for all Entergy Nuclear 'Fleet' plants, including ENVY, consistent with the vision and mission of the company. From these corporate level business plans, ENVY develops a Site Strategic Plan and associated Site Business Plan. Business Plans are further broken down into Action Plans for specific business areas or to address problem areas, such as the Action Plan to resolve high copper concentration in feedwater. NSA review of the Entergy Nuclear Corporate Business Plan for 2008-2009, Station Strategic and Business Plans for 2008, ENVY Nuclear Management Model, and selected Action Plans determined that they were consistent with industry standards.

Business Plan, Action Plans, and other goals set by management are cascaded down to managers, supervisors, and the workforce through the use of Personnel Performance Reviews (PPRs). These documents identify the goals for each individual, and performance to these goals is utilized to establish merit raises and incentive bonuses when specified station performance goals are achieved. The PPRs are reviewed annually or more frequently. A rating reliability process is also utilized for supervisors and above to ensure consistent station-wide application of the performance review process.

A variety of performance indicators is utilized to track and trend performance to established goals. Performance to high level goals (Level 1 goals) is reviewed monthly at the station Management Review Meeting (MRM) to ensure that station performance remains on track to meet established goals. Corporate senior management personnel attend these MRM meetings to provide high level management oversight of station performance. Level 2 & 3 performance indicators are available on the corporate intranet and are published and reviewed monthly. Lower level, individual goals, are the responsibility of managers and supervisors to track, trend and review with their personnel. NSA review of Level 1, 2, and 3 Performance Indicators, and attendance at various meetings, including the September 2008 MRM meeting, determined that these processes and practices were consistent with industry practices and were effective to set direction and goals for the station and staff.

There are indications that ENVY is making the transition from a single unit site to a member of a nuclear fleet, such as: capital funding allocation, fleet staffing numbers and organizational alignment, a standard computerized work planning program, standardized corporate level fleet procedures, and sharing of some resources from other Entergy stations. However, the overall direction and transition for ENVY to be part of a nuclear fleet has been slow. ENVY was purchased by Entergy in 2002. While some fleet elements have been implemented, as mentioned above, several initiatives have been slow to implement, such as organizational staffing and alignment, the computerized work management process, and fleet high level procedures, which were implemented about 18 months ago, 4 years after ENVY was purchased by Entergy.

The lack of Corporate direction for standardizing plant procedures has also hampered the procedure upgrade process at ENVY. NSA review of several ENVY procedures determined that, while they were technically correct, the current formatting did not readily support Human Performance (HU) tool usage. As a result, there have been plant events related to procedure quality or procedure use and adherence. Examples include the inadvertent trip of the 'A' Emergency Diesel Generator, and damage to a Service Water Pump lower motor guide bearing due to improper assembly.

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Conclusion

Once the full scope of procedure upgrades is identified, a detailed schedule will need to be developed to determine which procedures will be done and by when. It will also take a variety of resources (procedure writers, word processing, worker reviews, etc.) to make the necessary procedure upgrades. Additional funding from corporate or the station budget will have to be allocated for this project to succeed. In recognition of the potential costs and complexity of this project, NSA considers this a challenge to reliability.

In summary, it does not appear that Entergy has been rigorous or moved quickly in setting direction for and integration of ENVY into the fleet model. ENVY is managed in a more independent, traditional style than that seen at more fully integrated nuclear fleets in the industry. As a result, ENVY is a median performer in the nuclear industry with opportunities to leverage the benefits associated with fleet operations.

1.2.2 Performance Metrics

NSA reviewed various Key Performance Indicators (KPIs) at ENVY. The review included analysis of actual performance versus goals; gap analysis and identification of trends in performance data. Table 1 is the performance indicators typically used by ENVY in its monthly review meeting. The data spans 2006-2008.

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Table 1: Vermont Yankee MRM Data Selected Performance Indicators

SUBJECT	2006 (Dec)	2007 (Sept)	2008 (June)
Unit Capability (%)	x	x*	x
High Pressure System Unavailability (%)	x	x	x
RHR Aux Feedwater	x	x	x
Emergency AC Power	x	x	x
Chemistry Index	x	x	x
Collective Radiation Exposure	x	x	x
Unplanned Scrams (2yr)	x	x	x
Personnel Error Rate	x	x	x
Operator Work Arounds	x	x	x
Operator Burdens	x	x	x
Control Room Deficiencies Non-Outage	x	x	x
Control Room Deficiencies Outage	x	x	x
Non Outage Corrective Maintenance Backlog	x	x	x
Elective Maintenance Backlog	x	x	x
Other Maintenance	x	x	x
PM Overdue	x	x	x
PM Deferred	x	x	x
Non Outage Fluid Leaks	x	x	x
Engineering Request Backlog	x	x	x
Failure of Non-Run To Failure Components/month	x	x	x
Condition Report Inventory	x	x	x

*year end data END CONFIDENTIAL INFORMATION

Conclusion

For 2008, in addition to these metrics, ENVY highlights the PIs for capability factor, forced loss rate, chemistry, radiation exposure, and industrial safety accident rate as not meeting expectations on its overall performance index. Doing this calls attention to these areas and separates them out for gap analysis and follow through discussions to address underperforming goal areas. In addition, these PIs appear in their top 10 list and self-assessment sections, where the company further develops and analyzes, evaluates gaps, and develops performance improvement plans for these areas. There are also numerous examples of PIs that have been created to track watch areas, areas of importance, key project milestones, and changing business goals.

1.2.3 Organization and Staffing

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There was an active recruitment program to fill existing vacancies. A pipeline for recruitment of operations personnel has also been established to allow the hiring of seven or more additional personnel above the normal staffing limits, to allow for attrition, given the long training cycle for qualification. Most job vacancies appeared to be filled in a reasonable period of time with the exception of the engineering positions. The existing vacancies and ongoing turnover of engineering personnel have been more challenging for Entergy and ENVY to staff these positions. In response to the challenge in hiring engineering personnel, Entergy developed a guideline to offer accelerated employment offers for engineers (this has also been expanded to include operations personnel) during recruiting events. The use of recruitment and retention bonuses is also being utilized. NSA follow-up review of this area on November 18, 2008, determined that progress was being made to hire personnel and reduce the number of open positions. However, Entergy and ENVY will need to maintain vigorous recruitment and retention programs to effectively compete for personnel resources.

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A succession planning process is utilized to fill management positions for the Manager and Superintendent level positions. A Succession Planning book has been assembled that lists these key management positions, with personnel identified who are ready now, ready in 1-2 years, or ready in 3-5 years. NSA review of the Succession Planning book determined that most positions had personnel identified that could be moved into position should a vacancy occur.

Conclusion

The Succession Planning book meets industry standards, but there are opportunities for line management to more effectively utilize it.

1.2.4 Training

Introduction

The purpose of this evaluation is to assess the Training Department's capability to train and qualify operators, engineers, maintenance and technical personnel so they can operate and maintain the plant reliably for the duration of the 20-year life extension. Training performance, processes and organizational structure and staffing were evaluated for this assessment.

Organization and Staffing

The Training Managers at each Entergy plant report directly to a Fleet Director who is responsible for implementation of all training programs across the fleet. The Training Department has issued 24 fleet procedures for implementation of training. Training assessments at ENVY include personnel from other sites to draw on the experience of other programs.

Training for ENVY site personnel in all disciplines is conducted at the Station Training Center, which is located off-site in Brattleboro, VT. Each of the training programs has been evaluated and accredited to industry standards. Senior corporate management monitors the conduct of the Vermont Yankee training programs through the Executive Training Oversight Committee and periodic Management Review Meetings conducted at the site. The Training Oversight Committee (TOC), chaired by the Site Vice President, ensures that training goals and activities support performance improvement. Routine training needs are identified, communicated and scheduled by each discipline's Training Advisory Committee (TAC) and Training Review Groups (TRG). The TRGs, chaired by line management owners on a quarterly basis, establish continuing training schedules.

An interview with the Training Manager indicated that he felt the training programs, staff, facilities and associated departmental budget were adequate to support the overall training efforts. He also thought that the fleet model for staffing, which increases his staffing, would allow him to have consistent staffing with less reliance on contracted instructors.

In the maintenance area, I&C training is ongoing with 11 individuals in Initial Training. An additional I&C training class for 7 employees is planned to begin in February, 2009. Similarly, the 12 vacancies in engineering will require additional Engineering and Support Personnel training (ESP) classes. Currently, it is anticipated that the 2 ESP Initial Training classes scheduled per year, and the 2 existing instructors will be able to handle the training demand for ESP personnel. However, the large amount of ongoing training and planned additional training will require continued management focus to ensure that training quality and schedules are met.

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END CONFIDENTIAL INFORMATION Presently, the vacancies are being filled with experienced contract training instructors. It is planned to keep the contract instructors to support upcoming initial licensed operator classes. It is important that operations training has adequate resources to conduct scheduled training

Overall Training Performance

The Institute of Nuclear Power Operations (INPO) has established standards for nuclear training programs. Nuclear training programs are initially accredited to ensure that required industry standards are met. All accredited training programs are assessed against 6 performance objectives specified by

INPO. Each program has to demonstrate satisfactory performance in the 6 objectives every 4 years to have accreditation renewed for the next 4 years. The 6 objectives are:

1. training for performance improvement,
2. management of training processes and resources,
3. initial training and qualification,
4. continuing training,
5. conduct of training and trainee evaluation, and
6. training effectiveness evaluation.

This four year evaluation (Accreditation Self-Evaluation Report) is performed by both the utility and INPO. The results of the report are presented to an independent accreditation board to determine whether the programs are to have accreditation renewed. This evaluation provides a comprehensive assessment of the operations training programs and provides an industry standard for nuclear plant training programs.

ENVY uses industry standard Performance Indicators (PIs) to monitor training performance. There are PIs for the initial licensed operator program, the licensed operator requalification program, and the non-licensed operator program. Each program has a PI for each INPO training objective (6 objectives explained above). The PIs are designed to be a leading indicator to identify potential declining performance. For the second quarter of 2008, all training PIs for all training programs were 'Green' (meets goal). When performance issues are identified they are tracked in the "Analysis and Actions" section of the PI. When required, Condition Reports are entered into the corrective action system to resolve training issues.

There are 6 separate training programs that are administered by the operations training organization. The 6 training programs are for:

1. non-licensed operators,
2. reactor operators,
3. senior reactor operators,
4. shift managers,
5. continuing training - licensed personnel, and
6. shift technical advisors.

Oversight of the training programs is performed with multiple methods that conform to Entergy fleet procedures and processes.

Operator training programs in the US nuclear industry are closely monitored by independent organizations (NRC and INPO) in conjunction with internal oversight provided by a structured self - assessment process. ENVY routinely performs self-assessments of the operations training programs, and VY Quality Assurance audits the operations training programs for compliance with approved procedures and processes.

The NRC documents its routine evaluations of operator training in quarterly Integrated Inspection Reports. The NRC also periodically performs a comprehensive inspection of the Licensed Operator Requalification Program to ensure that it complies with specified NRC inspection criteria.

Three recent operations training self-assessments and one corporate operations training assessment were reviewed. The assessments were self-critical and identified both positive and negative aspects of the programs.

During the Entergy Corporate assessment conducted in April of 2008, the following comment was made. Observations of two control room crews were conducted to evaluate whether the crews had improved in previously identified performance areas. These areas included alarm response, use of slang, incomplete 3-part communications, not reporting alarms, and self-checking errors. Both observations concluded that crew performance met the requirements in EN-OP-115, Conduct of Operations and DP 0166, Operations Department Standards and no deficiencies were noted. One negative observation was that the quality of some exam questions was marginal.

A focused self-assessment of operations training programs was conducted in June of 2008 evaluating training performance with respect to the standards prescribed in ACAD 02-001, Objectives and Criteria for Accreditation of Training in the Nuclear Power Industry. Overall, the team concluded that the 6 objectives identified in ACAD 02-001 were being met by the Operations Training Programs and the programs are sound. It was also stated that students participating in all operations training programs are being provided training that is delivered in accordance with the Systematic Approach to Training (SAT) process. There was one area for improvement identified associated with exam security which was entered into the corrective action system.

NRC Integrated Inspection Reports routinely review various aspects of licensed operator requalification training. NRC Integrated Inspection Report (IR) 2008-002 performed an inspection and evaluation of a simulator-based licensed operator requalification exam in February of 2008. The crew was evaluated in the areas of clarity and formality of communications; ability to take timely actions; prioritization, interpretation, and verification of alarms; procedure usage; control board manipulations; and command and control. Crew performance in these areas was compared to the Instructor Guide for simulator scenario LOR-26-301 and Entergy management expectations and guidelines. The simulator control board configuration was compared to the actual control board configuration. The instructors were observed discussing identified weaknesses with the crew and/or individual crew members. There were no findings of significance identified.

A total of 11 NRC IRs performed during the period from 2002 to 2008 that assessed licensed operator simulator training, requalification simulator exams, and annual and biennial requalification exam administration were reviewed. There were no findings of significance identified in the IRs.

Training materials are routinely reviewed and updated. As part of the process, all plant engineering changes are reviewed by training to determine whether training materials need to be modified and/or whether simulator fidelity is impacted by the change. If training materials need to be revised due to an engineering change, a Training Evaluation and Action Request (TEAR) is generated to evaluate and implement the required changes. If a potential simulator change is identified, a Discrepancy Report (DR) is initiated and evaluated for applicability by the Simulator Review Board. The simulator and training support group maintains simulator fidelity by implementing required modifications to simulator hardware and software. The DR tracks the change from initiation through installation and testing. The processes used to evaluate engineering changes, to revise training materials and to make modifications to the simulator conform to industry standards. In the operations training Entergy corporate assessment conducted in April of 2007, it was identified that some simulator transient testing methodologies do not meet current industry practices. This was discussed by NSA with the Superintendent of Simulator and Training Support. It was explained that ENVY uses engineering judgment and design documentation to evaluate transient tests in lieu of using engineering analysis or actual plant data. Training is requesting funding in the 2009 budget to perform an engineering analysis for the simulator testing of 3 plant transients.

During an NSA observed simulator training exercise, the crew was assigned the task of reducing reactor power and then removing the main turbine and generator from service using Procedure OP 0105. The assigned control room operators routinely conducted control room panel walk-downs. Indicators were reviewed to identify changes in status lights, position indicating lights, trend recorders and equipment parameters as early indicators that action may be necessary to prevent unwanted plant conditions. Operators effectively monitored plant parameters. During the removal of the main turbine and generator from service, operators were observed to monitor multiple indicators and validate expected results. In addition, the Unit Supervisor was observed providing oversight of the plant evolution. The control room operators were observed to perform peer-checking whenever they were manipulating control switches. The conduct of this activity met operations standards.

The use of human performance tools, such as self-checking and procedure place-keeping overall met expectations. The use of 3-part communications did not consistently meet expectations; however, the deviations from the standard were either corrected on the spot or critiqued after the exercise was completed. It should be noted that the deviations were minor and there were no negative impacts to the operation of the plant in the simulator. The training instructors provided live-time coaching and the exercise was critiqued for lessons learned. The simulator training exercise met industry standards and expectations.

Conclusion

By using the results of the interviews, observations, and assessment reviews that were conducted, NSA concludes that ENVY training programs are meeting industry standards. Operations, maintenance and technical personnel are receiving the appropriate initial and continuing training to maintain proficiency in their respective areas of responsibility. There is one identified watch area associated with the large number of students to be trained and qualified in both the auxiliary and licensed operator initial training programs, I&C initial training, and 2 ESP classes in the next few years. This will require close management monitoring and oversight.

1.2.5 Continuous Improvement

Corrective Action Program

Assessment Scope and Objectives

One of the objectives of the ENVY Reliability Assessment was to evaluate the effectiveness of the Corrective Action program as per the requirements listed in Act 189. In this regard, Act 189 requires the following:

- What corrective action programs have been established for each of the seven plant systems listed in Act189?
- Have the corrective actions taken been properly integrated in the corrective action program?
- Have corrective actions been taken in a timely manner?
- Where recorded items have been deferred, have they been appropriately evaluated for risks and potential consequences of deferral and appropriately tracked while awaiting resolution?

Assessment Methodology

The evaluation team utilized multiple sources of input to evaluate the Corrective Action Program. These sources included:

- Reviews of procedures that provide instructions for the administration of the corrective action process.
- Interviews with Managers, Professionals and Department Process Improvement Coordinators (DPICs) responsible for administering the corrective action process.
- Reviews of the issue identification, issue categorization (A, B, C, or D - A being the most significant, D being the least significant), root cause/apparent cause analysis, corrective action identification and implementation, and effectiveness review processes.
- Reviews of the corrective action program metrics that are published on a monthly basis, including the timeliness and backlog of corrective action issues in Condition Reports (CRs).
- Reviews of quarterly trend reports that are used to proactively identify emerging issues.
- A review of the CRs identified in the corrective action database, Paperless Condition Reporting System (PCRS), the objective being to: (a) evaluate the effectiveness of the corrective action process; and, (b) determine if there were undesirable trends and/or repeat issues that had not been identified or effectively addressed.

The PCRS database reviews included:

- A review of Category A and B CRs generated during the past 2 years.
- A review of all of the CRs generated during the last 5 years for each of the systems recommended for vertical evaluation (Main Transformer, High Pressure Coolant Injection system, Condensate and Feedwater systems including the condenser, Residual Heat Removal system, Cooling Towers, Service Water system, and Condensate Storage system), as well as those generated regarding the Reactor Building Crane and Cable Separation.
- Reviews of CRs dealing with programmatic issues including Tagging, Procedure Compliance, the Work Order process, the Work Management process, and Human Performance.
- Reviews of NRC Inspection Reports focused on the Vermont Yankee Corrective Action program (Problem Identification and Resolution).

The above information sources were utilized to answer the questions posed by Act 189, and to determine the overall effectiveness of the Corrective Action program including issue identification, sensitivity to repeat events, root cause and apparent cause analysis quality, evaluation of extent of condition, as well as corrective action identification, implementation, and effectiveness.

Assessment Results

Entergy Procedure EN-LI-102, Corrective Action Process provides guidance for administration of the ENVY Corrective Action program. This procedure provides detailed instruction regarding issue identification and resolution, and the associated management oversight requirements.

Condition Reports (CRs) are utilized to identify and report a broad range of problems and areas for improvement, including equipment reliability/system issues and adverse conditions. (Adverse conditions are issues that detract from safe operation or impact nuclear safety, personnel safety, plant reliability, and/or regulatory compliance.) Each CR is entered into the PCRS, a computer program that tracks each issue from initiation to closure.

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Reviews of both the monthly CR metrics and PCRS database indicate that RCAs and ACEs are performed within 30 days or less, consistent with the requirements listed in EN-LI-102. Examination of individual RCAs and ACEs indicated that the organization utilizes structured analysis techniques, is sufficiently self-critical, considers both internal and external operating experience, and adequately evaluates extent of condition and cause. Identified corrective actions are consistent with the criteria provided in the guiding procedures, and are appropriate for and consistent with the significance level of the CR. For example, Category A CRs, receive attention from the CARB, which tends to increase the rigor and scope of both the cause analyses and corrective action plans. Corrective actions are effectively tracked to closure in PCRS, and are typically performed in a timely manner for all of the CR categories. However, these reviews also suggest that corrective actions dealing with programmatic issues such as Tagging, Procedure Compliance, and Work Management have not been as timely or effective as those dealing with purely technical issues.

Guidance for performing trending of CRs is provided in EN-LI-121. The goal of the trending process is to recognize issues at the precursor level so that underlying causes can be investigated and actions taken before significant issues or conditions occur. The CA&A Manager is responsible for assuring that the appropriate trend coding is performed at CR screening. The coding includes equipment deficiency and reliability issues. This individual is also responsible for communicating with and supporting the DPICs who perform trending, analyze trend data, identify emerging and adverse trends, and document lessons learned from trending. The DPICs also provide status updates on all open trend CRs as part of their quarterly reports. A comprehensive site report (developed from departmental reports) is published quarterly to make the above trending data and information available to management and other site personnel. These quarterly reports include information on new adverse trends, existing adverse trends, emerging trends, monitored trends, and resolved trends. The reports are reviewed and approved by the site's Safety Analysis Review Board (SARB). CRs are developed for adverse or emerging trends. The CRG reviews these CRs and assigns corrective actions as appropriate.

The PCRS database greatly enhances the site ability to identify and monitor emerging and/or adverse trends. The trend coding system includes a comprehensive set of codes and sub-codes for work groups, administrative controls, configuration controls, equipment deficiencies, work practices, keywords, equipment, human performance, as well as a number of INPO criteria. This system allows the user to identify a date range of interest and the sub-code(s) of interest, perform a search of the database, and view all of the relevant CRs, the corrective actions and their status, and most of the documentation associated with each CR. PCRS's trending capability is a ENVY Corrective Action program strength.

Although there is no specific plan in place to improve the Corrective Action program, a focused self-assessment is performed every two years to identify and eliminate program deficiencies. The self-assessments are augmented by snapshot assessments which target specific processes within the corrective action program. Actions identified as a result of the above assessment methodologies are tracked to closure in PCRS.

The U. S. Nuclear Regulatory Commission (NRC) has performed inspections relating to the identification and resolution of problems on a biennial basis. The last 3 inspections (August of 2003, September of 2005, and December of 2007) did not have any findings of significance, and concluded that the Corrective Action program was functioning as intended -- problems were being properly identified, evaluated, and corrected.

Two relatively recent incidents (the 2007 cooling tower collapse and the 2008 reactor building crane equipment problem) have received significant publicity and have raised some questions regarding the effectiveness of the ENVY Corrective Action program. The reviews of the PCRS indicate that there were precursor CRs and corrective actions for both of these systems/components. However, it also appears that the trending process did not successfully flag these systems for implementation of a more comprehensive analysis or corrective action plan. This appears to be due in part to the organizational focus on NSSS and nuclear safety-related systems and components. The ENVY organization has since increased its recognition that the trending process must look deeper, and be more sensitive to the fact that isolated issues may be indicative of more systemic problems – particularly when there are repeat events occurring in an aging facility. The ENVY organization recognized that structured management decision-making, via application of such tools as Organizational Decision Making Instructions (ODMIs) and other Human Performance tools, needs to be more broadly applied.

Conclusions

The NSA Team has concluded that the ENVY corrective action program meets expectations with minor exceptions. This conclusion is based on the following observations:

- Quality guidance is provided for the corrective action program via the implementing procedures.
- The organization implements the program in a manner consistent with the implementing procedures.
- Problem identification is effective and is inclusive of processes which heavily influence equipment reliability, e.g. Operations, Maintenance, Work Management, Equipment Reliability, Engineering and Training.
- Causal analyses are self-critical and generally effective.
- Corrective actions are generally comprehensive and lasting.
- Trend analyses are performed and are generally effective at identifying adverse emerging trends.
- Self-assessments are utilized to improve the corrective action program.
- Management has effective oversight of and involvement in the above processes.

The minor exceptions identified during the course of the assessment are listed below. These exceptions represent opportunities for ENVY to strengthen its corrective action program:

- Increased utilization of effectiveness reviews should be considered – particularly for CR's dealing with programmatic issues.
- The CR Interim and Periodic Review process should be considered for routine application to CR's dealing with programmatic issues.
- The Human Performance Error Review (HPER) process should be applied to any CR that has identified a Human Performance issue. This would strengthen both the corrective action and Human Performance programs.
- Consideration should be given to more thorough analysis of issues associated with non-NSSS and non-nuclear safety equipment/systems to assure consistency and an appropriate level of conservatism in decision making. Consideration should also be given to utilizing the Human Performance 'Devil's Advocate' tool with these systems on a more routine basis.
- CRs can be closed against a Work Order (WO), even if the WO is open and the required corrective action has not been completed. This leaves room for error (failure to complete a corrective action) and is not considered to be good industry practice.

Conclusions reached relative to the questions posed by Act 189 were as follows:

Question 1: What corrective action programs have been established for each of the seven plant systems listed in ACT 189?

Answer: All plant system issues are included in the corrective action program. Issues are entered as Condition Reports (CR's) and tracked to closure by a computerized system (PCRS). The type of causal analysis performed and corrective action taken is dependent on/appropriate to the significance of the issue. Trend analyses are preformed and corrective action taken with respect to any identified adverse trends.

Question 2: Have the corrective actions taken been properly integrated in the corrective action program?

Answer: All issues and corrective actions are entered in and tracked via the corrective action program (PCRS).

Question 3: Have corrective actions been taken in a timely manner?

Answer: There is significant emphasis on the timely completion of corrective actions. The due date guideline in EN-LI-102 is 180 days or less from the categorization date. Extensions require management approval. Escalated management approval levels are required for each due date extension that is requested. CR reviews indicate that due date extensions are uncommon.

Question 4: Where recorded items have been deferred, have they been appropriately evaluated for risks and potential consequences of deferral and appropriately tracked while awaiting resolution?

Answer: As indicated in question 3 above, deferral of corrective program action items requires management review and approval, and is rarely done. The few cases that were observed in the CR database were appropriately justified. All CRs are tracked in PCRS, and have a Responsible Manager to assure completion consistent with the requirements of EN-LI-102.

RCA of Main Transformer and Cooling Tower

Introduction

The NSA Team assessed ENVY's root cause assessment response to 2 significant sets of events that occurred at ENVY's facility in Vernon, Vermont. Additionally, through NSA's assessment of several other root cause efforts, NSA assessed ENVY's performance in conducting root cause efforts in general.

This assessment was conducted in the context of ENVY's Corrective Action program, and its subsequent contribution to station reliability going forward, i.e., beyond the present license expiration in 2012. NSA's assessment is presented in that regard.

Scope

The first event evaluated was the Main Transformer Electrical Fault/ Fire that occurred in June, 2004, and the second entailed a series of events that involved the station cooling towers during the period August 2007 – October 2008.

The June 2004 Electrical Fault/ Fire entailed the main transformer isophase bus work that, through a series of latent factors and fast-acting events, experienced a fire that resulted in the station being shut down while repairs were made to restore the components to operability.

The cooling towers experienced a series of events in 2007 and 2008 that necessitated several unplanned reductions in station power, and extended periods of time with one or both of the cooling towers out of service for inspection and/or repairs.

Root cause teams were commissioned to evaluate the: 1) mechanistic; 2) Organizational & Programmatic (O&P); and, 3) Human Performance (HP) components for each of the cooling tower events. The Electrical Fault/ Fire event had a mechanistic, but no O&P or HP evaluations. The NSA team's charge was to evaluate each of the primary and related root cause evaluation reports, apparent cause reports, their subsequent corrective actions, and to the extent feasible, the effectiveness of those corrective actions.

The Electrical Fault/Fire event entailed a primary Root Cause Analysis Report (RCAR), and numerous other related evaluations. The cooling tower events were assessed by ENVY staff with two prime RCARs, two prime ACE efforts, one supporting RCAR, and one supporting ACE. There were a number of lower level Condition Reports that entailed varying degrees of 'level of effort' to address remaining issues for each set of events.

Objectives

The over-arching objectives of the NSA evaluation of ENVY's root cause assessment process, through the lens of two very significant event responses, were to:

1. Provide an effectiveness review to understand the degree to which the station root cause efforts identify problems, and establish consistent corrective action processes to address issues of the same or similar events from recurring: and,
2. Render an opinion on the effectiveness of the root cause assessment process in terms of maintaining/ improving station reliability going beyond 2012.

Methodology and Evaluation Criteria

NSA's effort utilized several approaches and analytical tools. The NSA primary tools used to characterize the events as portrayed by the ENVY RCARs and ACEs were the comparative time line and event factor trees. NSA used 22 specific evaluation criteria to assess each set of evaluation reports, and employed complementary evaluation criteria to provide an integrated RCAR assessment for each event.

Overview and High Level Conclusions

Transformer Electrical Fault/Fire Event

At 6:40 AM on June 18, 2004, ENVY experienced a severe electrical fault that caused a generator trip, plant scram, fire, and declaration of an Unusual Event. There were no injuries associated with the event. The electrical fault and fire caused severe damage to the low voltage bushing box on top of the

Main Transformer, to the Generator PT Cabinet in the Turbine Building, and to the generator isophase bus duct itself. No significant damage occurred to the Main Transformer, the Unit Auxiliary Transformer, the Main Generator stator, or reactor systems.

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Independent Conclusions Regarding Transformer Electrical Fault/Fire Response

After review of evaluation documents and related procedures, and interviews with individuals, the NSA assessors concluded the following:

- Problem identification as documented in CRs appeared reasonably complete and accurate;
- Corrective actions, when considered in the aggregate, appeared to address the likely event causal factors as well as plant restoration but the basis for some corrective actions was not always clearly documented;
- While ENVY did not explicitly document consideration of either human performance deficiencies or procedural weaknesses as causal factors, corrective actions cited in the various evaluations appeared to address both human performance deficiencies and procedural weaknesses that the NSA reviewers consider as likely contributors to the event;
- Evaluation documents did not always fully explain the basis for conclusions; interviews were sometimes necessary for full understanding, e.g.:
  - ➢ The basis for concluding that the only foreign material (FME) involved in the isophase bus duct came from delamination of the B phase flexible electrical connection;
  - ➢ The impact of isophase bus duct cooling fans on the event; and,
  - o How criminal activity was ruled out as a potential factor;

- ENVY reviewed, but did not document, the aggregate ERO response to the event. The NSA assessors concluded this after reviewing EP procedure requirements in effect at the time that required such a review, interviewing the EP manager, and considering the approximately one dozen EP-related CRs initiated by event participants and the subsequent informal EP review.

Overall Conclusions Regarding Transformer Electrical Fault/Fire Response

NSA concluded that the ENVY response to the electrical fault/fire was generally adequate in terms of the corrective actions identified and documented in the various evaluations.

Evaluations associated with the electrical fault/fire tended to focus on mechanistic aspects (not behavioral or programmatic aspects) of the event. Full understanding of the event, and the ENVY response required:

- Detailed review of the 19 documents previously noted;
- Interviews of key people involved with the emergency response and/or evaluations; and,
- Use of tools beyond those used by ENVY to evaluate analytical completeness, and adequacy of corrective actions.

RCA of Cooling Towers

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Overall Conclusions Regarding the Cooling Tower Evaluations

NSA concluded that the ENVY response to the cooling tower series of events was generally adequate in terms of the corrective actions identified and documented in the various evaluations.

Evaluations associated with the cooling tower RCARs and ACEs tended to focus on mechanistic aspects; however, later efforts have placed more emphasis on behavioral and programmatic aspects of the event under review. Full understanding of the event, and the ENVY response, required a great deal of effort, including:

- Detailed review of the documents previously noted;
- Interviews with a number of individuals involved with the tower work and/or evaluations; and,
- Use of tools beyond those used by ENVY to evaluate analytical completeness and adequacy of corrective actions.

Overall Conclusion on RCA and ACE Processes

NSA concluded, in the context of its corrective action program and in support of station reliability, that ENVY's *Root Cause Assessment* and *Apparent Cause Assessment* processes are generally adequate in characterizing problems and providing corrective actions to address the factors leading to the events evaluated.

Self-Assessment

Assessment Methodology

This assessment utilized the following sources of input:

- Review of Entergy procedure EN-LI-104 Self-Assessment and Benchmark Process
- Reviews of the PCRS self-assessment database from January 1, 2000 to November 17, 2008

Assessment Results

Procedure EN-LI-104 provides comprehensive guidance regarding the performance of self-assessments. This guidance includes:

- Organizational responsibilities
- Scheduling requirements
- Performance requirements and methods
- Disposition of Areas for Improvement (AFIs), Performance Deficiencies, and Negative Observations
- Due date extension requirements
- Feedback methodology
- Review and closure requirements
- Reporting requirements
- Training and tools
- Effectiveness review requirements

A comprehensive set of tools is provided in the procedure in the form of attachments. These attachments include:

- Topic selection guidance
- Preparation timelines
- Preparation checklists
- Planning worksheets
- Data collection techniques and forms
- Problem development guidance
- Report format guidance
- Feedback forms

Key roles and responsibilities include the Vice-President of Oversight who has responsibility for the Corporate Assessment Process described in EN-QV-103; the Senior Assessment Review Board (SARB) that has oversight of the overall status of the site assessment program; Department Managers who establish assessment schedules and ensure that assessments and associated corrective actions are performed in a quality and timely manner; the Corrective Action & Assessment Manager who develops and maintains a consolidated site schedule of committed assessments, and assures that applicable information is entered into PCRS to support tracking and trending, and Departmental Performance Improvement Coordinators (DPICs) who assist Department Managers in the scoping, scheduling, and execution of department assessments.

Four Tier Levels of assessments are listed in PCRS:

- Tier Level 1: External Assessments, e.g. INPO
- Tier Level 2: Independent Assessments performed by the Corporate Assessment Group
- Tier Level 3: Assessments committed to be performed during the coming year by the site management/Department Managers
- Tier Level 4: Additional assessment activities, e.g. 'snapshots' performed during the year that are not part of the committed schedule

During the timeframe reviewed (January 1, 2000 – November 17, 2008) there were 10 Tier Level 1 assessments, 32 Tier Level 2 assessments, 200 Tier Level 3 assessments; and 374 Tier Level 4 assessments. PCRS was utilized to document 146 Effectiveness Reviews during the same timeframe.

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Conclusion

Reviews of the PCRS database described above indicate that management:

- Effectively tracks external and Corporate assessments to assure corrective actions satisfy quality and timeliness requirements (extension of deadlines requires management approval, and approval level escalates for each deadline extension requested)

- Utilizes focused self-assessments and snapshot assessments to address identified problem areas and adverse trends
- Ensures that self-assessments are objective and self-critical

These reviews also indicate that corrective actions dealing with programmatic issues have not been as timely or effective as those dealing with purely technical issues. This was an observation that was also made during the review of the Corrective Action Program.

Operating Experience

Assessment Methodology

This assessment utilized the following sources of input:

- EN-OE-100 Operating Experience Program
- Interviews with Manager, Corrective Action & Assessment Group
- Observation of OE utilization by workgroups

Assessment Results

ENVY does have an Operating Experience (OE) program. The overall process is driven through a Corporate OE program, with an OE Coordinator at the station. Daily searches of industry OE databases, regulatory databases, and Fleet and station OE events are reviewed by corporate personnel and fed into the process. Current procedure practice also requires that any station condition reports (CR) graded as level A or B and selected Cs be shared with the OE Coordinators. The OE Coordinators review all of this information, and then determine which work groups will receive the OE data.

Routine OE is sent to the station work groups and is utilized by them during their morning meetings. For higher level OEs a station CR is generated for assignment and tracking purposes. If OE is driven by a station CR, then an evaluation must be completed within 30 days, with corrective actions assigned specific due dates. Those categorized as routine Nuclear OE, have a 90-day clock to complete. The station's Paperless Condition Reporting System (PCRS) is used to track OE items and they are discussed by station management at the daily Condition Report Review Group (CRG) meeting. START CONFIDENTIAL INFORMATION

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During observation of workgroup and management meetings, NSA personnel verified that OE was discussed at these routine daily meetings. In addition, there was also active participation by craft personnel at the workgroup meetings, who shared their own personal OE experiences. Questioning of station personnel was also done by NSA personnel, which determined that station personnel were knowledgeable of recent OE events.

While routine nuclear OE has a well-defined process, work-specific OE is left to the supervisors to identify and collect. The OE data is available in the computer databases. However, leaving it to each individual supervisor can result in some inconsistent utilization of OE based on supervisory experience levels, time for OE research, and available time to focus on OE versus all of the competing priorities for first-line supervisory time. Industry practice at some nuclear stations is to include job-specific OE into individual Work Orders for ready reference by workforce personnel, thereby relieving the first-line supervisor of this administrative burden. In addition, there is an opportunity to utilize more Just-In-Time OE for specific higher risk activities.

Conclusions

In summary, ENVY does have a fleet-standard OE process similar to that found at other nuclear facilities. The process is utilized to collect, analyze and learn from industry operating experiences to reduce errors and improve performance. The overall process meets industry standards with exceptions as evidenced by the insufficient research for OE relative to cooling tower failures. After the fact, research by ENVY personnel determined that there was industry OE that could have been utilized to more effectively assess cooling tower problems before the partial collapse occurred in 2007.

1.2.6 Organizational Communications

Effective channels of communication are necessary in any large nuclear organization to keep personnel informed of such things as: company vision, mission, and goals; set direction for the company; communicate organizational and personnel changes; provide news relevant to the company; and, to communicate management standards and expectations. Typically, these lines of communication cascade downward from corporate through the entire organization.

As mentioned previously, Entergy Corporate has provided written communications for the vision, mission and goals for the overall corporation. Corporate Business Plans are then written and updated annually to set direction and goals for all fleet plants, including ENVY. ENVY, subsequently, develops a written Site Strategic Plan and associated Site Business Plan. Business Plans are further broken down into Action Plans for specific business areas or to address problem areas. The status of the higher level written plans and action items are typically reviewed monthly at the station Monthly Review Meeting (MRM). Corporate and station management personnel attend these meetings to obtain firsthand knowledge of the status of these items and to communicate any changes. Other forms of written communication exist between corporate and the station, including a company newspaper entitled "*inside Entergy-Vermont Yankee.*"

Verbal communication channels between Corporate and ENVY have been established through a formalized process of fleet telephone calls. Fleet station status calls are held daily to communicate the status of each station, share operating experiences and to coordinate the response and support for fleet and station issues. Special fleet phone calls are also held monthly to review the status of Peer Groups.

Other verbal lines of communication between corporate and station personnel include periodic skip-level meetings called Compliments and Concerns meetings, which are held between the Senior Vice President-Nuclear and station staff. At these meetings, the Senior Vice President can update personnel on what is happening in the company, and employees can ask questions directly to senior management. These meeting are intended to provide an open environment to foster better relationships between management and the employees, and to develop more transparency in the company.

Within the ENVY station itself, there is a wide range of written and verbal communication channels, some examples include: Monthly Management Review Meetings (MRMs) to review high level station performance metrics and station Update (All-Hands) meetings, which are held approximately every 6 weeks to share metrics; review human resource issues; and, discuss fleet initiatives. Leadership and Alignment meetings between station senior management and first line supervisors are held every 2 weeks to discuss safety status, focus areas for improvement, and general information sharing. Tail Gate Meetings between supervision and employees to share information from these leadership meetings include: use of written 'Point Papers' to allow supervisors to brief their personnel on important station and company matters; and, Pre-Job briefings that focus on job completion, safety, use of operating experiences, etc. A variety of daily/routine meetings for work management and operations to focus on corrective actions, continuous improvement, and outage readiness are conducted. These meetings also provide additional avenues for communication.

In addition to these written and verbal formats, a unique television communication tool called 'WVTY' is utilized to communicate news of interest to station personnel. Staffing of WVTY consists of volunteers from various station work groups, who act as reporters and anchor personnel. The reporters conduct infield interviews on station/company relevant issues, which are then reported on by anchor personnel. NSA review of some of these newscasts determined that the use of the WVTY concept is an effective communication tool to help keep station personnel informed on a range of issues of interest to the employees.

NSA team personnel observed utilization of many of these communication channels by attending various daily/routine meetings, including MRM meetings, operations focus meetings, work management meetings, condition report review meetings, pre-job briefings, outage safety fair, continuous improvement meetings, and review of a summary broadcast of WVTY. In addition, NSA Team members questioned station personnel while conducting infield observations, and found employees to be well informed on current issues.

Conclusions

Based upon NSA review of the number and quality of PIs, it was concluded that ENVY has a wide variety of effective communication channels, which meets industry standards.

1.3 Department Level Nuclear Plant Processes

1.3.1 Operations

Introduction

The NSA team members evaluated the effectiveness of the ENVY Plant Operations department performance, processes, organization and practices to identify issues which could impact plant reliability. The techniques described in the Introduction Section were used to gather pertinent information for the assessment.

The specific Plant Operations direction is provided as part of the recently developed Entergy Fleet Operations Standards contained in the following documents:

- Protective and Caution Tagging EN-OP-102
- Reactivity Management EN-OP-103
- Operability Determinations EN-OP-104
- Drywell Leakage EN-OP-109
- Operational Decision-Making Issues (ODMI) Process EN-OP-111
- Night and Standing Orders EN-OP-112
- Operations Interface with the Work Management Process On-Line EN-OP-114
- Conduct of Operations EN-OP-115
- Infrequently Performed Tests or Evolutions EN-OP-116
- Operations Assessments EN-OP-117

Organization and Staffing

The ENVY Operations Department is transitioning to the Entergy fleet standard organizational structure. The organizational changes at the manager/supervisor level were completed in 2007 to align with the fleet organizational structure. The transition for the remainder of the organization is planned to be completed in 2009. Roles and responsibilities for the Operations Department are contained EN-OP-115 Conduct of Operations. This procedure provides requirements and guidance for the performance of operations personnel. It also provides direction for compliance with procedural requirements, plant programs and regulatory requirements. The Conduct of Operations procedure is consistent with industry standards.

The department is led by the Manager of Operations who reports to the General Manager of Plant Operations. He directs the department through 3 Assistant Operations Managers. The Assistant Manager of Operations – Shift is responsible for operations shift personnel and their activities. This also includes the Work Control Center operations staff personnel. The Assistant Manager of Operations – Support is responsible for the administrative activities required for the department to function efficiently and effectively such as writing and revising procedures, addressing Condition Reports (CRs), and monitoring operations performance. It also includes tagging preparation. The Assistant Manager of Operations – Training is responsible for overseeing both the initial and re-qualification training programs and provides direction and oversight for the operations personnel in training and qualification programs. The shift crews are comprised of personnel in 4 qualification classifications. Senior Reactor Operators (SROs) are licensed operators (licensed by the NRC) assigned to the positions of Shift Manager and Shift Supervisor. The Shift Technical Advisor (STA) provides engineering technical support to the shift. Reactor operators (RO) are licensed operators that are responsible for operating the plant from the main control room. Auxiliary Operators (AO) are non-licensed operators that are responsible for conducting operational activities in the plant.

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The work hour rules for certain nuclear workers (such as operations shift personnel) are controlled by federal regulations. All nuclear sites will have to implement the new Fatigue Management Rule (10 CFR 26, Subpart I) as of October 1, 2009. This new Rule is more restrictive than the existing regulation. There is a fleet CR tracking all the actions that are being implemented for the Rule. There will be fleet implementing procedures and software to be used at each site. The software will be used to track compliance with the Rule. ENVY operations will have additional qualified licensed and auxiliary operators available when the current training classes are completed in April of 2009 and September of 2009 respectively. These personnel will be critical to meeting the new rule requirements. The implementation of this rule requires close management oversight and should be considered a watch area.

Overall, the actions taken to date for operations staffing have been effective. The staffing plan for the future is aggressive, but achievable. The operations training organization will be challenged with more frequent and larger than normal classes. This requires close oversight by management to ensure expected results are achieved. The implementation of the staffing plan is considered a watch area.

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Implementation of this staffing plan will require close monitoring and oversight of the training processes and programs. It will also require close monitoring and oversight of the transition of newly qualified operators to on-shift crew positions. The experience levels of on shift operations personnel now and in the immediate future is considered a watch area for operations management.

Procedures

Procedure Use - Procedures are used by operators for all operational activities conducted in the plant unless specified otherwise. It is a fundamental principle that operators understand the requirements for procedure use. Disciplined use of procedures is a vital part of error-free operations. Operators are expected to follow the procedures verbatim and in a step-by-step manner. If the operator cannot follow the procedure as written, the operator is then expected to stop, put the plant in a safe condition, and then get the procedure revised before proceeding. Fleet procedure EN-AD-102 Procedure Adherence and Level of Use specifies expectations and standards for procedure use.

Interviews with operations managers indicated that procedure use and compliance is a focus area for operations. A common cause analysis was performed for Condition Report CR 2008-02152 to evaluate the causes for recent human performance events. The following contributing causes were identified:

- implementation of the self verification process, and
- procedure use practices.

The following are 2 recommended corrective actions in the CR:

- develop a procedure use and adherence improvement plan, and
- develop a procedure quality enhancement program.

Although observations during this evaluation assessed procedure use and compliance as satisfactory, this area has been identified as a watch area based on interviews, review of assessments and CR reviews.

The following plant observations of procedure use and compliance all met expectations. Control Room Operators were observed in the main control room and in the simulator using procedures during the performance of activities associated with surveillance testing, reactivity maneuvers, removal of the main turbine/generator and annunciator response. The operators demonstrated an understanding of applicable procedural requirements and followed procedure steps in sequence. Place-keeping and peer-checking were appropriately performed. For these observations procedure use expectations were met. The performance of the monthly fire pump run was also observed in the plant, and the Auxiliary Operators that conducted the test also met the standards for procedure use. (See Plant Observations for more detail)

At ENVY all procedures are assigned to one of 3 classifications:

1. **Continuous Use** – Read each step of the procedure prior to performing that step, perform each step in the sequence specified, and where required, sign off each step as complete before proceeding to the next step.
2. **Reference Use** – Refer to a procedure periodically during the performance of an activity to confirm that all procedure segments of an activity have been performed, perform each step in the sequence specified and, where required, sign appropriate blocks to certify that all segments are completed. The procedure should be at the work location.
3. **Information Use** – An activity may be performed from memory, but the procedure is available for use as needed to ensure that all necessary steps are performed.

   This 3 level classification system clarifies expectations for when and how procedures are to be used. This is an industry standard used to provide consistency for procedure use.

Approximately 20 procedures were evaluated to determine whether the appropriate level of use as described above was assigned to each procedure. It was determined that the procedures reviewed were properly classified. Observations conducted in the control room and in the plant concluded that procedure level of use standards were met by operations.

Procedure Quality – Procedure quality is critical to ensure that all operators can perform their tasks error free. ENVY procedures are written as prescribed in AP0098 Procedure Writer's Guide which is a site procedure. Operations procedures are technically accurate and overall have been effectively implemented for years. The quality of the procedures with respect to human factoring, however, does not meet industry standards.

A review of selected ENVY operations procedures was conducted and compared to comparable procedures written to industry standards. Operating and surveillance procedures OP 2120 High Pressure Coolant Injection (HPCI) System, OP 4120 HPCI Surveillance, OP 2121 Reactor Core Isolation Cooling (RCIC) System, and OP 4121 RCIC Surveillance were compared to comparable industry procedures. Human factoring of ENVY operations procedures does not meet industry standards. The ENVY operating procedure for RCIC System contains both startup and shutdown procedures. The industry standard is to have individual procedures for both startup and shutdown. This results in a greater number of procedures but overall they are easier to use because the decision for what portion of the procedure to implement is less complex. The HPCI surveillance procedure covers multiple surveillance testing requirements. The industry standard normally divides this procedure into smaller self-contained surveillance procedures making the procedure less complex to implement.

The following CRs and assessments demonstrate that since 2004 procedure quality has been a continuing issue that has not been satisfactorily addressed:

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- Self assessment VTYLO-2007-0054 identified the following negative observations. First, it was stated that procedure writing is an ancillary duty rather than a dedicated position, and that procedure quality varies and is not consistent with station procedure AP0098 Procedure Writer's Guide. Second, Entergy has been slow to develop a fleet procedure writer's guide which would provide common procedure writing standards.
- CR-2008-02152 evaluated the causes for recent human performance events. One of the recommended corrective actions was to develop a procedure quality enhancement program.

Corrective actions specified in CRs and self-assessments associated with procedure quality from 2004 to the present have identified the need to implement a procedure improvement plan. This is a long-standing issue for which previous corrective actions have been ineffective. The General Manager Plant Operations presented the plan for the ENVY procedure upgrade during an interview. The plan includes revising the procedure writer's guide, developing a priority list for procedure upgrades and developing budget and resource estimates. The operations procedure upgrade program is in its early stage.

Overall, ENVY operations procedures are technically correct; however, a procedure upgrade initiative is required to align them with industry standards. The upgrade initiative will become more significant as the number of newly qualified on-shift operators increases. Less experienced operators rely more heavily on well-written procedures than more experienced personnel. One of the keys to error-free operations is procedure use and compliance. Having high quality procedures is a significant contributor to proper procedure use and compliance especially for less experienced operators. The completion of the procedure upgrade initiative is considered a watch area.

In-Plant Observations

Control Room Operations – Three control room observations were performed. Control room operator performance was observed in the areas of control board monitoring, annunciator response, 3-way communication, and crew pre-job briefing. Routine activities were performed in a controlled and deliberate manner using effective briefings, procedure use, self-checking, peer checking, and supervisory oversight. The control room atmosphere was controlled, formal and professional. Operators were observed performing control room panel walk-downs to monitor and assess control room indications. Shift Managers and Unit Supervisors were observed performing assessments to ensure the team understands activities, priorities and task risk level. The Unit Supervisor provided oversight for all control room activities. Operator performance met operations standards

The pre-job briefing for emergency diesel generator surveillance was performed using the prescribed checklist and included expected actions, priorities, contingencies, roles/responsibilities, risk level and use of tools and test equipment. The use of human performance tools such as 3-way communications, self-checking, peer checking, procedure use and adherence, and procedure place-keeping were observed. The communication between the main control room and the field operators was clear. The conduct of the surveillance met operations standards.

During a simulator training exercise the crew was assigned the task of reducing reactor power and then removing the main turbine and generator from service using procedure OP 0105. The assigned control room operators routinely conducted control room panel walk downs. Indicators were reviewed to identify changes in status lights, position indicating lights, trend recorders and equipment parameters as early indicators that action may be necessary to prevent unwanted plant conditions. Operators effectively monitored plant parameters. During the removal of the main turbine and generator from service operators were observed to monitor multiple indicators and validate that expected results were achieved. In addition, the Unit Supervisor was observed providing oversight of the plant evolution. The control room operators were observed to perform peer-checking when they were manipulating control switches. The conduct of this activity met operations standards.

Shift Brief – A shift brief is a meeting conducted for the oncoming operations shift to prepare themselves for their shift duties. The observed shift brief was conducted in accordance with AP 0152 Shift Turnover. The meeting was started with a brief safety message and a discussion of the expectations for procedure place-keeping (human performance focus). Each watch stander reported on their responsible area, which included the status of equipment out of service for maintenance, activities planned for the day and required compensatory actions. Industry operating experience was reviewed and discussed. The shift brief was assessed as comprehensive, effective and met operations standards.

Operator Rounds – Rounds are conducted each shift to monitor plant conditions, evaluate equipment performance, and take readings for evaluation and trending purposes. Operators assigned to each shift are required to record data from various locations throughout the plant. Rounds are conducted every shift in the main control room by control room operators and rounds are conducted every shift in the Turbine Building, Reactor Building, and Radwaste Building by Auxiliary Operators. Operators perform checks on structures, systems, components and controls within their assigned areas.

An observation was performed of the Turbine Building operator rounds. The Auxiliary Operator (AO) used a computerized data collection system. The AO was very conscientious in taking readings and performing appropriate housekeeping on the inspected equipment. Oil and water leaks were wiped up and the AO when questioned knew that the leaks had been appropriately identified in the corrective action system.

Overall the Turbine Building rounds were conducted in accordance with operations standards with one minor exception. Temperature and vibration monitoring of equipment are to be performed by placing a hand on the equipment per OP 0150 'Conduct of Operations and Operator Rounds'. Checking of temperature and vibration with the "hands on technique" was not observed. This technique is normally used by AOs when checking equipment performance.

There was an issue identified with temporary equipment in the plant. Items such as catch containments, temporary electrical power supplies, hoses and barrels containing liquids were observed with no tags to identify the responsible individual/group, date placed in the plant, and why it was placed in the plant. The SRO accompanying the AO on rounds stated that there was no formal tracking program for these items. This does not meet industry standards

Fire Pump Run – The performance of the Monthly Operational Check of Fire Pumps (VYOPF 4105.01) for the electric fire pump run was observed. The pre-job briefing was conducted by a control room operator with two Auxiliary Operators using the appropriate pre-job check list. The briefing was comprehensive and conducted professionally. Critical steps were identified by the Shift Supervisor and noted in the test. The following points were reviewed in the pre-job brief:

- Procedure identified as Level 1 – Continuous Use
- Procedure precautions and limitations
- Error likely situations
- Potential consequences and worst-case scenarios
- Contingency plans
- Expectations for procedure use and use of human performance tools
- Operating experience
- Critical steps

In the field the two Auxiliary Operators performing the test reviewed the procedure prior to performance. They posted a fire impairment notification in the pump room. The procedure was followed step-by-step and the appropriate self-checking, peer checking, and place keeping techniques were utilized. Three part communication was used and appropriate notifications to the control room were made. The test was performed in accordance with the procedural requirements. Test data was documented and verified to meet test requirements. The system was properly returned to service following the test. Overall the operators performed the test satisfactorily and operations standards were met.

Overall the conduct of operational activities observed in the main control room and in the plant was assessed as meeting operational standards and expectations.

Human Performance

Safety and Human Performance is one of 4 site focus areas identified in the 'Good to Great' program. Human Performance Program EN-HU-101 is the fleet-governing procedure for the implementation of the human performance program. Human Performance Tools EN-HU-102 is the procedure that delineates the specific human tools that are to be used by plant personnel to minimize errors.

Operations has had some human performance issues, most of which were categorized as low significance. When human performance issues are identified they are entered into the corrective actions system. For the more significant issues, corrective actions are developed and implemented. For the less significant issues, a trending analysis is performed. This assessment has identified two human performance watch areas: 1) improving performance in the preparation and application of tagging orders, and 2) improving procedure use and compliance. Both of these issues are discussed in their respective sections (Procedures and Overall Performance) in this report.

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The following event is included in this report because there were several breakdowns that are fundamental to error free plant operations in both the area of human performance and procedure quality and use. The plant appropriately performed a root cause which resulted in corrective actions to prevent recurrence. The corrective actions taken as a result of this event have had a positive impact on operations human performance.

Emergency Diesel Generator (EDG) Trip – On 2/12/07 operators performed the A EDG slow start operability test in accordance with OP 4126 (Diesel Generator Surveillance). Concurrent with this test, operations commenced a diesel fuel oil transfer test and discharge valve operability test in accordance with OP 4195 (Fuel Oil Transfer System Surveillance). In order to establish test conditions, an EDG must be run to allow the level in the EDG fuel oil day tank to decrease to less than or equal to 30 inches. Once this condition is established, the procedure directs the EDG to be secured and to throttle the fuel oil day tank inlet isolation to obtain a specified transfer pump discharge pressure. The operators failed to secure the EDG and instead of the fuel oil day tank inlet isolation valve being throttled the outlet valve was incorrectly throttled. Since the Operator did not see the expected response the valve was eventually throttled to the closed position. This isolated the oil supply to the operating EDG and it automatically tripped as designed.

The event was entered into the corrective action system (CR 2007-0483). A root cause analysis determined there were 2 root causes. First, operators failed to correctly execute the EDG and fuel oil transfer system surveillance procedures and allowed the A EDG to remain running during the transfer system surveillance. Second, the Auxiliary Operator failed to effectively self-check to ensure the correct valve was operated which caused the fuel oil supply to the operating EDG to be isolated.

In response to this event the following corrective actions were implemented:

1. High Intensity Training for human performance and operator fundamentals was conducted. This included expectations for self checking, peer checking, procedure adherence and pre-job walk downs. Also the EDG trip event was covered as Operating Experience.
2. Procedures OP 4126 and OP 4195 were revised to remove ambiguity.
3. The Operations Field Support Supervisor position was created to provide oversight and mentoring for auxiliary operators.
4. Expectations for pre-job brief package preparation were communicated to operations (NRC Inspection Report 2007-002).

Overall, the Operations Department meets expectations in the area of human performance as indicated by the monthly operations human performance indicator with two exceptions. These exceptions are procedure use and tagging process implementation. These are discussed in the Procedure section and Overall Departmental Performance section respectively. These issues are considered watch areas.

Overall Performance

The Operations Department overall performance routinely meets the conduct of operations standards for the areas reviewed in this reliability assessment. The exceptions to this are noted in this report and are summarized at the end of this section.

Plant Configuration – Plant configuration is routinely monitored by the NRC by performing system walk downs to verify system components are aligned in the proper position. Also the NRC conducts adverse weather evaluations for conditions such as cold weather, hot weather, high winds and flooding to ensure systems critical to plant operations are properly aligned and that procedurally required compensatory actions have been taken. Ten NRC Inspection Reports from 2001 through 2008 were reviewed to evaluate the results of system alignment walk downs and adverse weather evaluations. The following are examples typical of the types of walk downs that were completed:

Primary Containment Equipment Alignment Walk Down - The NRC performed a complete equipment alignment walk down of the primary containment pressure suppression vacuum breakers. The walk down was performed by comparing actual valve alignments to approved piping and instrumentation diagrams and to system lineups contained in OP 2115 "Primary Containment". There were no findings of significance (NRC Inspection Report 2007-002).

Adverse Weather Evaluation for Flooding – The NRC reviewed the plant's flood protection barriers and procedures for coping with external flooding events. External flooding information contained in Entergy's External Events Design Basis Document was compared to the required flooding actions delineated in OP 3127 Natural Phenomena. There were no findings of significance (NRC Inspection Report 2007-002).

The results of the 18 separate NRC evaluations in the 10 inspections had no findings of significance. This indicates that in the areas inspected system alignment and component configuration met expectations.

Non-routine and Off-Normal Operations - The performance of operating crews during non-routine plant operations and off-normal plant conditions (unexpected conditions) is a critical factor for safe and reliable plant operations. Several non-routine and off-normal plant operations that occurred over the last five years were reviewed with respect to operator performance. The following evaluations are provided to demonstrate operator performance.

- **Loss of Control Board Annunciators** – On 7/31/08 the control room operators identified that a portion of a control room alarm panel would not light when tested and the associated alarm horn did not sound. Operations appropriately determined that there were no actions required by Technical Specifications, Emergency Action Levels (EAL), or NRC reportability requirements. Operations established the appropriate compensatory actions. Operator response was evaluated as adequate. (NRC Inspection Report 2007-004)
- **Automatic Reactor Scram** - On 8/30/07 with the unit at 63% power all four Turbine Stop Valves (TSV) unexpectedly closed during troubleshooting activities. As a result an automatic scram occurred. It was evaluated that the operator response was in accordance with station procedures and training and that mitigating systems responded as expected. Operations appropriately evaluated Emergency Action Levels (EAL). (NRC Inspection Report 2007-004)
- **Trip of C Reactor Feed Pump (RFP)** – On 8/4/07 with the plant at 100% power control room operators responded to an unexpected trip of the C RFP. During the transient an automatic Reactor Recirculation Pump runback occurred that reduced power to approximately 80% as designed. The operators took prompt action to insert control rods to reduce reactor power to within the required limits in accordance with operating procedures. The actions taken in response to the RFP trip were evaluated as adequate. (NRC Inspection Report 2007-004)

Non-routine Evolutions – Operator performance during the following evolutions was observed:

1. Performance of high flow testing on the B Residual Heat Removal Service Water System on January 19, 2006,
2. Power reduction to approximately 50% to support a planned control rod sequence exchange on February 2, 2006, and
3. The first 5% power increase for the Extended Power Uprate on March 4, 2006. The operators performed adequately with respect to procedure compliance, Technical Specification compliance, and appropriate use of the corrective action system. (NRC Inspection Report 2006-002).

The above examples demonstrate satisfactory operating performance. Operations met standards and expectations for response to off-normal and non-routine evolutions.

Equipment Tagging - The tagging system is designed to primarily remove equipment from service to provide personnel safety. Tagging performance also can impact system and unit reliability in addition to personnel safety if errors in preparation and application are significant. The governing procedure EN-OP-102 is an Entergy fleet procedure and is utilized by ENVY for the implementation of the tagging process. Tagging orders are written, approved and applied by operations. Interviews with operations managers identified tagging as a focus area that requires attention due to the number and significance of mistakes that have occurred during the time period from 2005 continuing into 2008.

Operations self-assessment LO-VTYLO-2005-00360 evaluated worker behavior, emergent tagging activities occurring after work week milestones, tagging preparations, and effectiveness of previous corrective actions addressing previously identified issues. It was determined that a weakness existed in the tagging order preparation phase. It was also identified that management oversight of the tagging desk personnel was weak. These issues contributed to a larger number of tagging preparations errors than was acceptable.

In March of 2007 a ENVY Integrated Organizational Assessment Team identified in CR-2007-00723 that equipment tagging errors continue to occur in the preparation, review, approval and implementation of tagging orders and that previously identified corrective actions were not effective.

In 2008, CR 2008-00275 documented that corrective actions to address tagging issues were not effective. The apparent causes were: 1) inadequate program monitoring and not meeting tagging preparation and approval milestones; and, 2) insufficient operations manpower to meet identified goals and objectives. Missed milestones put additional pressure on the tagging office personnel and lead to an error likely situation. The lack of consistently having an adequate number of personnel in the tagging office also contributed to the number of tagging errors.

The above assessments and CRs identify a long-standing problem with tagging preparation and application. The following actions have been taken to improve tagging performance. The tagging office is now consistently staffed with the required number of personnel. The 5-year staffing plan addresses the short and long term manning requirements of the tagging office. A Senior Reactor Operator (SRO) has been assigned full time to the tagging office to manage tagging preparation and approval. The physical tagging office layout was modified to provide a more controlled environment which is more conducive to writing and approving tagging orders. The operations Field Support Supervisor has been assigned the responsibility of issuing tagging orders and overseeing tagging activities in the plant. Training has been conducted to improve operator knowledge and skills for tagging preparation. The cumulative effect has been a reduced number of tagging errors and overall improved tagging performance. Performance indicators (PI) are used by operations to monitor tagging performance. The PI for "Weekly Online Readiness Indicator" tracks the number of tagging orders that are prepared and approved as scheduled. Operations is routinely meeting its goal of all tagging orders prepared and approved two weeks prior to the execution of the work week (T-2). The Protective Tagging Index PI tracks the number and significance of tagging errors. The Protective Tagging PI indicates an improving trend from February 2008 through September 2008. The monthly goal for tagging index has been met from February 2008 through September 2008 (latest data) and the 6 month cumulative goal has been met for July, August, and September of 2008 (latest data) for the first time in 2007 and 2008.

There was one significant tagging issue associated with a control room annunciator power supply during the most recent Refuel Outage which indicates that tagging errors are still occurring. Although performance has improved and the corrective actions have had a positive performance impact, tagging preparation and application is a watch area. This warrants continued management oversight to ensure that good performance is sustained.

Conclusion:

Overall, operations meets the standards and expectations in areas assessed, however the reliability assessment team has identified the following four issues that are considered watch areas for future plant reliability:

1. The Operations Department is implementing a plan to fully staff the operations organization and fill existing vacancies. This is critical for operations to reliably operate the plant and to effectively transition to the new NRC work hour rule to be implemented by October 2009. This requires close oversight by operations management and is considered a watch area.
2. Procedure use and compliance by operators is considered a watch area.
3. Procedure quality has been identified as an area for improvement by operations for several years. The completion of the procedure upgrade project is important for error free operations. This is especially important with the large number of new operations personnel projected to assume on shift positions. This is considered a watch area.
4. The continuing issues with tagging preparation and application are considered a watch area.

1.3.2 Engineering

Introduction

The Engineering organization at ENVY has recently been restructured to align with the Entergy fleet model. It consists of four primary groups: System Engineering; Project Management; Design Engineering; and, Programs/Components. The Director of Engineering reports to the Site Vice President and also has a reporting relationship to the Corporate Engineering Vice-President. The site organization is responsible for the day-to-day engineering support of plant operations as well as the programs and modifications that support the long term operation of the plant. This is consistent with nuclear industry standard organizations for Nuclear Plant Engineering.

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Direction for the Engineering Department is provided from the Corporate Organization in the form of the Entergy Nuclear Operating Standards (ENOS) and Governance, Oversight, Execution and Support (GOES) models, discussed in the Corporate Evaluation Section 1.1. The specific Plant Engineering direction is provided as part of the recently developed Entergy fleet Engineering Standards:

- Conduct of Design Engineering EN-MS-S-016
- Conduct of Program and Component Engineering EN-MS-S-015
- Conduct of System Engineering EN-MS-S-011
- Conduct of Reactor Engineering EN-MS-S-018
- Director of Engineering 'Expectations'

The Entergy conduct of engineering standards meet industry standards and provide sufficient direction and guidance for the plant engineering functions with some minor exceptions discussed in the individual subsections.

The Conduct of Engineering documents are intended to provide guidance for consistent performance among engineering personnel, explanation of respective roles and responsibilities and basic/fundamental expectations for engineering personnel.

These standards have been recently developed and communicated to the Entergy fleet (between March and July of 2008). There were no comprehensive change management plans included with the rollout of these standards (Refer to the Management and Organizational Performance, Section 1.1 of this report for evaluation of change management process effectiveness at ENVY).

A number of personnel within the ENVY Engineering organization have high levels of plant experience which contributes to the historical overall good levels of plant and equipment reliability. However recent personnel turnover discussed in the "ER (Systems, Components/ Programs) Experience Levels and Training" section below will warrant management attention over the next few years. The system and component/program groups have been slow to move to the Entergy Fleet and industry standard ER processes.

Design Engineering

Design Engineering Organization and Staffing

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The experience levels within the Design engineering group are consistent with industry standard practice. Even with the retirements noted above, experience levels should still remain at a level that meets expectations with proper succession planning.

The training program for engineers within the design engineering group is consistent with industry standard practices. Interviews with design engineers and review of engineering documents confirmed the effectiveness of the training program.

Design Engineering Processes and Procedures

All design engineering work at ENVY is procedurally controlled. Compliance with these engineering procedures is monitored by various means, including Quality Assurance audits and Performance Indicators. Some of the more relevant procedures that govern the design engineering process are:

- Modifications EN-DC-115, Rev. 5, Engineering Change Development
- Modification Closure EN-DC-118, Rev. 2, Engineering Change Closure
- Configuration Management Program, EN-DC-105, Rev. 2
- Engineering Calculations Procedure, EN-DC-126, Rev. 1
- Procurement Procedure EN-DC-313, Rev. 2
- Margin Management Procedure, EN-DC-195, Rev. 2
- Licensing basis maintenance is controlled by Licensing Basis Document Change Procedure, EN-LI-113, Rev. 3

These procedures are aligned with the Entergy fleet model. They are complete in terms of maintenance of the design bases, margin management, and control of engineering work.

The quality of the design control procedures utilized by ENVY was consistent with those utilized in the industry by good performing organizations. They are comprehensive and provide an effective means of controlling the design and design basis of the plant. Procedure compliance meets expectations with minor exceptions.

Design Engineering Human Performance

Discussions with design engineers and managers confirmed a satisfactory level of knowledge of procedure and program requirements. Likewise, a satisfactory level of understanding of the importance of adherence to the original design basis and margin maintenance was demonstrated.

There were no instances noted of calculation errors identified by NRC Inspection Reports. However, there were some instances of procedure non-compliance associated with design engineering processes which were noted in NRC Inspection Reports. These instances appeared to be in line with industry numbers. In addition, there were no common causes identified, nor were there any adverse trends identified. A review of selected engineering documents and modification packages confirmed that the level of procedure compliance is consistent with industry standard practices.

Individual goals and objectives are set by the Manager of Design Engineering. Some of those goals and objectives are related to plant performance and reliability. Individual engineer's performance is tracked against those goals and objectives, is set annually and is reviewed quarterly. ENVY has instituted an individual performance bonus structure as a means of reinforcing management expectations about performance.

Human performance within the Design Engineering group is consistent with industry practices.

Design Engineering Department Performance

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END CONFIDENTIAL INFORMATION Similar situations have been observed across the U.S. commercial nuclear industry in the area of system and component engineering.

A review of the Engineering Technical training program was conducted based on:

- An interview with the Superintendent of Technical Training
- A review of the governing procedure for Technical Training EN-TQ-104
- And a review of the qualification cards for the HPCI Engineer and the FAC Program Engineer per ENN-TK-ESPG-042 Entergy Nuclear Northeast Engineering Qualification card, to ensure compliance with the approved procedures and processes

There were two observations made based on interviews and procedure reviews:

1. The Engineering Training organization is benefiting from the fleet through sharing of information and lessons learned as a result of the peer group process
2. In one random sample where the FAC engineer qualifications were evaluated, it was determined that the individual was qualified; however, the qualification card was not recently updated. The administration of the qualification documentation was not performed in a timely manner in accordance with the station procedure

Overall the Technical Training Program meets industry standards.

ER (Systems, Components and Programs) Processes and Procedures

Within the ENVY Nuclear Engineering organization, the System Engineering and Programs/Components Engineering group have primary responsibility for the ER Processes. The procedures utilized at ENVY to provide specific guidance for these ER processes are:

- Zero Tolerance for Equipment Failure Policy EN-PL-161
- System Monitoring Program EN-DC-159
- System Health Reports EN-DC-143
- Component Performance Monitoring EN-DC-325
- Predictive Maintenance Program EN-DC-310
- Preventive Maintenance Component Classification EN-DC-153
- Preventive Maintenance Program Implémentation EN-DC-324
- Long Term Asset Management EN-PL-170

The Entergy corporate ER organization in cooperation with the fleet-wide ER Peer Group have recently created these procedures and rolled them out fleet-wide. The recent (2008) emphasis on fleet wide standards for ER processes seems to be well designed by the corporate headquarters and the newly functioning ER Peer Groups. However, based on the NSA staff experience and knowledge of industry top quartile ER process performers, Entergy and ENVY have been slow to move to this fleet-wide standard approach for ER processes. Moving to this fleet-wide approach has significant advantages such as:

- Improved sharing of good practices by all sites to improve performance
- Improved transparency and therefore more effective assessment of ER processes
- Reduced costs
- Improved PM templates based on more collective knowledge and experience
- Enables better use of fleet-wide resources to support personnel moves and turnover effectiveness
- Improved prioritization and allocation of capital funding
- Improved availability of critical spares

The ENVY culture is based on being a 'single plant company'. Being part of a larger organization (fleet) and changes in process are a cultural change for many people. The Systems and Component/Programs Engineering Managers and Supervisors are very experienced and do not appear to fully appreciate the value of fleet ER process standardization which may be contributing to the slow movement to the new processes.

This judgment is based on various facts and observations by the NSA team members including:

- The overall Equipment reliability Index used at ENVY to indicate performance as part of the Monthly Performance Review Meeting (MRM) is unique. The industry ERWG created a standard ER Index in 2006 and not until September 2008 did ENVY use the industry ER Index as their standard at the MRM.
- Just recently (September 2008) an ER coordinator was designated at ENVY within the programs and components group. Industry good performers have had a single point accountability for this function for a few years.
- When many system, program and component engineering personnel were interviewed, the recognition of the industry focus on this standard approach was low and the emphasis on the new fleet-wide procedures was low.
- Many interviewees indicated that ENVY was already consistent with the fleet standard, historical plant performance has been good and there was no sense of urgency to move to the new procedures. There were instances where individuals were interpreting and applying the ER procedure guidance inconsistently for their systems and components. This was self-identified by ENVY.
- An overall ER Process procedure does not yet exist across Entergy or at ENVY that integrates each of the individual ER sub-process procedures. There is an initiative identified by the corporate ER Manager to create this document in the near future.

Scoping and Equipment Criticality Determination

The scoping and equipment criticality analysis for components at ENVY has been performed in accordance with procedure EN-DC-153. This initiative is complete and has been performed consistent with industry good practices as described in INPO AP-913 ER Process document.

As part of the system vertical slice assessments, the NSA team members randomly selected components to have the ENVY System Engineer describe the scoping and equipment criticality analysis applied. In each case an effective set of questions was applied and documented to determine component importance. These results were documented in the ENVY database. This meets industry good practices.

Maintenance Bases Creation and Management

ENVY has completed the development and documentation of the PM basis for all systems. An ENVY developed database is used to store and access the maintenance bases.

As part of the system vertical slice assessments, the NSA team members randomly selected components to have the ENVY System Engineer describe the PM task, frequency and basis for the selected component. In each case the PM's were clearly implemented in the Computerized Work Management system. The maintenance basis development and management process at ENVY meets industry good practices. However, ENVY has not yet converted to the Entergy fleet-wide standard database which limits sharing of information. An initiative is currently underway at ENVY to re-visit various PM templates within the database to compare these against industry standard PM templates.

System and Component Performance Monitoring

The Performance Monitoring Sub-process includes:

1. Component Health Monitoring:
   a. Individual component status reporting (including PDM technologies and related equipment condition indicators)
   b. Cross system component trending
2. Program Health Monitoring (such as In-Service Inspection, Flow Accelerated Corrosion, Snubbers) Health Monitoring
3. System Walk Downs and Notebooks
4. System Performance Monitoring
5. System Health Monitoring

Component Health Reporting

The Procedure that governs the comprehensive process for individual component health reporting at ENVY is EN-DC-325 'Component Performance Monitoring'. The components included in the component monitoring program are:

- Large Motors (> 200 horsepower)
- Transformers (Large oil filled)
- Circuit Breakers
- Valves (Motor & Air Operated)
- Check Valves
- Relief Valves
- Pumps
- Heat Exchangers

Basic expectations are described in this procedure. This procedure has just recently been issued as a fleet procedure (April 2008). The level of detail provided in this procedure is adequate and meets industry standards. The Predictive Maintenance Procedure EN-DC-310 describes the three condition monitoring technologies (vibration monitoring, infrared thermography and oil analysis) applications which are critical inputs to determining component condition. A component health status report is now created for numerous components as described in the procedure.

There is no standard software applied at ENVY or across the Entergy fleet that easily enables the capture of individual component health reporting. Unique Excel spreadsheets are used at ENVY to document component health. Capture of component case histories and sharing of component health and case histories is not easily enabled without a common software/database and process. The corporate Equipment Reliability organization has identified this as a future program enhancement initiative but no formal plans are in place.

The integration of the individual component health reports to the system health reports is not clearly defined in each of the respective procedures. For example in the component performance monitoring procedure there are no responsibilities identified for the System Engineer to specifically access and utilize the component health information as part of the System Health Reporting Process.

Cross system component health trending is also governed by Procedure EN-DC-325. This is an Entergy fleet standard and is consistent with industry standards. The respective component engineer periodically creates component health reports and identifies actions to address deviations from normal conditions.

Program Health Reporting

The ENVY Program/Component Engineering Group currently includes 16 Component/Program engineers managing various programs such as Flow Accelerated Corrosion, Appendix J, AOVs, In-Service Inspection, Snubbers, and Large Electric Motors. Individual procedures govern the specific program

As part of the system and technology focus areas, Section 2.11 of this report, the "large electric motor program" was evaluated in detail. A horizontal evaluation of this specific program was performed as a sample to determine the effectiveness of a cross-system component program. Also the Flow Accelerated Corrosion Program was evaluated in section 2.12 as it pertained to the Condensate Feedwater System.

System Walk-downs and Notebooks

Procedure EN-DC-178 provides the guidance for performing system walk-downs and notebooks. This procedure was recently created as a fleet standard (February 2008). No formal change plan was provided for the roll out of this new procedure.

As part of the individual system vertical slice reviews, the NSA team members observed various system walk-downs and reviewed system notebooks. Each System Engineer is required to perform these functions, but their methods were inconsistent and in some cases did not meet procedure compliance expectations.

System Performance Monitoring

The System Performance Monitoring Program is primarily governed by EN-DC-159 System Monitoring Program. The System Health Reporting EN-DC-143 process and the System Walk-down Process EN-DC-178 are also part of the Performance Monitoring program, which is consistent with industry standard practices. System Performance Monitoring Plans are required on all Category 1 systems.

The process for creating and implementing performance monitoring plans was evaluated as part of each system vertical slice review (refer to Sections 2.1 through 2.6). Based on how the monitoring plans were developed, implemented and maintained, inconsistencies from system to system were observed. In addition, based on interviews with System Engineers it was concluded that due to station challenges and personnel turnover, the ability to develop and maintain current monitoring plans in accordance with the applicable procedures was often a challenge. This deficiency was self-identified by ENVY in the recent system health assessment (performed September 22 through September 24, 2005) draft report.

System Health Reporting

The process for system health reporting at ENVY is governed by a recently created Entergy Fleet Procedure EN-DC-143 (August 2008). There was a change management plan created for the roll out of this procedure; however, the NSA team did not believe that the change plan contained adequate detail to ensure effective implementation. Items lacking from the plan included:

- Detailed communication plan
- Clear dates for implementation milestones
- List of performance measurements
- Identification and mitigation strategy for risks

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END CONFIDENTIAL INFORMATION The criteria, identified in EN-DC-143 to monitor system health, are consistent with industry good practices. Each system is designated in one of four condition categories (Red, Yellow, White, and Green) which are described in EN-DC-143.

The application of system health reporting was inconsistent and in some cases did not meet expectations of the new procedure. This deficiency was self-identified by ENVY in the recent system health assessment (performed September 22 through September 24, 2008) draft report. The database for capturing and retaining system health information was readily accessible and is a fleet-wide Entergy standard application.

System Health Reports from the six vertical systems were adequate for indicating current system conditions and actions; however, they were not created in a standard way and the quality varied. In addition, based on an interview with the System Engineering Manager, the average number of systems per System Engineer at ENVY ranges from 4 to 6 systems. This number depends on system size/complexity or operating issues (example-the Service Water System Engineer has one system). Managing more than five systems is a heavy workload compared to industry standards with a norm of 2 to 4 (refer to Section 1.4 Benchmarking Report). Some of this is due to six System Engineering vacancies which the station is currently attempting to fill. This has created an added burden for the existing staff to manage in the interim.

Preventive Maintenance (PM) Process Implementation

The implementation of Preventive Maintenance Program and maintenance basis was reviewed. ENVY personnel effectively entered required tasks into the Computerized Work Management System.

As part of the system vertical slice assessments, the NSA team members randomly selected components to have the ENVY System Engineer show that the selected PM tasks were performed at the appropriate frequency, that the work described was performed and that any records resulting from the PM activity were documented. In each case the review indicated that the process was working effectively.

Part of the PM Implementation process, PM deferrals, and the PM deferral process were evaluated. Before the RFO 27 outage, START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION were still not accomplished and were to be further deferred, some until the RFO 28 outage. The deferral paperwork was reviewed. The deferrals did not provide adequate justification for deferring these PMs or identify risk mitigating strategies to ensure reliability until the PM could be accomplished. The quantity of PM deferrals and the processing of deferrals does not meet industry standard practices.

Long-Term Asset Management (including aging and obsolescence)

A review of various ENVY asset management programs and processes was performed, based on a number of procedures that define elements of the program such as:

- EN-PL-170 Entergy Nuclear Asset Management Planning
- EN-DC-320 Identification and Processing of Obsolete Items (draft)
- EN-ENN-DC-176 Equipment Obsolescence Program
- EN-DC-143 System Health Reporting
- (Draft) Project Plan VY License Renewal Commitments and Implementation

The overall observation based on a review of the above procedures and discussions with System and Component Engineers, Managers, and Supervisors is that the LTAM program has not yet been fully implemented. A few specific observations include:

- The ENVY asset management process can be initiated through various entry points. One entry point is EN-DC-143 'System Health Reporting' which includes one of eight cornerstones called 'System Planning and Long Range Vulnerabilities Overview'. This is where the System Engineer identifies any short or long term plans for improving or maintaining system health. This section of the Health Report is intended to align with EN-PL-170 Entergy Nuclear Asset Management Planning. NSA concluded that this process has not been fully developed.
- The 3rd quarter Condensate Health Report does not provide or refer to a long term plan for the Condenser re-tubing effort which is a potential reliability issue for the station. This issue is discussed in more detail in the Condensate/Reactor Feedwater System section of this report.
- A review of the HPCI System Health Reports identified that 'no short or long term plans are needed for improving system health'. This does not meet industry standards. Long term plans should be developed for all major plant components to adequately address aging and obsolescence.
- The process/procedures that are in place to address aging and obsolescence issues at ENVY provide guidance for how to identify and manage obsolescence issues with respect to the 13 week work management process. This guidance is covered in EN-PL-170 Entergy Nuclear Asset Management Planning. This procedure is expected to be replaced by a fleet-

wide directive called EN-DC-320 Identification and Processing of Obsolete Items (draft at the time of this evaluation). The assessment efforts were not able to identify the change management plan for the implementation of this procedure.

The full implementation of the long-term asset management program is important to the long-term reliability of plant equipment and components and should remain a focus area for ENVY.

Aging Management Programs

ENVY's application for a license extension was submitted to the NRC on January 27, 2006. The application included a list of those safety-related structures, systems, and components subject to aging management reviews. The application was based on USNRC NUREG-1801, Volume 2, Revision 1, 'Generic Aging Lessons Learned (GALL) Report', dated September of 2005. The GALL Report contains the NRC's generic evaluation of existing plant aging management processes and documents the technical basis for determining where existing aging management programs are adequate without modification, and where existing programs should be augmented for extended operation.

As part of its application to the NRC for license renewal, ENVY committed to implement a comprehensive Aging Management Program, consistent with the GALL Report, by 2012. The details of the program are contained in the Vermont Yankee Nuclear Power Station License Renewal Application. For structures, systems, and components subject to aging management reviews, the application identified the materials, the environment the materials are exposed to, the aging effect the environment has on the materials, and the applicable Aging Management Programs, both existing and augmented, credited with managing and monitoring the aging effects. Each program includes preventive actions, parameters to be monitored, detection mechanisms of aging effects, monitoring and trending methodologies, acceptance criteria, corrective actions, confirmation processes, administrative controls, and associated operating experiences.

The NRC conducted a comprehensive audit of ENVY's License Renewal Application, including the Aging Management Programs, and issued its Safety Evaluation Report in March of 2007.

Interviews with plant personnel determined that they were familiar with the Aging Management Program commitments in general but not the program specifics. This is consistent with industry practices for future commitments.

ENVY recognizes the need to initiate an implementation plan, including appropriate training, so that all the elements of the Aging Management Program will be in place by the committed date of 2012. It is recommended that the existing Aging Management Programs credited in the application, be identified as such in order to ensure their integrity between now and 2012.

As part of the commitment to the NRC requirement for life extension, a draft Project Plan "ENVY License Renewal Commitments" identified 39 Long Range programs that will be required in support of the license renewal for implementation by 2012. The programs are listed below in Tables 2, 3, and 4, including the current overall status as indicated by ENVY.

Table 2: Programs Not Requiring Enhancement

17 Programs In Place Not Requiring Enhancement	
• BWR CRD Return Line Nozzle program	• BWR Feedwater Nozzle Program
• BWR Penetrations program	• BWR Stress Corrosion Cracking
• BWR Vessel ID Attachment Welds program	• Containment In-service Inspection Program
• Containment Leak Rate program	• Environmental Qualification
• Flow Accelerated Corrosion	• In-service Inspection
• Instrument Air program	• Oil Analysis program
• Masonry Wall program	• Reactor head Closure Studs program
• Water Chemistry Control- BWR Program	• Water Chemistry Control-Closed Cooling Water Chemistry
	• Vernon Hydroelectric Station

Table 3: Programs Requiring Enhancement

13 Programs Requiring Enhancement	
• Diesel Fuel Monitoring	• Water Chemistry Control-Aux Systems
• Fire protection	• Bolting Integrity
• Buried Piping	• Fatigue Monitoring
• Reactor Vessel Surveillance	• Service Water Integrity
• Structures Monitoring	• BWR Vessel Internals
• System Walk-down	• Fire Water System
	• Periodic Surveillance And Preventive Maintenance

Table 4: Programs Required

9 Programs Required	
• Heat exchanger program • Metal Enclosed Bused Inspection • Thermal Aging and Neutron Irradiation Embrittlement of Caustic Austenitic Stainless Steel (CASS) • Electrical Connections One –Time Inspection	• Non—EQ Insulated cables one time Inspection • Selective Leaching • Non-EQ Inaccessible Medium-Voltage cable • Non-EQ Instrumentation Circuits Tests Review • Bolted Cable Connections

The current Long Range Asset Management process is not yet fully developed. However, recognizing that the forthcoming decision regarding license renewal is a significant decision facing the continued operation of ENVY, the need going forward will be to establish a more comprehensive/integrated asset management and long range planning program.

Equipment Reliability Process Performance

At ENVY the current industry standard ER Index was reviewed and compared to the overall US Nuclear industry (91 units), the Entergy fleet (12 units) and a group of 'Sister Plants' selected by the State of Vermont. A benchmark study was performed as part of this initiative (refer to Section 1.4 of this report) that provides ER performance comparison data. START CONFIDENTIAL INFORMATION

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Analysis of all ER Index data included is included in the benchmark data report (in Section 1.4).

Conclusions

Engineering's staffing for each of the design, systems, and components/programs departments is a watch area due to the fact that attracting and retaining qualified, experienced personnel has been challenging at ENVY. Retirements and attrition over the next five years could impact the organization's effectiveness if not properly managed.

In Engineering at ENVY, all groups are now aligned with the Entergy fleet standardization model. In the design group, the procedures reviewed were fleet-wide procedures. Likewise, communication within the department is good and the exchange of information with other Entergy plants is good.

The overall Design Engineering group's performance is in line with industry good practices. There are occasional performance deficiencies, but the number and significance of those deficiencies are also consistent with industry standards and do not represent a pattern of performance deficiencies. The processes and procedures implemented by the design engineering group to manage modifications, margin, and configuration management are consistent with industry standards. The application of these procedures as evaluated in the system vertical slice reviews were applied adequately to ensure plant equipment/components are designed and installed to ensure plant reliability.

In the Equipment Reliability area, the standardization of processes with the Entergy fleet is progressing slowly.

- In addition, the overall ER Index performance should be a focus area for ENVY. START CONFIDENTIAL INFORMATION

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1.3.3 Maintenance

Introduction

Management direction for the Maintenance Dept. is contained in procedure EN-MA-101, "Conduct of Maintenance." This procedure provides requirements and guidance for the performance of work by Maintenance Department personnel. The procedure also provides instructions for complying with regulatory requirements, plant programs and procedures that control task performance in addition to, or in conjunction with, work specific procedures. Review of the procedure determined that it was consistent with industry standards for managing the Maintenance Dept. and for conducting maintenance activities.

Organization and Staffing

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END CONFIDENTIAL INFORMATION Overall, the organization is typical for the nuclear industry and staff size appears to be adequate to conduct station online maintenance activities, with supplemental personnel required during Refuel Outages.

Experience Levels and Training

To assess staff experience levels and organizational effectiveness, the NSA team conducted interviews with various levels of the Maintenance organization, including the Maintenance Department Manager, selected Section Superintendents and/or Supervisors, and first-line workers with less than 1 year to greater than 30 years of experience. Management experience levels were judged to be consistent with industry standards, with individuals having multiple years at ENVY or having experiences at multiple sites. An experienced and stable workforce was also observed to be in place for the Mechanical area.
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END CONFIDENTIAL INFORMATION In addition to the review of staffing and qualifications, NSA personnel also toured the maintenance facilities and shop areas. The facilities and shop areas were observed to be adequate to support the staff and their associated work activities.

The number of new employees in the I&C and Electrical sections has prompted ENVY to conduct training in both areas. In the I&C area 11 individuals were in Initial Training. An additional I&C training class is also planned to begin in February, 2009. However, it was noted that there is currently only one I&C instructor, with a vacancy for a second I&C instructor yet to be filled. Efforts were ongoing to fill this position, which is needed before the next class begins. In the Electrical area, most of the classroom training has been completed, so the Electricians were participating in the on-the-job training (OJT) and training performance evaluations (TPE) phases of training. NSA observation of a mixed class of maintenance department personnel for "Flagging and Robust Barriers," and an I&C class for review of Root Cause Analysis (RCA-06-0977), associated with single point vulnerabilities, determined that the training met the lesson plan objectives.

Procedures

Personnel with limited experience may require more detailed procedures and oversight. This has been recognized by ENVY Maintenance Management, which has embarked on a maintenance procedure upgrade project. This project was focused on upgrading approximately 250 maintenance procedures from early 1990's vintage formats, to new human-factored formats that provide additional clarity and remove human performance error traps.

The Maintenance Peer Group did benchmark industry peers to identify the latest industry standards for procedure formats. An attempt was made to adopt a standardized fleet format; however, fleet agreement on format could not be reached even with the support of the Peer Group sponsor. Subsequently, ENVY developed a plant format that utilized input from two peer stations (Limerick and Fitzpatrick) for I&C procedures. The procedure upgrade process, using the new format, has begun in the I&C area, with plans for the Electrical and Mechanical areas to follow.

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To evaluate improvements to the limited number of I&C procedures that have been revised, NSA personnel selected two I&C procedures and compared the before and after revisions of the following two procedures:

- OP-4307 Revision 11, 13 total pages; and OP-4307 Revision 17, 51 total pages.
- Op-4319 Revision 28, 10 total pages; and OP-4319, Revision 29, 30 total pages.

Observed changes included the addition of human performance factors, by eliminating data sheets and adding sign-off steps and data collection results within the text steps. There was also the addition of capitalization and bolding of all action words within the text steps. In general, the revised procedures were noted to be improved and consistent with industry accepted procedure formats. However, due to the slow progress in the procedures area, it is considered a watch area.

Observations of Work, Housekeeping and Foreign Material Exclusion Controls

Observations of on-line (non-outage) maintenance activities were conducted for the following on-line maintenance activities:

- Electrical-(OP 4210, Rev 54, "Maintenance and Surveillance of Lead Acid Storage Batteries").
- I&C-(OP 4335 Rev 20, "Reactor Building Ventilation Isolation and Standby Gas Treatment System Isolation Logic Bus Power Monitor Functional Test").
- I&C-(OP 4384, Rev 27, "Area Radiation Monitoring System Functional/Operating Cycle Test").
- Mechanical- Post Maintenance Test-(WM-105-01, 'C-1-1C Instrument Air Compressor Preventive Maintenance' per Work Order WO-51668010-01,").

These limited field observations focused on pre-job briefs, procedure usage and place-keeping, tool usage, interfaces with Operations personnel, response to alarms, use of 3-part communications, peer checking, housekeeping, foreign material exclusion controls and overall work quality. All observed on-line maintenance work was properly conducted in accordance with procedures, work order guidance and management expectations, with minor exceptions. Some of the minor exceptions included the formality and precision of 3 part communications and a poor work practice by I&C personnel, who operated panel switches while holding a pen in-hand. This practice could result in the pen inadvertently contacting and activating other adjacent switches located on the panel.

To further evaluate maintenance performance practices and standards across a broader time and range of activities, NSA observed housekeeping and Foreign Material Exclusion (FME) practices during three different time periods at ENVY. The time frames included about one month before the outage, during the outage, and about three weeks after the outage was over.

Overall, the plant cleanliness condition before the outage was good and consistent with the findings made during observations of the on-line maintenance activities. The plant was relatively clean with a minimal amount of material and tools lying around and would be characterized as meeting expectations. However, there was some dust and dirt on equipment at higher elevations (above six feet) which can be a source of contamination if not controlled properly.

During the outage housekeeping standards degraded and did not meet expectations. Contractors and ENVY personnel did not maintain proper control of their tools and equipment in designated areas, and housekeeping was observed to be poor. Specifically, work areas had debris, gloves, tools and material scattered everywhere, and in some cases it was impossible to tell which work area was associated with which job, or who owned them. In addition, Personnel Contamination Events exceeded site goals during the past two outages which could be related to poor housekeeping practices.

Three weeks after the outage conditions were not improved. Most contractors had already been released, but many areas were still in poor housekeeping condition and there was no indication when the situation would improve because of heavy vacation schedules in Maintenance. Five weeks after the outage, some areas in the plant improved. However, outside areas including the cooling tower did not meet industry standards.

Foreign Material Exclusion controls were also inconsistent. During the outage some piping had material covers taped over the ends, but in some cases the material was ripped and tape was peeling off, leading to potential debris inside piping. Some piping had rags or other material stuffed in the end of the pipe, in lieu of FME covers, which is a poor work practice.

A review of the CRs database identified that approximately 20 FME related events were documented during the outage. Many of these events would not meet industry standards or expectations for effective FME controls. The following are examples of FME events that were identified in and around the Reactor Vessel:

- In the annulus region of the reactor near Jet Pump 17
- On top of two different fuel bundles
- On the outer edge of the Reactor Vessel top guide
- In three locations on the grid between the core and the Reactor Vessel wall
- In a wedge area near Jet Pump 20

In addition, a Refuel Bridge pushbutton fell into the core and was retrieved, protective covers from the Steam Dryer inspection tool came off in the storage cavity, and a 'brownish-orange tinted residue' was found on the bottom of one of the Control Rod Blades.

Other FME issues identified during 2008 included:

- Large amounts of debris and trash in the Cooling Towers, including fasteners, studs, nuts, washers, soda cans, wood debris, and Fiberglass Reinforced Plastic (FRP)
- Sandy material at bottom elbows of new Service Water check valves
- FME around the Spent Fuel Pool (non-outage)

Only three of the CRs reviewed identified potential Human Performance issues associated with FME:

- The A Feed Pump Motor was installed during the outage without baffles making it more prone to FME intrusion
- Workers were cutting tubing in the drywell hatch without cleaning up shaving when they were finished
- An FME log review identified 171 items in an FME controlled area that were not on the FME log

The Housekeeping and FME Programs at ENVY do not meet industry standards. Plants with good FME programs have high standards for housekeeping and tight FME controls. ENVY's performance, especially during the outage when it is most critical, was less than adequate. The low number of CRs identifying poor worker practices associated with FME indicates an inadequate threshold for identifying FME and Housekeeping issues. This is a potential reliability issue because operating experience shows that poor housekeeping and FME controls can lead to fuel failures, equipment problems, and loss of generation. ENVY should develop an aggressive program to address these issues in order to prevent future problems.

Human Performance

Beyond the FME and housekeeping issues discussed above, there have been other human performance (HU) issues in the maintenance area. These issues are captured in Condition Reports, then trended and analyzed in the "Vermont Yankee Maintenance Department Quarterly Trend Reports." NSA review of the trend reports for the first and second quarters of 2008, and associated graphs going back to the first quarter of 2007, identified the following aggregate trend scores (average number of events per quarter) for the five quarters spanning 2007 and the first quarter of 2008 as shown in Table 5.

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Table 5: Aggregate Trend Scores

Area	Trend Score
Work Control	x
Procedure Non-Compliance	x
Failure to Self-Check	x
Unsafe Work Areas/Practices	x
Contamination/Radworker Violations	x
Housekeeping	x
Accidents, Near Misses/Injuries	x
Procedure Inadequate	x
Training Issues	x
FME Work Practices	x
Contractor Oversight	x

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The data shows that compliance to work schedule and controls; procedure non-compliance and failure to self check are the most prevalent issues, followed by unsafe work areas/practices. The HU events were spread across all of the work groups in Maintenance, but the majority of them occurred in Mechanical Maintenance. NSA personnel verified that a trend CR (CR-VTY-2008-1326) was initiated by ENVY to document and investigate this issue; results are still pending. In addition, while Contractor Oversight had a low number of events, it was stated by Maintenance Management that there is not a program, procedure or specific training for the Project Managers in the Maintenance Department, who oversee contractor work. This gap in procedural direction and personnel training and qualification is not consistent with strong performing nuclear plants. Also, the low number of CRs identifying poor worker practices associated with FME, considering the number of FME issues observed during the outage, indicates an inadequate threshold for identifying FME and Housekeeping issues

NSA review and analysis of maintenance HU issues identified that many were minor in nature, such as failure to wear some personnel protective equipment, missed place-keeping in procedures, and minor radworker practice issues. However, several HU events have been more serious, such as:

- The improper reading of a torque wrench, which was used on the High Pressure Coolant Injection system rupture disc flange by an experienced mechanic and a peer checker, resulted in a system leak (CR-VTY-2008-1229)
- The failure to properly follow procedure OP 5235 ‘AC and DC Motor Maintenance’ resulted in overheating and subsequent early failure of a Service Water Pump bearing (CR-VTY-2008-00601)
- Violation of Technical Specification 4.12.G.2, when Reactor Building Crane travel limiting mechanical stops were not properly installed on the crane trolley rails per procedure, prior to cask handling operations to prohibit cask travel over irradiated fuel assemblies (Note: actual crane operation over irradiated fuel did not occur) (CR-VTY-2008-02471)

Human Performance, especially related to work control, procedure quality and usage, and self-checking needs to be monitored closely. This is considered a watch area.

Maintenance Backlogs

NSA personnel reviewed, discussed and analyzed several key maintenance performance indicators (Corrective Maintenance; Elective Maintenance; Preventive Maintenance and "Other Maintenance" categories of maintenance backlogs, as well as "Rework"), to evaluate Maintenance Departmental performance and backlogs. Team members also attended various work management meetings to assess Maintenance department involvement with, and the overall effectiveness of the work planning and execution process at the station.

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END CONFIDENTIAL INFORMATION It was clear from NSA's review that there is an appropriately high level of focus on corrective maintenance. However, the elective maintenance and preventive maintenance (PM) deferral categories showed a slowly increasing upward trend. START CONFIDENTIAL INFORMATION

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Additional details relative to preventive maintenance issues are located in Work Control, Section 1.3.

Parts Obsolescence

NSA personnel also investigated if parts obsolescence contributed to the maintenance backlogs. Interviews with maintenance management personnel, attendance at work management meetings, and review of work management documents determined that parts obsolescence occasionally did present a challenge to work completion (e.g. GMAC power supplies). However, this was not a routine occurrence and all those interviewed believed that their existing processes to obtain or repair needed parts was successful in obtaining these items. Follow-up interviews with the Manager-Material, Purchasing and Contracts and a Configuration Management Supervisor confirmed that an effective process to locate and obtain obsolete parts or spares, using an industry shared database (rapidpartsmart.com) was in place. Contracts for the rebuilding of difficult to obtain items like GMAC power supplies were also being utilized. In addition, Engineering has a long range asset management plan to budget and repair/replace various station components.

Conclusions

From our review of the Maintenance area, NSA team members concluded that there is a documented process to control station maintenance activities. This process is effective to control and complete most station work activities, especially corrective maintenance. Overall, Maintenance meets the standards and expectations in the areas assessed, with the following exceptions:

- -Housekeeping and FME Controls
- -Loss of experienced personnel, and resultant additional training required
- -Slow progress on procedure upgrades
- -The amount of work categorized as 'elective' and 'other' maintenance and slow progress in reducing the backlogs
- -Accuracy of PM tracking data

Each of these areas is considered to be watch areas.

1.3.4 Work Control

Introduction

The purpose of this evaluation was to assess the Planning, Scheduling, and Outage Department's ability to identify, schedule, track, implement and maintain plant equipment during outage and non-outage periods. The scope included an assessment of the following key areas to identify issues which could impact plant reliability:

- Organization and staffing
- Overall departmental performance
- Online scheduling and execution performance
- Outage scheduling and execution performance

Organization and Staffing

The outage and work management functions are governed by a fleet Director and two senior manager positions. These fleet positions function through the peer groups to establish common processes and practices. The fleet has implemented 6 outage procedures and 5 fleet work management procedures. In addition, all sites are utilizing the Indus Asset Suite tool for work management. A standard set of outage milestones are used by all plants to plan and prepare for Refuel Outages.

ENVY uses an industry standard work management process. Work is identified and then scheduled according to its priority, availability of parts, and outage requirements. Work is scheduled up to 24 weeks in advance. Routine meetings begin in week T-15 and reviews occur at weeks T-15, T-6, and T-2. At week T-8 planning and walk downs should be complete and at T-6 parts available. Operations tagging orders should be prepared by week T-2.

The review meetings are run by one of the 4 Work Week Mangers (WWM). The Work Week Managers rotate responsibility such that each manger is responsible for work execution every 4 weeks. Observation of several status meetings indicated that all appropriate personnel were available and were prepared for the meeting. Open issues were challenged and there was good ownership by the WWM for their weeks.

Overall Work Management Performance

Based upon NSA observations and interviews, work execution is a challenge for all groups involved. Although all preliminary work is 'complete' it is difficult for all involved to implement the work. Maintenance Supervisors and Technicians indicated during interviews that the scheduling process is a trouble spot because the different Operations Department's Shift Managers have different criteria on what equipment will be available for maintenance. One estimate is that 10% to 20% of I&C work is limited each week due to operations department changes in available equipment. The Technicians believe the schedule could work if it didn't change so often.

Work Package quality is an issue with Maintenance Supervisors and Technicians especially with the completeness of packages. However, a QA Audit in June, 2008 identified that "there has been minimal feedback from the shop for T-8 acceptance reviews and there has been minimal post job feedback from the mechanical shop." Feedback is necessary to improve work package quality, which becomes more important as experience levels decrease in Maintenance.

In addition, the Work Management process was converted from the EMPAC system to the Indus Asset Suite (IAS) in November, 2006 because this was the Entergy fleet standard. Personnel interviews identified many issues with IAS and frequent history searches back to the old system because IAS is deemed too difficult to use. This indicates inadequate use of the Change Management process to implement a major change such as migration to a new Work Management system. More than 18 months after introduction of the new system there are still many issues with its use and usefulness and it is not clear when these issues will be resolved.

Preventive Maintenance Issues

During the assessment period 4 PM's went overdue. This was attributed by ENVY personnel as a failure to properly schedule the work, rather than not performing the work. Further review by NSA personnel, however, identified that the number of overdue preventive maintenance (PM) tasks identified on a recent "Backlog Trending Report" did not agree with the data published in the MRM book. Investigation determined that the Backlog Trend Report is inaccurate relative to PM tracking. This shortcoming was reported as being known to exist for some time, but has not been rigorously pursued for resolution by management. In addition, during a work week management T-6 meeting it was noted by NSA personnel that two PM's were scheduled to go over due on November 26, 2008, but weren't scheduled to be worked until the week of December 8, 2008. When this was pointed out to the Work Week Manager, he stated that the work week schedule document had errors in PM tracking due to data migration problems, but that he would investigate to ensure these two PMs wouldn't go overdue. Increased management attention and organizational focus to improve PM tracking is warranted and is considered a watch area.

Refuel Outages 25, 26, and 27

ENVY performs Refuel Outages every 18 months. Refuel Outage (RFO) schedule development has been conducted in-house following RFO-21, which had a 34 day outage schedule. Prior to RFO-21, contractors had been involved with outage scheduling. Since taking scheduling in-house, regular Refuel Outages have averaged less than 22 days with the exception of RFO-24, which was a 30-day Extended Power Uprate modification outage. During this outage ENVY replaced the High Pressure Turbine and four High Pressure Feedwater Heaters and rewound the Generator Stator and Field.

The Public Oversight Panel had questions relative to the work scope that was identified and scheduled to be performed in recent outages versus the actual work performed. To assess this concern, NSA team members observed and evaluated the recent 2008 outage, and interviewed personnel and reviewed records associated with the previous two Refuel Outages. During this review NSA personnel evaluated if there were any indications that work scope had been inappropriately removed to control costs or to achieve shorter outage durations. Either of these situations could cause or contribute to potential reliability issues.

As part of this assessment, NSA personnel interviewed station personnel and reviewed a variety of historical documents associated with two previous Refuel Outages:

RFO-25 Documents:

- RFO 25 Post Outage Report
- Scope Change Index for RFO-25
- Deferred 2005 (Outage) Work Orders
- Selected Scope Change Request Forms
- ENVY QA Surveillance Report QS-2005-VY-024, 'Outage Preparation through Shutdown'
- ENVY QA Surveillance Report QS-2005-VY-025, 'Assessment from Reactor Shutdown through Outage Week One'
- ENVY Oversight Observation Checklists October 29, 2005 through November 05, 2005

RFO-26 Documents:

- RFO 26 Post Outage Report
- Scope Change Index for RFO-26, which now included Deferred Outage Work Orders 2006
- Selected Scope Change Request Forms
- Corporate QA 'Master Outage Plan,' Rev 1 dated January 29, 2007
- ENVY QA Report '2-Day Rollup May 13-14, 2007'

For RFO 27, which was conducted during the assessment period, NSA personnel attended outage meetings, reviewed outage schedules and documents, interviewed personnel, and observed in-field outage activities. These outage observations and document reviews focused on the outage scope, schedule, work quality, and assessed schedule/scope changes to evaluate the process for and appropriateness of any changes. An assessment of the three outages is as follows:

Review of RFO 25

Outage performance compared to outage goals is reflected in the following Table 6.

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Table 6: Outage Performance Compared to Outage Goals – RFO 25

Goal	RFO-25 Goal	RFO-25 Actual
OSHA Recordable Accident	x	x
Lost Time Accident	x	x
First-Aids	x	x
Human Performance Event	x	x
Shutdown Safety Margin	x	x
Accumulated Dose	x	x
Personnel Contamination Events	x	x
Outage Duration	x	x

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Major Projects and Modifications Scope:

- 10 year Low Pressure Turbine Inspections
- Capacitor Bank Tie-In and Testing
- Generator Relay Protection Modifications
- Condensate Demineralizers Panel and Controls Upgrade
- V10-26B RHR Containment Spray Valve Replacement
- Drywell Steam Line Strain Gage Installation
- Main Steam Branch/Bypass Chest Leak-Off Line Replacements
- 24 VDC ECCS and Appendix 'R' Power Supply Modifications
- Condenser Tie Rod and Tube Impingement Shield Replacements

Major Maintenance Scope:

- 115KV and 345 KV Disconnect and Insulator Inspections
- 115KV and 345KV Breaker & Line function Testing
- 4KV Breaker Remote Racking Operations Modifications
- Bus 8 and 11 Supply Breaker Replacements
- MCC-89A and B Switchgear Modifications
- DC-1 Panel and Breaker Testing
- MG-1-1A & MG-1-1B Recirculation Motor Generator Rotor Cleaning
- MG-1-1A & MG-1-1B Fluid Coupler Inspections/Oil Change
- P-18-1A Recirculation Pump Seal Replacement
- P-1-1C Feed Pump Overhaul
- RRU-5 and 7 Service Water Valve Replacements
- Main Steam Relief/Safety Valve Replacements
- RPS HFA Relay Contact Change Out
- Instrument BOP Calibrations
- Instrument Logic Testing
- Cooling Tower Structural Inspections & Repairs
- Circulating Water CW-126" Line Repairs

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Since RFO-25 was completed about 36 months ago, direct observation of work practices and work quality could not be conducted as part of the NSA review. Following completion of the outage, the unit ran breaker-to-breaker for 547 days. It can be inferred from this long run that the 'right' work had been identified and properly completed during RFO-25 to allow a reliable run of this duration.

Review of RFO 26

Outage performance compared to outage goals is reflected in the following Table 7.

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Table 7: Outage Performance Compared to Outage Goals-RFO 26

Goal	RFO-26 Goal	RFO-26 Actual
OSHA Recordable Accident	x	x
Lost Time Accident	x	x
First-Aids	x	x
Human Performance Event	x	x
Shutdown Safety Margin	x	x
Accumulated Dose	x	x
Personnel Contamination Events	x	x
Outage Duration	x	x

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Major Projects and Modifications Scope:

- Low Pressure Turbine Repairs
- Noble Metals Re-application
- Core Spray Pipe Weld Clamp Repair
- Steam Dryer Inspections
- In Vessel Inspections
- HPCI Exhaust Check Valve V23-3 Replacement
- P-7-1D service Water Pump Check and Isolation Valve Replacements
- P-1-1C Feed Pump Motor Replacement
- 345KV Breaker 1T Replacement
- Switchyard Insulator Replacements
- De-Icing Gate Replacement

Major Maintenance Scope:

- 115KV and 345KV Disconnect and Insulator Inspections
- 115KV and 345KV Breaker and Line Function Testing
- Bus 1, 3, 5A and 8 Switchgear Inspections
- 4KV Breaker Remote Racking Operation Modifications
- T-8-1A Leak Repair
- T-6-1A, T-8-1A, T-11-1A and T-12-1A, 480 V Transformer Testing
- P-18-1B Recirculation Pump Seal Replacement
- P-1-1A Feed Pump Seal Replacement
- P-2-1B Condensate Pump Motor Replacement
- P-7-1D Service Water Pump and Motor Replacement
- E-6-1B Condenser-Turbine Expansion Joint Replacement
- Condenser Structural Support Weld Repairs
- Main Steam Relief/Safety Valve Replacements
- Instrument BOP Calibrations
- Instrument Logic Testing
- Cooling Tower Structural Inspections and Repairs
- Deep Basin Draining and Cleaning

For RFO-26, ENVY transitioned from EMPAC to Indus Asset Suite (IAS) as its software for maintenance planning and control. Use of the IAS software resulted in the increase in the number of outage work task activities, as work order steps were broken out into single actions rather than multiple actions in one step. For example, using EMPAC a work order step for support may state: Step 1: Erect and remove scaffold. Therefore, using IAS, the same activity would be two separate work order tasks: Step 1: Erect scaffold; and Step 2: Remove scaffold. Therefore, the number of work activities for RFO-26 may appear higher; however this is mainly due to a change in software.

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In summary for RFO-25 and RFO-26, both of these outages included extensive modification and maintenance work. Some of the work was associated with the six systems selected for detailed review during the reliability assessment of ENVY. The outages were both of short duration, with some additional time (approximately 1-1.5 days) required to address added and emergent scope; and in both outages scope additions outnumbered scope deletions. NSA review of the scope additions and deletions determined that there is a procedurally driven process to control scope changes and document approvals. Based upon NSA review of records and personnel interviews, it was determined that this process was followed during both outages. In addition, NSA review of the independent QA Reports associated with RFO-26 did not identify any reference to work that was inappropriately removed from the outage scope.

Following RFO-25, the unit ran breaker to breaker for 18 months. Following RFO-26, the unit experienced an automatic scram and several down power periods described above. These were determined not to be attributed to work that had been conducted or omitted during the outage, but did result in a loss of power output and impacted unit reliability. Based upon our view of RFO-25 and RFO-26 work activities, NSA concluded that while outage durations were short, work had been properly scoped, scheduled, controlled and completed. NSA review of the independent QA reports associated with these past outages, determined that QA did not identify any findings concerning the inappropriate removal of work activities or reduction of outage scope.

Review of RFO 27

Outage performance compared to outage goals is reflected in the following Table 8.

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Table 8: Outage Performance Compared to Outage Goals RFO 27

Goal	RFO-27 Goal	RFO-27 Actual
OSHA Recordable Accident	x	x
Lost Time Accident	x	x
First-Aids	x	x
Human Performance Event	x	x
Shutdown Safety Margin	x	x
Accumulated Dose	x	x
Personnel Contamination Events	x	x
Outage Duration	x	x

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Major Projects and Modifications Scope:

- Turbine Valve Inspections and Overhauls (Control, Stop, Bypass and CIVs)
- CRD Replacements (6)
- LPRM Replacements (7)
- Steam Dryer Inspections
- In Vessel Inspections

- Vessel Nozzle ISI Inspections
- VAC Inspections
- 'B' Diesel Generator Service Water Piping Replacement and Isolation Valve Installation
- P-7-1A and B Service Water Pump Check and Isolation Valve Replacements
- P-1-1A Feed Pump Motor Replacement
- Isophase Bus Duct Inspection Viewing Ports
- T-10-1A and Condensate Pump 4KV Cable Replacements
- HPCI and RCIC Vent Valve Hot Tap Installations

Major Maintenance Scope:

- 115KV and 345KV Disconnect and Insulator Inspections
- 115KV and 345KV Breaker and Line Function Testing
- Startup Transformer Hi Side Bushing Replacements
- Neutron Monitoring Battery Replacements
- Main Station and UPS Battery Performance Discharge Tests
- 4KV Breaker Remote Racking Operation Modifications
- P-18-1A Recirc Pump Seal Replacement
- P-1-1B and C Feed Pump Seal Replacements
- P-2-1C Condensate Pump Motor and Mechanical Seal Replacements
- P-2-1A Condensate Pump Bushing and Mechanical Seal Replacement
- Circulating Water Butterfly Valve Repairs (5 Valves)
- Circulating Water Discharge line Inspections
- Condenser Waterbox Tube Plug Replacements
- 'B' Recirculation Motor Generator Fluid Coupler Inspections
- Main Steam Relief/Safety Valve Replacements
- AOV, MOV and Mechanical Valve Inspections, Testing and Repairs
- Instrument BOP Calibrations
- Instrument Logic Testing
- Meter & Relay Calibrations

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The scope change process used during RFO-27 was identical to the one used for RFOs 25 and 26. NSA attended outage and scope control meetings during the outage. Scope additions and deletions were appropriately reviewed and authorized by plant management.

During RFO-27, NSA personnel observed steam dryer inspection work performed by AREVA. AREVA used state-of-the-art technology to inspect all areas of the dryer and record any indications that required analysis. Previous cracks were inspected and new cracks identified. General Electric reviewed the findings before unit start-up and determined that the indications were acceptable for operation.

Based upon our review of RFO-25, RFO-26, and RFO-27 work activities, NSA concluded that while outage durations were short, in general, the appropriate work had been properly scoped, scheduled, controlled and completed. NSA review of the independent QA reports associated with these past outages, determined that QA did not identify any findings concerning the inappropriate removal of work activities or reduction of outage scope.

Conclusions

The Work Management process is an industry standard which has been in use for many years and is an Entergy standard process. ENVY is making progress in implementing a standard process but is still not meeting expectations in some areas. A Work Management Academy, started in June of 2008, was intended to improve understanding and adherence to Work Management processes; however, communications, planning, work package quality, and schedule adherence continue to challenge the organization. The site must provide additional oversight and accountability in order for the Work Management process to meet goals.

1.4 Plant Benchmarking Report

Unit Performance Summary: Generally, the benchmark data suggests that Vermont Yankee has been performing at a "median" level. Although, there is concern that some recent ENVY indicators (Industry Performance Index and Forced Loss Rate) suggest unit performance might be declining.

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The data for ENVY, compared to the rest of the industry and the sister plant benchmarks, showed:

1. Considering the contractors that were not reported in the ENVY data, ENVY meets industry staffing standards.
2. ENVY overall performance, as measured by the 2-year rolling average indicator has declined from 2006 to 2008. This decline is influenced by higher than industry forced loss rates, lower capacity factors and a greater number of recordable injuries.
3. The ENVY System Engineers have more systems per person than engineers at the benchmarked sister plants.

2.0 System Analyses and Technical Focus Areas

Introduction

The System and Technical Focus Area section of this report includes the observations, results and conclusions of 6 'Vertical Slice' evaluations of selected systems. These evaluations were conducted to address the 13 specific 'Vertical Slice' criteria described in Act 189. hese criteria are as follows:

1. Initial Conditions - What were the codes and standards with which the system was designed to comply and what was the design basis? Is the design of the system in keeping with the expected initial conditions and its design basis?
2. Procurement - If there were procurement changes, was a new set of review calculations completed for those procurement changes and were those procurement changes compared against the original design and all of its calculations?
3. Installation - "as-built." Do plant records adequately represent the as-built condition of the plant? Are all changes reflected in all documents from the design basis through as-built and through current operations?
4. Operation - What changes or compensations have been made to accommodate unanticipated operations outcomes? Have those changes, compensations, and accommodations been duly noted in procedural manuals and logs? Have root cause analyses been conducted to reflect unanticipated outcomes? If root cause analyses were not conducted in any particular instance, why not? If root cause analyses were not conducted in any particular instance, have any unanticipated system operations outcomes been duly corrected or compensated in all safety and reliability operations and procedures?
5. Testing - When systems have undergone periodic tests, what have been the results? Are resulting corrective actions reflected in all documents from design through as-built through current operations?
6. Inspection - When systems have undergone periodic inspections, have those inspections been successful? Are the resulting changes reflected in all documents from design through as-built through current operations?
7. Maintenance - Has the management system for aging components been adequately maintained to assure the components meet the design basis? Is there a track-change system in place to determine what components have been reviewed, repaired, or replaced? Is there an accurate system in place to record when those reviews and repairs were completed? Is there a program of operations or a schedule of operations, that specifically delineates what aging management systems, as identified in the industry-wide database, are being reviewed and when? Is adequate time allowed in each outage for aging management review and adequate maintenance? Are the aging factors discovered actually being repaired in a timely manner?
8. Repairs - Have repairs been performed which assure the system will operate as expected? Are all repairs completed as soon as possible? Are repairs sufficiently in-depth to effectively invest in the plant and its operational systems?

9. Modifications - Do all modifications to the system also comply with the system's original design basis? Have all procedure manuals and operations manuals been updated to reflect the impact of any modifications made to any system?
10. Redesign - Have changes made to the plant since its original construction been reviewed to ensure that safety margins have not been reduced? Has each component modified for uprate been reviewed to assure that operational margins have not been reduced and to assure that design basis redundancy has not been compromised? Have any repairs, maintenance, or modifications impacted the original design of the redundant safety systems? Are all systems still "single failure proof"?
11. Seismic analysis - When was the most recent modern, computer generated, finite element seismic analysis performed on each of the seven vertical slice systems examined in the audit? Does ENVY remain capable of withstanding design basis events beyond the original 40-year design life of the plant to reflect the age-related changes in the plant and weight changes from all modifications during the first 35 years of operation?
12. Training - Has an adequate review and evaluation of operator training and operating procedures been conducted? Has each change been adequately reflected in the operations procedures? Have operations personnel been adequately trained in all modifications to all systems? Are operations personnel frequently updated and trained regarding any troublesome issues other plants have uncovered which may compromise operations and safe shutdown?
13. Corrective action programs. What corrective action programs have been established for each of the systems audited? Have the corrective actions taken been properly integrated in the corrective action program? Have corrective actions been taken in a timely manner? Where recorded items have been deferred, have they been appropriately evaluated for risks and potential consequences of deferral and appropriately tracked while awaiting resolution?

These criteria were used to guide the evaluation process for the following six systems

- Transformer and Switchyard System
- High Pressure Coolant Injection (HPCI) System
- Residual Heat Removal (RHR) System
- Condensate/Feedwater System (Including Condenser)
- Cooling Tower (CT) Structure
- Service Water (SW) System

For 2 specific systems (HPCI and SW) an additional criterion was evaluated. This was to address ACT 189 Section 2 (2) which requires the assessment to 'Identify all relevant deviations, exemptions, or wavers, or any combination of these from any regulatory requirements to ENVY and from any regulatory requirements applicable to new nuclear reactors, and verify whether adequate operating margins are retained dispute the accumulative effect of any deviations, exemptions, or wavers for the present licensed power level for the proposed period of license extension.' The NSA team evaluated these 2 systems against applicable sections of the current Standard Review Plan (SRP). The differences between the ENVY Design Basis Document and the SRP were evaluated and the difference(s) were assessed for their potential effect on plant reliability.

The methodology applied to evaluate these systems included:

- Interviews with design, system and component engineering personne
- Observation of field activities such as walk downs, testing, and inspection
- Review of numerous procedures and programs and drawings
- Review of NRC Inspection Reports
- Review of Design Basis Documents and the UFSAR
- Review of NRC Component Design Basis Inspection Reports
- Review of Condition Reports (CRs) and associated corrective actions
- Review of the industry standard Equipment Reliability (ER) sub-process application
- Review of Extended Power Uprate and License Renewal Applications and Safety Evaluation Reports
- Review of selected Modification Packages and calculations

Also various specific technical focus areas were evaluated for their potential impact on plant and equipment reliability. The technical areas reviewed included:

- Underground Piping and Tanks Inspection Program
- Cable Separation Practices
- Large Electric Motor Program Horizontal Review
- Primary Containment System (Drywell Shell, Torus Supports, Isolation Valves)
- Electrical System: Back up and Stand-by
- Flow Accelerated Corrosion Program
- Crane and Hoists Maintenance and Testing
- Seismic Analysis Program

Each of these technical focus areas was evaluated based on the specific methodology and objectives described in the respective sub-sections of this report. The seismic analysis review included a review of ENVY's seismic analysis program to determine the company's approach to seismic design requirements, a review of the modification process to determine how seismic design considerations were addressed, and a review of selected modification packages to determine if appropriate consideration was given to seismic analysis requirements.

Conclusions are included at the end of each sub-section and then summary conclusions are included at the end of the System and Technical Focus area (Section 2.14).

2.1 Transformer and Switchyard System

System Description

For the purposes of this assessment, the Transformer and Switchyard System consists of all connected equipment located between the generator terminals and the 345KV switchyard located roughly 200 yards northwest of the turbine building. This includes the Isolated Phase Bus Duct that runs from the generator terminals to the Low Voltage (LV) side of the Generator Step-Up Transformer (GSU), the GSU itself, the strain bus sections that run over to the switchyard and the equipment in the switchyard. This system only includes equipment within the 345KV switchyard out to the disconnect switches for the three 345KV lines and the disconnect between the 345KV North Bus and the 345/115KV autotransformer (T-4-1A), excluding the autotransformer. The 115KV switchyard is excluded from this system. Also excluded is the Unit Auxiliary Transformer located on the 22KV bus between the generator and the GSU. The diagram below depicts the major pieces of equipment (minus metering and protective relaying equipment), with the included components outlined. START CONFIDENTIAL INFORMATION

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2.1.1 Criterion 1 - Initial Conditions

Assessment Response

The function of the Transformer and Switchyard system is to conduct the generated electricity from the generator terminals, transform the power from the generator voltage to the required transmission voltage and distribute the power among the three 345KV transmission lines and the 345KV/115KV autotransformer that connects the 345KV grid to the local 115KV sub-transmission system.

The 22KV and 345KV electrical systems are classified as "non-safety related" systems. The design bases of these systems are not specifically established 'in the Design Basis Documents (DBD) maintained by the plant. The DBD for the 'Safety Related 4.16KV/480V System' describes portions of this system in relation to off-site power supply sources, but does not enumerate design bases for these components. The design basis in the UFSAR states that this system 'shall provide [the generated electrical energy] reliably for use on the transmission network and for use on the station auxiliary buses.'

The original designs and component ratings, as well as the uprated design, were compared with best industry practices and published standards to assess the adequacy of the system components to provide reliable electricity supply to the bulk transmission system. The Transformer and Switchyard system and components are generally governed by IEEE Standard Series C57 andC37 and the National Electric Safety Code. In addition, certain electrical aspects of the system that may affect overall reliability and operation of the bulk electric system are governed by ISO-NE (formerly NEPOOL) system in accordance with the 'ISO-NE Reliability Standards' and the ISO-NE 'Minimum Interconnection Standard'. The regulatory requirements that pertain to bulk electric system reliability are still evolving as a result of the Energy Policy Act of 2005.

The evaluation of the design changes with respect to the original design bases is included in the design change process outlined in EN-DC-115, 'Engineering Change Development'. Review of the GSU design change (VYDC-2002-002) and the isophase bus uprate modification (VYDC-2003-018) demonstrate sufficient attention to changes in design basis and comparisons of new designs to the existing design basis to ensure compliance. The design basis is documented in the various drawings and specifications as appropriate.

In addition to the local plant design basis, ISO-NE may impose requirements to ensure system reliability. A complete evaluation of the system impacts of the EPU were performed by GE Power Systems and documented in a report titled 'Final Report to ISO-New England for Vermont Yankee Uprate System Impact Study' dated October 1, 2003. This report documented additional design basis and initial conditions, in terms of area bus voltages, power flows and system stability, both for the original design conditions and the EPU design conditions.

Isolated Phase Bus and Bus Duct: In general, the isolated phase bus was originally designed, manufactured and installed in accordance with IEEE C37.20-1969 and C37.23-1969, as well as the original EBASCO Specification No. VYNP – IV-B-4. The uprated design was performed in accordance with IEEE C37.20-1987 and VYS 2003-010.

GSU Transformer: The GSU was specified and manufactured in accordance with IEEE C57.12.00-2000. This standard specifies general performance requirements and design criteria. Specific requirements and additional criteria are specified in VYSP-EE-062 'Specification for Generator Step-Up Transformer'. The requirements specified in this specification are consistent with general industry practices. The GSU underwent factory acceptance testing in accordance with IEEE C57.12.90. This testing is intended to ensure that basic performance parameters meet the guarantees and that there are no gross defects or design flaws.

Based on the document reviews and interviews noted above, the design of the Transformer and Switchyard System is in keeping with its expected initial conditions and its current design basis. There should be no negative effects on the future reliability of the Transformer and Switchyard system based on design modifications.

2.1.2 Criterion 2 - Procurement

Assessment Response

There have been several significant modifications to components of the 22KV and 345KV electrical systems since original plant construction. Some of the more recent major and minor modifications were reviewed. This included the replacement of the GSU in 2002 (VYDC-2002-002), modifications to increase the isophase bus ampacity in 2004 (VYDC-2003-018), repairs to the isophase bus following the electrical fault and fire in June 2004 (VYDC-2003-018 CR15) and the modification to install secondary protection for the GSU transformer as required by ISO-NE reliability standards (MM-2003-049). For each modification, appropriate comparisons and calculations were performed to ensure that replacement equipment would perform in a manner compatible with the original equipment and did not impact safety-related functions. (This report assesses reliability at ENVY, not safety. However, when a safety-related function does not perform properly, it can impact reliability by causing forced outages or power derates.)

With most of the equipment in the Transformer and Switchyard system, only the design inputs are specified as necessary. The detailed design is the responsibility of the vendor and is often viewed as proprietary to the vendor. Detailed calculations are generally not shared with the purchaser. The purchaser may perform design reviews, but these reviews are limited in scope. Factory testing is often done to ensure compliance with the design inputs and to provide some assurance that the equipment is free from defect.

When the new ABB GSU transformer was specified in 2002 (as part of VYDC 2002-002), the transformer was specified to be a direct physical and electrical replacement of the previous transformer, with a few notable exceptions. The electrical impedance of the transformer was specified to match the previous GSU. This was done to minimize the effects of a change in impedance on fault current levels, relay settings and voltage regulation through the transformer. Deviations from the original equipment were limited to areas where prior reliability concerns have been raised (e.g. availability of transformer cooling equipment with loss of DC power) or to anticipate the needs of the then anticipated EPU (e.g. increase of nameplate rating from 650MVA to 675MVA and LV nominal voltage from 20.9KV to 21.9KV). In each case, these deviations were evaluated to assess potential impacts of the change.

The isophase bus and bus duct was subject to a significant modification in 2004 to support the EPU, documented as VYDC 2003-018. In this modification, the bus cooling equipment was replaced with equipment with an increased capacity to support an increase in bus rating from 17.3kA to 19.0kA. Design calculations were performed by the bus manufacturer, Delta-Unibus, and documented in 'Report Engineering Study for Isolated Phase Bus Duct at Vermont Yankee Nuclear Unit-1' dated May 18, 2003. This report is referenced in VYDC 2003-018. This report appropriately reproduced design calculations for the original bus duct 'as built' and for the proposed uprated design. Comparisons were made to assess impact on associated equipment and systems and determine any additional modifications needed to support the uprate.

One specific Procurement Engineering Technical Evaluation was reviewed (ENN-05-0290). A replacement 15A fuse for control wiring associated with the GSU transformer had no direct replacement available. For this component, a thorough engineering comparison with a replacement 15A fuse from a different manufacturer than the original was performed. All dimensions and ratings were verified the same, as well as the time-current characteristics to ensure compatibility. In this instance, it appears that all procedures for procurement changes were appropriately followed.

In summary, for the design changes and modifications examined, review calculations were performed as appropriate for new equipment and compared to the original design and its associated calculations.

Based on the document reviews and interviews noted above, for the Transformer & Switchyard System, new sets of review calculations were completed for the respective procurement changes and the procurement changes were compared against the original design and all of its calculations. There should be no negative effects on the future reliability of the Transformer & Switchyard system based on the process for maintaining procurement records.

2.1.3 Criterion 3 – Installation

Assessment Response

Reviews of Design Change documentation and minor modification packages indicate that due consideration is given to the update of all applicable documentation, both in the system for which the design change directly applies as well as other affected systems. Detailed Installation and Test (I&T)

Procedures are developed for each major modification. These procedures provide step-by-step instructions for completing the installation work and subsequent acceptance testing, in addition to the acceptance criteria. As part of the closeout procedure associated with the design change completion, the Implementing Cognizant Engineer (ICE) is specifically required to verify that all design documents and drawings reflect the as-built conditions.

Two major modifications were reviewed with specific attention to adherence to Installation and Test Procedures and update of documents to reflect as-built conditions. This included the GSU replacement in 2002 and the bus duct modifications and cooler replacement in support of EPU in 2004. In each case, the ICE signed off on the verification of as-built conditions on the drawings.

For further verification of adherence to procedures, drawings were reviewed for inconsistencies. Specifically, drawing G-191298-SH-01 rev. 44, "345KV & 22KV one-line drawing" was reviewed to ensure that all notations and configurations that may have changed as a result of the above modifications were reflected in the drawing. All modifications resulting from VYDC-2002-002 (GSU Transformer Replacement), VYDC-2003-018 (EPU Isolated Phase Bus Cooling Modification), VYDC-2003-018 CR15 (Isophase Bus Fault Repair) and MM-2003-049 (Install Additional Main Transformer Protection to Support EPU) were appropriately reflected in drawing G-191298-SH-01 rev. 44.

Safety Classification Worksheets are required to be completed as part of the procurement process per EN-DC-313, "Procurement Engineering Process". One such evaluation was reviewed, as required for implementation of VYDC 2002-002, "GSU Transformer Replacement". The Safety Classification Worksheet SCW-2002-033 was properly completed, classifying the new ABB transformer as "NNS".

Based on the document reviews noted above, the Transformer & Switchyard plant records adequately represent the as-built condition of the plant. All reviewed Transformer & Switchyard system changes have been adequately reflected in all documents from the design basis through as built and through current operation. ENVY meets industry expectations as it pertains to the installation criterion for the Transformer & Switchyard System.

2.1.4 Criterion 4 - Operation

Assessment Response

Unanticipated operations outcomes for this assessment considered plant operating or equipment conditions that did not result in the desired or expected outcome. Interviews with operations management were conducted in order to review the processes used to identify and correct each unexpected operational issue. How Operations implements the governing procedures and processes was discussed with operations management. An example was selected of an event within the transformer/switch yard and reviewed to ensure appropriate compensatory actions were taken, and that plant procedure deficiencies were addressed as appropriate. Additionally it was evaluated to determine if the appropriate level of analysis was performed in accordance with the corrective action system.

As part of the CR evaluation it is determined whether an Operability Evaluation (in accordance with EN-OP-104, Operability Determination procedure) is required for degraded safety-related equipment. The CR evaluation also determines whether an Operational Decision Making Instruction is required for degraded equipment.

An Operational Decision Making Instruction is a formal process that justifies continued operation of a degraded system or component. It is developed by operations and/or engineering and reviewed by Condition Report Review Group (CRG). In some cases the Operational Decision Making Instruction specifies compensatory actions that must be completed to continue to operate the system.

The compensatory actions that must be completed by operations on an interim basis can be included in operator equipment inspection instructions, operator logs, standing orders, turnover sheets and operating procedures. The operating crew at the beginning of each shift discusses these compensatory actions.

Condition Report, CR-VTY-2005-01043 Switchyard Breaker 79-40 SF6 gas pressure low, was selected as an event driven action taken in response to an unanticipated plant event. On April 5, 2005 an Operator found 79-40 Breaker differential pressure greater than 5 psig and below 80 psig during performance of his rounds in the switchyard. CR was initiated and classified as Significance Level 'C' (condition adverse to quality requiring correction). Based on this classification 'C' no root cause was required or requested. This is in accordance with procedure EN-LI-102 Corrective Action Process. In response to corrective action 1, assigned to investigate and correct, an Operational Decision Making Instruction (ODMI) was initiated. ODMI is a plant process, which involves formulation of a peer group or individual in order to evaluate the condition and determine if interim actions would be required pending repairs. The ODMI was initiated to provide Operations with monitoring and trigger (breaker 79-40 SF6 gas pressure) guidance for the 79-40 Breaker. Actions were delineated based on breaker SF6 gas pressures. Review of the ODMI found relevant information on actions to be taken at defined pressures. These actions included opening of the breaker if gas pressure reaches 74 psig (including the possible need to decrease power) and a contingency should the breaker reach 72 psig with the breaker still open. Action assigned to Operations involved increased readings to be taken on breaker 79-40 SF6 gas pressures. These readings were verified logged by review of the Heightened Awareness section of the Operator Rounds taken on May 6, 2005.

Based on interviews, review of procedures, and review of the response to unanticipated plant conditions and events for this system it was evaluated that the processes, procedures, and the actions taken meet industry standards.

2.1.5 Criterion 5 - Testing

Assessment Response

Multiple interviews with the System Engineer and supervisor were conducted to determine the operating performance of the Transformer and Switchyard System over the past 3 years, with a focus on the past twelve months. As stated in the system description section, the Transformer and

Switchyard System is designed to conduct the generated electricity from the generator terminals, transform the power from the generator voltage to the required transmission voltage and distribute the power among the three 345KV transmission lines and the 345KV/115KV autotransformer that connects the 345KV grid to the local 115KV sub-transmission system. Additionally the system supplies power to the station service loads and cooling towers. Based on this system being categorized as non-safety related, the assessment focused on preventive maintenance and predictive maintenance activities over the past three years.

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ENVY has formal processes for identifying testing requirements and methods as well as performing the actual testing of the Transformer and Switchyard system. The corrective action system tracks the implementation of corrective and compensatory measures for the Transformer and Switchyard System when issues arise. These processes are consistent with industry practice.

2.1.6 Criterion 6 – Inspection

Assessment Response

Transformer and Switchyard System Inspections consisted of a wide range of activities including the System Health/Component Health Reporting process (EN-DC-143), System Engineering Walk-downs (EN-DC-178) and the System Monitoring Program (EN-DC-159). The output of these core system-engineering processes, provide the basis for this assessment.

A review of the past six months of system walk-down reports as well as participating and observing System Engineer during October of 2008 walk-down provided an opportunity to see the actual implementation process. In general the walk-down was consistent with industry practices based on reviewing the implementation procedure and past reports. Condition Reports were initiated for appropriate material condition issues.

A review of the Transformer and Switchyard Component Monitoring Plans (no issue date on report), included the following:

- Predictive Maintenance activities, such as
  - ➢ Semi-Annual Corona testing
  - ➢ Monthly DGA (oil analysis)
  - ➢ Quarterly Infrared Thermography Surveys
- Diagnostic testing including
  - ➢ Power factor
  - ➢ Tinning tests
- Weekly operator walk-downs

These tasks are consistent with industry standards.

During RFO 27 scheduled inspections and tests of the Transformer and Switchyard system were performed and identified the following major issues in support of Transformer and Switchyard system:

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ENVY has formal processes for identifying inspection requirements and methods as well as performing the actual inspection of the Transformer and Switchyard system. The corrective action system tracks the implementation of corrective and compensatory measures for the Transformer and Switchyard System when issues arise. These processes are consistent with industry practice.

2.1.7 Criterion 7 – Maintenance

Assessment Response

The maintenance philosophy for the Transformer and Switchyard system is based on the utilization of condition based-maintenance. This type of maintenance is consistent with industry practices and includes on-line diagnostic and monitoring technologies to ensure reliability of the equipment. Even though the Transformer and Switchyard system is non-safety related, portions of the system are critical to delivering power from ENVY to the power grid. Therefore, online diagnostic and monitoring technologies are necessary to allow ENVY to test and maintain these components while they are in-service. Several tests and inspections are performed during Refuel Outages when the equipment is not in operation due to the nature of the testing and thus ensures thorough assessment of each of the components within the transformer and switchyard system. Additionally, operational requirements make maintenance of components within the transformer and switchyard system difficult. Most maintenance is completed during refuel or other scheduled plant outages.

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It should be noted that VELCO operates the electrical grid in the State of Vermont and is contracted by ENVY to conduct the majority of work in the switchyard. VELCO has a highly trained and experienced internal maintenance group for high voltage distribution and switchyards. Currently, there are ongoing discussions between ENVY and VELCO concerning a change in ownership of the switchyard from ENVY to VELCO. This assessment reviewed the reliability of the switchyard as maintained by the current owners (ENVY and Entergy), but the proposed change in ownership has the potential to impact the reliability of the switchyard. This should be addressed in the change management plan for the transfer of responsibilities form ENVY to VELCO.

Based on discussions with the System Engineer, a review of EN-DC-153 (Preventive Maintenance Component Classification) was conducted to determine the basis for how the Transformer and Switchyard system is maintained. It was concluded that the procedural processes utilized by ENVY are consistent with current industry standards and practices**.**

A review of the specific PM basis documents called Templates (EN – Transformer – Oil Immersed Rev.0 – January 2, 2007 and EN – Switchyard – Rev. 0 – December 31, 2006), was conducted for Transformer and Switchyard and it was concluded that the overall approach and requirements were consistent with industry standards. This document establishes component boundaries, Common Failure Causes, and Maintenance Risk. Additionally the PM basis database item (EO52A), was identified to its' link with Switchyard and Large Power Transformer Preventive Maintenance Guideline ENN-EP-G-004 Att. 7.6 (Rev 0 1/15/07) and a specific maintenance task selected for review. In this case an air switch PM at a 6-year frequency was selected and shown to be completed and data and test results were documented with the completed work package scanned into ENVY's INDUS which is a computerized work management system.

It should be noted that the Main Transformer is required to move power from ENVY to the power grid and a reliability issue exists should the transformer fail and replacement is required. Large power transformers such as ENVY's Main Transformer manufactured by ABB are not off-the-shelf or in stock items from a manufacturer. If a utility were to place an order for a transformer similar to the ENVY Main Transformer it could potentially take several years to take delivery of that order. Because this is a known issue within the power industry the availability of spares is critical.

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ENVY has formal processes for identifying maintenance of the Transformer and Switchyard system. These processes are consistent with industry practice.

2.1.8 Criterion 8 – Repairs

Assessment Response

The assessment of repairs to the Transformer and Switchyard System was based on a review of maintenance activities conducted during RFO-27 (refer to Section 2.1.7) and selected CRs, which documented specific repair activities that could affect system reliability:

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END CONFIDENTIAL INFORMATION The NSA team noted, that this was a quick turnaround from the morning of October 20, 2008 when the System Engineer started the investigation to repairs completed on October 23, 2008 – four days. Also noted is that the breaker was out of service at the time the problem was discovered. This event was entered into ENVY's Corrective Action Process, but had not been completed at the time this report was issued.

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ENVY has formal processes for identifying repair requirements and methods as well as performing the actual repairs on the Transformer and Switchyard system. The corrective action system tracks the implementation of corrective and compensatory measures for the Transformer and Switchyard System when issues arise. These processes are consistent with industry practice.

2.1.9 Criteria 9 – Modifications

Assessment Response

Modifications are documented as part of the Engineering Change Process described in procedure EN-DC-115. This procedure requires discussion of design basis impacts, design input considerations and discussion of impact screening results. In addition, impacts to procedures are assessed and any required changes are tracked as open items. Closure of the open items is required as part of the change closeout process, ensuring that procedures are updated as necessary.

There have been several significant modifications to the Transformer and Switchyard system in the past 6 years, resulting in the replacement of the GSU transformer and a significant portion of the isophase bus, including complete replacement of the bus forced-air cooling equipment.

VYDC-2002-002 'GSU Transformer Replacement'

The replacement of the GSU is documented in VYDC-2002-002. A design flaw in the previous GSU that limited the previous transformer's reliability at full load prompted the replacement of the GSU. The new transformer was specified to support the anticipated EPU, with an increased rating from 650MVA to 675MVA and a change in rated low-side voltage from 20.9KV to 21.9KV. The transformer impedance was specified to match the previous transformer, in order to limit the impact of the change on fault current levels, protection system and voltage regulation. As much as was practical, the new transformer was specified to be physically and electrically interchangeable with the previous transformer.

The new GSU transformer is cooled by 6 heat exchangers with 1 pump and 2 fans each. It is important to note that this transformer is not designed for self-cooled operation (without pumps and fans). Loss of pumps or fans will severely limit the load carrying capability of this transformer. The manufacturer, ABB, has provided a table of load capability for partial loss of cooling. This table is given in VY OP-2140. Total loss of cooling would require the unit to be taken off line immediately. OP-2140 properly reflects this.

The new GSU transformer was equipped with an automatic transfer switch for the power supply to the cooling equipment. Loss of transformer cooling would require a manual trip of the unit. To avoid this situation, a 480V automatic transformer switch was included to transfer from the "normal source" to the 'emergency source' if the normal feed was unavailable. This was done in response to SMRC 2000-045, 'Undesirable Conditions on Loss of DC'. With the previous configuration, loss of DC power to bus DC-3A would result in loss of power to the transformer cooling equipment, ultimately resulting in a unit trip. The new configuration provides a source of backup power. The load on the 480V bus due to the cooling equipment of the new transformer (135.6A) was compared to the load from the original transformer (180A). Since the new cooling equipment load is less than the original design load, no further action was deemed necessary.

VYDC-2003-018 'EPU Isolated Phase Bus Cooling Modification'

In order to support the increased bus current to 19,000A, increased cooling capacity needed to be added to the bus forced air-cooling equipment. To do this, the coolers were replaced with higher capacity coolers, larger fans were added to increase air flow and deflectors were added to the generator disconnect GD-1 to increase air flow around the current carrying parts of the switch. These modifications were documented in VYDC-2003-018, 'EPU Isolated Phase Bus Cooling Modification'. As part of the design change procedure, Design Basis Documents were reviewed to ensure there was no impact. Specifically, TBCCW DBD, Section 4.1 was determined to be unaffected. In addition, ENVY technical specifications were determined to be unaffected. A change was required and completed in the UFSAR Section 8.2.3 to reflect the increase in bus rating from 17.3kA to 19.0kA.

VYDC-2003-018 CR15

Following the electrical failure and fire on June 18, 2004 (CR-VTY-2004-2019), a large section of the isophase bus and bus duct was damaged. A majority of the LV bushing box at the GSU was destroyed. All insulators were cracked or broken. There were arc marks on the bus conductor and bus duct from the turbine-building wall out to the GSU. To repair this, the entire bus and bus duct from the turbine building wall to the GSU was removed and replaced, all insulators were replaced, and the remaining bus duct thoroughly cleaned. In addition, new lightning arrestors were added to the bus duct at the turbine-building wall and the generator potential transformers and potential transformer cabinets were replaced with new equipment.

The new bus duct sections utilized a conductor with a round cross-section versus the square cross-section of the original bus. The isophase bus vendor, Delta-Unibus, performed detailed design calculations and determined that the design change had no adverse impact on the original design. The current rating of the bus with the new conductor matched the current rating of the bus with the old conductor, as did the bus through fault capability.

In addition to the replacement conductor, all flexible connections were replaced with new flexible connections that utilized a thicker lamination than the original design (0.032 inch vs. 0.020 inch). This was done to prevent loose laminations in the future that could potentially result in another failure of the bus or of the GSU. The new flexible connections were also fabricated with more weld to minimize the length of "free sheets". An increased inspection frequency of the flexible connections has been implemented to ensure that the updated design performs as intended. Inspection ports were added to provide easy access to the flexible connections.

The replacement of the Generator PTs also entailed the removal of a neutral connection to the generator ground fault neutralized. The cable connecting the PT neutrals to the ground fault neutralizer was damaged in the fault. In considering the replacement, it was determined that standard practice is to connect generator PTs directly to ground. In doing so, the UFSAR required revision to remove the reference of a connection from the generator PT neutrals to the ground fault neutralizer. This change is reflected in the UFSAR.

Operating procedures were reviewed to determine if impacts from the modifications were considered. VY OP-2140 Rev. 50 'Operating Procedure for 345KV Electrical System' was consistent with the 'as built' system. Specifically, a section discussing operation with loss of cooling equipment on the GSU properly reflected the new GSU installed in 2002. In addition, though not part of this system, it was also noted that the generator reactive capability curve was updated to reflect the post-EPU generator ratings.

Based on the document reviews and interviews noted above, all modifications to the Transformer & Switchyard System comply with the system's original design basis. There should be no negative effects on the future reliability of the Transformer & Switchyard System based on design modifications.

2.1.10 Criteria 10 – Redesign

Assessment Response

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Table 9: Transformer Output

Main Generator Output				Station Service Load			Main Power Transformer		
							Losses		Output
x	x	x	x	x	x	x	x	x	x
x	x	x	x	x	x	x	x	x	x
x	x	x	x	x	x	x	x	x	x
x	x	x	x	x	x	x	x	x	x
x	x	x	x	x	x	x	x	x	x

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In conclusion, changes made to the Transformer and Switchyard System since its original construction have been adequately reviewed to ensure that margins have not been reduced. Each component modified for uprate has been adequately reviewed to assure that operational margins have not been reduced and to assure that design basis redundancy has not been compromised. This system does not directly involve a redundant safety system, so no repairs, maintenance, or modifications have impacted the original design of the redundant safety system. Appropriate reviews have been completed to ensure that modifications to this system have not negatively impacted other redundant safety systems. The practices at ENVY pertaining to redesign applied to the Transformer and Switchyard System meet industry standard expectations.

2.1.11 Criteria 11 - Seismic Analysis:

Assessment Response

The Transformer and Switchyard System does not contain any nuclear safety-related systems and is therefore not subject to the seismic design bases described in 10 CFR100, Appendix A.

2.1.12 Criteria 12 – Training

Assessment Response

The training organization has a process for evaluating all engineering changes to determine whether training is required. This evaluation was performed by conducting interviews with training personnel and by reviewing training processes, governing procedures and training materials. Training is notified of all engineering changes, which are entered on the Modification Training Matrix. The matrix is routinely evaluated to determine if the change requires the training materials for any training program to be modified. If the evaluation determines training may be required a Training Evaluation/Action Request (TEAR) is initiated. The TEAR assigns actions to training personnel to evaluate specific training materials to determine whether the materials require revision. Actions are also assigned to develop training materials and to conduct training as required. The engineering change is also reviewed to determine if any changes to the simulator are required. If a simulator change is required a Discrepancy Report (DR) is generated to implement required changes. All actions are tracked to completion. Training also has a process to evaluate whether site and industry operating experience (OE) is to be incorporated into related instructor guides (IG) for classroom and simulator training. An interview with the Operations Training Superintendent revealed that operating experience is to be included in the development and revision of training material. There is a general understanding of the Entergy Operating Experience procedure (EN-OE-100) including interface with the Site OE Coordinator. There is also a proceduralized requirement for use in the development of training material (EN-TQ-201), Systematic Approach to Training Process) that reinforces the need to consider using Operating Experience to re-enforce learning objectives.

Modification ER 04-526 (115 KV Switchyard Capacitor Bank Project) was selected based on review of the modification. Review of the modification found that training would be required. This modification consists of the addition of 115 KV capacitor banks to ensure that bus voltage is not degraded as a result of the uprate project. The Operations Training Department reviewed this modification and initiated Training Evaluation Action Request (TEAR) VTY 2005-206. This initiated a NEEDs analysis, which identifies if training is required and if so, what Training Department Information Guides (IG) and/or changes to the Simulator would be required. There are then actions assigned to address the changes. The following Instructor Guides were verified complete and reflect the changes made under modification ER 04-526: Licensed Operator Re-qualification (LOR) LOR-24-

905-2 and Non-licensed Operator Training (NLOR) NLOR-24-905-2 for modification training. There were no changes required for the simulator. Modification training was verified through Training department tracking documentation to have been completed during the August, October, and February of 2005 time frame. Operating Experience references were validated to have been referenced in Instructor Guide LOR-24-905-2 and included Significant Operations Event Records (SOER) 99-01, 02-03, and Significant Event Notifications (SEN) 234 and SEN 242. Procedure OP 2141 115 KV Switchyard was reviewed and the appropriate modification changes were verified to have been incorporated. Operations Training performance in the review and implementation of modification ER 04-526 (115 KV Switchyard Capacitor Bank Project) meets expectations. As a result of review of the process and interviews with Training department management, along with review of the actions taken in response to this particular modification, it is concluded the Training department meets industry standards with respect to evaluating modifications and taking the appropriate actions to incorporate the required changes into training material.

ENVY has formal processes to ensure that when engineering changes are implemented for the Main Transformer and Switchyard System, that reviews are conducted and changes made as required to operating procedures, training materials and the simulator to accurately reflect plant conditions. These processes are consistent with industry practices.

2.1.13 Criteria 13 - Corrective Action Program

Assessment Response

An overall assessment of the corrective action program and its effectiveness was completed and is documented in Section 1.2.5 that addresses the Criterion 13 (Corrective Action Program) questions for the overall Corrective Action Program.

A list of Condition Reports (CRs) back to the year 2000 was reviewed. Certain CRs from this list are discussed in detail in the sections above.

Based on these reviews it was determined that issues related to the transformer and switchyard systems are being identified and entered into the Corrective Action Process. Corrective actions assigned were appropriate to address these issues.

The transformer and switchyard systems CR review supports the conclusion of Section 1.2.

Transformer and Switchyard System Conclusions

Based on the document reviews and interviews noted above, the following is a summary of Transformer and Switchyard System assessment conclusions. More detail is provided at the end of each individual section above.

The ENVY engineering design process is well documented, controlled and consistent with industry practice. The current design of the Transformer and Switchyard System is consistent with its original design basis and is adequately reflected in plant records and procedures.

ENVY has formal processes for identifying and correcting unanticipated operations outcomes. The corrective action system tracks the implementation of corrective and compensatory measures for the Transformer and Switchyard System when issues arise, including the revision of appropriate documents. These processes are consistent with industry practice.

The data reviewed and interviews indicate that the Transformer and Switchyard system is tested, inspected, maintained and repaired to industry standards such as AP-913 Equipment Reliability Process. Component classification was performed; PM Basis documents created, and testing and inspections were scheduled and carried out.

Transformer and Switchyard system issues, as described in individual sections above, are being entered into the Corrective Action Process. Corrective actions were appropriately assigned to address issues.

There are 2 issues that could potentially affect the reliability of the Transformer and Switchyard system. The first issue is the request to the Public Service Board by ENVY and VELCO concerning a change in ownership of the switchyard from ENVY to VELCO. A final decision has not been issued by the Public Service Board as of this writing. This assessment reviewed the reliability of the switchyard as maintained by the current owners (ENVY and Entergy). The proposed change in ownership needs to be closely monitored to ensure an effective transfer. This should be addressed in the change management plan for the transfer of responsibilities from ENVY to VELCO.

The second issue is the current main transformer spare. Large Power transformers such as ENVY's Main Transformer manufactured by ABB are not off-the-shelf or in stock items from a manufacturer. If a utility were to place an order for a transformer similar to the ENVY Main Transformer it could potentially take several years to take delivery of that order. Because this is a known issue within the power industry the availability of spares is critical. ENVY is in the process of addressing this issue as part of the Transformer and Switchyard system long-term plan. The current designated spare for the Main Transformer at ENVY is the previous Main Transformer manufactured by Peebles, which was removed from service prior to the uprate of the plant. START CONFIDENTIAL INFORMATION

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References

Procedures and Specifications:

1. VYSP-EE-062 Rev. 0, Generator Step-Up Transformer
2. EN-OE-100 Rev.5, Entergy Operating Experience procedure
3. EN-TQ-201 Rev. 7, Systematic Approach to Training Process
4. EN-LI-102 Rev. 12, Corrective Action Process
5. OP 2141 Rev.18, 115 KV Switchyard
6. EN-DC-143, System Health and Component Health Reporting
7. EN-DC-159, System Monitoring Program
8. EN-DC- 178, System Engineering Walk-down Process

Drawings

1. Drawing G-191298 Sh. 1 Rev. 44, 345KV and 22KV One-Line Wiring Diagram

Documents

1. Task Report WBS 1.4.9.5, Powerblock Equipment EPU Task Report for ER-04-1409
2. Final Report to ISO-New England for Vermont Yankee Uprate System Impact Study, GE Power Systems, dated October 1, 2003
3. VYDC-2003-018, EPU Isolated Phase Bus Cooling Modification
4. VYDC-2003-018 CR15, EPU Isolated Phase Bus Cooling Modification (following CR-VTY-2004-2019)
5. VYDC-2002-002, GSU Transformer Replacement
6. Report Engineering Study for Isolated Phase Bus Duct at Vermont Yankee Nuclear Unit-1, Delta-Unibus Corp, dated May 18, 2003
7. I&T 2002-002.01, I&T Procedure for Implementation of VYDC 2002-02 – GSU Transformer Replacement
8. I&T 2003-018.01, EPU Isolated Phase Bus Cooling Modification
9. VY OP-2140 Rev. 50, Operating Procedure for 345KV Electrical System
10. VYC-2333 Rev. 0, Main Transformer Protection Calculation
11. MM-2003-049, Install Additional Main Transformer Protection to Support EPU
12. ENN-05-0290 Rev 00, Item Equivalency Evaluation for 15A Fuse
13. SCW-2002-033, Safety Classification Worksheet for T-1-1A ABB
14. EO52, A PM Basis database item
15. Licensed Operator Re-qualification LOR-24-905-2 rev. 0, modification training
16. Non-licensed Operator Training NLOR-24-905-2 rev.0, modification training

17. ENN-EP-G-004 Att. 7.6 (Rev 0 1/15/07), Large Power Transformer Preventive Maintenance Guideline
18. LO-VTYLO-2007-00136, Preventive Maintenance and monitoring activities for maintaining the Peebles transformer
19. PDM Watchlist Report

Condition Reports

1. CR-VTY-2005-01043

2.2 High Pressure Coolant Injection (HPCI) System

System Description

The following is a summary level description of the High Pressure Coolant Injection (HPCI) System. A more detailed description of the HPCI System is provided in the VYNPS UFSAR and the High Pressure Coolant Injection System DBD.

The High Pressure Coolant Injection (HPCI) System is an Emergency Core Cooling System (ECCS) at the ENVY. It is designed to supply high pressure coolant to the reactor core, to prevent excessive fuel clad temperatures, in the event of a small-break loss of coolant accident (LOCA) which does not result in a rapid depressurization of the reactor vessel, and can supply reactor makeup during periods when the Feedwater System is isolated or otherwise unavailable. For small break LOCA, the HPCI System is designed to maintain sufficient vessel coolant inventory to permit the reactor to be shut down and depressurized to the point at which the Low Pressure Coolant Injection (LPCI) or Core Spray System can provide core cooling. The injection of cold condensate acts to accelerate depressurization of the reactor vessel such that the lower pressure systems can provide core cooling more quickly. The Automatic Depressurization System (ADS), initiated if the HPCI System is unavailable, will automatically depressurize the vessel to allow injection from low pressure ECCS. Moreover, the HPCI System is designed to act as a backup to the Reactor Core Isolation Cooling (RCIC) System should that system become unavailable when needed.

The HPCI System consists primarily of a steam turbine driving a main pump and booster pump assembly which is controlled automatically to provide a constant flow of water from either the Condensate Storage Tank (CST) or the suppression pool to the core, via the feedwater spargers, for reactor cooling following postulated small breaks in the nuclear system process barrier. The turbine is powered by steam from the reactor which is generated by residual heat and decay heat. The turbine exhaust steam is directed to the suppression pool.

The HPCI System remains in a standby condition during normal plant operations with its suction and discharge water lines filled. The Turbine Steam Supply Inboard and Outboard Containment Isolation Valves are open allowing reactor steam to the Turbine Steam Admission Valve to keep the steam line warm. Condensate is drained from the steam line back to the Main Condenser when HPCI is in standby. Upon system initiation the admission valve opens admitting steam to the turbine. The auxiliary oil pump is started providing hydraulic power to the turbine stop valve and the turbine control valve which open to start the turbine and provide pump flow. Turbine exhaust steam is directed through two Exhaust Line Check Valves and a normally-open, locked-open, gate valve to the suppression pool. A line containing two Turbine Exhaust Line Vacuum Breaker Check Valves is provided. This line allows the inflow of Suppression Pool air to prevent drawing suppression pool water up into the turbine exhaust line following system shutdown. Two Turbine Exhaust Rupture Discs are also provided to prevent over pressurization of the exhaust line. Normal flow is from the CST through the CST Pump Suction Isolation Valve to the pump. When the CST level is low, the CST to Pump Suction Isolation Valve shuts automatically and the Suppression Pool to Pump Suction

Isolation Valve opens transferring suction to the suppression pool. From the pump, discharge flow is through the Pump Discharge Valve, Injection Valve and Feedwater Check Valve to the reactor vessel via the feedwater spargers. The min-flow line, which contains the Pump Minimum Flow Bypass to Suppression Pool Check and Isolation Valves, directs flow from the pump discharge line to the suppression pool. Also connected to the HPCI Pump discharge is the full flow test line, containing a check valve and two MOVs, which diverts pump flow back to the CST. It is available in order to allow full flow testing of the HPCI pump and turbine assembly during plant operation. Cooling water is supplied from the booster pump discharge, through the Turbine Cooling Water Shutoff Valve and the Turbine Cooling Water Pressure Control Valve. The flow splits and is directed to the HPCI Oil Cooler and the Gland Seal Condenser in parallel after which it is returned to the HPCI pump suction.

The HPCI System starts automatically upon receipt of a low-low reactor vessel water level or a high drywell pressure signal and provides its design flow rate within a specified initiation time and over a wide range of reactor vessel pressures. It can also be started manually. The HPCI turbine trips automatically when a reactor vessel high water level signal is received. The HPCI System is designed to perform its function following a Design Basis Accident (DBA) without reliance on station auxiliary power other than the safeguard DC power supply.

2.2.1 Criterion 1 - Initial Conditions

Assessment Response

ENVY has verified and documented the original design codes and standards as well as the current design bases in ‘Vermont Yankee Nuclear Power Station Design Basis Document (DBD) For High Pressure Coolant Injection System, Rev. 32,’ dated May 6, 2008. The DBD verification also included an as-built walk down of the HPCI System to confirm compliance with the design basis. Details can be found in ‘Validation Report Design Basis Document HPCI,’ dated November 24, 1998. The design bases for the HPCI System are also documented in the ENVY UFSAR. As part of ENVY’s response to NRC letter 10 CFR 50.54(f), ENVY committed to complete an FSAR verification program to ensure consistency between the FSAR and plant documentation. Those codes and standards are as follows:

- IEEE 279-1968, Proposed IEEE Criteria for Nuclear Power Plant protection systems
- USAS B31.1.0, Power Piping Code 1967
- ASME Boiler and Pressure Vessel Code Section III, Nuclear Power Plant Components
- Standard of the Hydraulic Institute
- ASME Boiler and Pressure Vessel Code section XI, Rules for the In-service Inspection of Nuclear Power Plant Components

The design basis for the HPCI system is also documented in the DBD. The design basis consists of system design bases and major component design bases. The system design bases include the following:

- System functions
- Regulatory requirements
- System design requirements
- Operating requirements
- Electrical and instrument and control requirements

Major component design bases are typically performance requirements such as pump flow rate with developed head. They are too numerous to list here but are captured in the DBD.

As other NRC regulations and regulatory requirements and requests evolved, the ENVY plant evaluated its compliance with these regulations or requirements to assure the NRC of its compliance with their intent. Those that were applicable to the HPCI System generally ranged from NRC regulations in 10 CFR50, Regulatory Guides, to Generic Letters and Inspection & Enforcement (I.E.) Bulletins. All of these are also discussed in the HPCI System Design Basis Document without any exemptions or deviations noted. As part of the DBD process, the HPCI System was walked down to verify it's conformance with the original design basis.

It should be noted that the HPCI System is a safety-related system and as such is subject to strict controls by the NRC. Any deviations or exemptions from the original design basis are not permitted without prior approval by the NRC. Likewise the system is subject to Technical Specification limits and failure to comply with those limits could cause the unit to be shut down. Thus, non-compliance with these requirements could have an impact on unit availability and for this reason is subject to tight controls by the plant.

It should also be noted that the HPCI System is an Emergency Core Cooling System. It does not operate during normal unit operation but rather is in a standby mode in case of an accident. There are therefore few challenges to the operability of the system.

Design changes are being controlled by procedure EN-DC-115, Engineering Change Development. That procedure mandates reviews to ensure compliance with regulatory, code, and industry requirements. It also requires initiation and processing of updates to plant configuration documents and plant operating and design margins. Procedure EN-DC-118 requires that all design basis documents be updated prior to modification closure.

ENVY utilizes a fleet-wide Configuration Management Program, EN-DC-105, Rev. 2. The purpose of the procedure is to ensure consistency between the design requirements, physical configuration, and plant configuration information. This program is also used for primary suppliers of equipment and software for ENVY.

Interviews with plant engineers and managers confirmed their knowledge of these procedures, as well as the system design codes and standards and the design basis for the system. Review of selected design documents, including modification packages, has determined that these procedures are being effectively implemented and the original design requirements and bases are being adequately maintained.

A number of NRC Component Design Basis Inspection (CDBI) Reports, including one this year, support this assessment and concluded that sufficient design controls are in place to assure the components would meet their intended safety functions. The CDBI Report this year inspected numerous plant components and found three findings of very low risk significance. None of the findings were related to maintenance of the original design basis. Details can be found in NRC Component Design Basis Inspection Report 05000271/2008008, dated September 26, 2008. A CDBI Report in 2007 looked at selected components in the HPCI System to verify the HPCI System's ability to meet its design basis. The inspection included HPCI motor operated valves, turbine and pump. There were no findings relative to maintenance of design bases. Details can be found in NRC Component Design Basis Inspection Report 05000271/2006007, dated September 29, 2006. Although not related to EPU, one open issue relative to maintenance of the original design basis is potential overtaxing in the HPCI suction line from the condensate storage tank. If vortexing were to occur under certain conditions, HPCI ability to deliver design flow during an accident event could be compromised. A CR was initiated to address this issue. The resolution requires hydraulic modeling at Alden Research Laboratories and resolution is expected during the 1st quarter of 2009. The HPCI System Health Report is tracking this issue as an open item.

The EPU Program included a comprehensive analysis of the effect of Power Up-rate on the design and operating basis of the plant system and components and evaluated the acceptability of any changes. The review looked at margin changes at both the component and system level. The review analyzed Power Up-rate impact on margins to ensure there were no unacceptable reductions. The analysis was reviewed and approved by every affected organization at the site. The analysis is comprehensive, thorough, and consistent with industry standards. START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION

Based on the document reviews and interviews noted above, the design of the HPCI System is in keeping with its expected initial conditions and its current design basis. There should be no negative effects on the future reliability of the HPCI system based on design.

2.2.2 Criterion 2 – Procurement

Assessment Response

The ENVY processes governing engineering changes, calculations, procurement etc., referenced below, require that applicable calculations be performed to support the engineering changes, that the changes be compared against the original design requirements and design bases, and that they be verified as built prior to return to service of the implemented engineering change. The procedure for performance of calculations, EN-DC-126, prescribes the requirements and the format for the preparation of the calculations. Discussions with plant engineers and managers confirmed their familiarity with these procedure requirements. It was verified that this procedure was used for HPCI System procurement changes. Quality Assurance (QA) audits are also used to verify procedure compliance. There have been 24 modifications to the HPCI System since original plant construction. Some of these modifications were procurement changes. A review of the modification package for Modification 2001-051, HPCI Turbine Exhaust Check Valve Replacement, was reviewed to ensure compliance with the original design bases for procurement changes. Turbine exhaust check valves V23-3 and4 have affected outage durations due to leak test unreliability resulting from sensitivity of the valves to flange loading. The modification replaced the valves with new valves with thicker walls and soft seats. The review determined that appropriate calculations to support the use of the thicker wall valve were performed and the new design was compared against the original valve design.

Based on the document reviews and interviews noted above, for the HPCI System, new sets of review calculations were completed for the respective procurement changes and the procurement changes were compared against the original design and all of its calculations. There should be no negative effects on the future reliability of the HPCI system based on the process for maintaining procurement records.

2.2.3 Criterion 3 – Installation

Assessment Response

The Engineering Change Closure procedure sets forth the requirements of updating pertinent engineering, operations, training, maintenance, program, and licensing documentation. The procedure specifies which updates are required prior to return to service and which are required to be tracked and completed after return to service. The procedure specifically requires that calculation changes be verified as-built prior to returning the system to service. Discussions with plant engineers and managers confirmed their familiarity with these procedure requirements. Quality Assurance audits are also utilized to verify procedure compliance. Reviews of plant records indicate these procedures are being implemented in a timely fashion. Modification package reviews for the HPCI System also verified that system changes are being properly reflected in these design documents. The NRC has also performed an inspection of the EPU Program and concluded that sufficient design controls are in place and have been implemented. There is reasonable evidence and insurance that plant records adequately represent the as-built condition of the plant. Details can be found in NRC Inspection Report 05000271/2004008, dated December 2, 2004.

In February 1997, ENVY responded under oath to the USNRC 10CFR50.54(f) letter to Licensees requesting information regarding adequacy and availability of design bases. In its response, ENVY stated it has reasonable assurance that the ENVY design bases have been adequately translated to the plant design and procedures and that plant configuration is maintained in an appropriate manner. ENVY went on to state that it was committing to provide improved configuration management including completion of a Design Basis Documentation program, improved Technical Specifications program and a FSAR Verification program. The subsequent development of the DBDs, which took place in 1997 and 1998, included a DBD validation process in which validation teams verified both the design basis information and its application to design documents, and its application to operations, maintenance, surveillance, testing functions and physical configuration. The physical configuration validation included as-built walk downs. Walk downs were performed to verify that the system configuration reflects the design basis, is in agreement with plant drawings, and the component labeling is adequate.

Based on the document reviews and interviews noted above, HPCI System plant records do adequately represent the as-built condition of the plant. All HPCI System changes are reflected in all documents from the design basis and through as-built to current operation. ENVY meets industry expectations as it pertains to the installation criteria for the HPCI System.

2.2.4 Criterion 4- Operation

Assessment Response

Unanticipated operations outcomes for this assessment considered plant operating or equipment conditions that did not result in the desired or expected outcome. Interviews with operations management were conducted in order to review the processes used to identify and correct each unexpected operational issue. How Operations implements the governing procedures and processes was discussed with operations management. An example was selected of an event within the High Pressure Cooling Water (HPCI) system and reviewed to ensure appropriate compensatory actions were taken, and that plant procedure deficiencies were addressed as appropriate. Additionally it was evaluated to determine if the appropriate level of analysis was performed in accordance with the corrective action system.

As part of the CR evaluation it is determined whether an Operability Evaluation (in accordance with EN-OP-104, Operability Determination procedure) is required for degraded safety-related equipment. The CR evaluation also determines whether an Operational Decision Making Instruction is required for degraded equipment.

An Operability Evaluation is a formal process that is performed in two phases. The first step is performed by operations as an interim evaluation to determine if there is reasonable assurance that a degraded safety-related system or component can perform its intended safety function. The second step is a more comprehensive evaluation by engineering that justifies operability for the system or component. In some cases the Operability Evaluation specifies compensatory actions that must be completed to ensure operability.

An Operational Decision Making Instruction is a formal process that justifies continued operation of a degraded system or component. It is developed by operations and/or engineering and reviewed by CRG. In some cases the Operational Decision Making Instruction specifies compensatory actions that must be completed to continue to operate the system.

The compensatory actions that must be completed by operations on an interim basis can be included in operator equipment inspection instructions, operator logs, standing orders, turnover sheets and operating procedures. These compensatory actions are discussed by the operating crew at the beginning of each shift.

Condition Report CR-VTY-2007-00132 HPCIC/ RCIC vortexing issue (Operations Work Around) was selected as an Operator Work Around issue. In accordance with Operations Department Standards procedure (DP 0166), an Operator Work Around is defined as any plant condition that significantly affects or could affect abnormal or emergency plant operations or cause operators to take significant compensatory measures beyond the intended design. The Condition Report was initiated in response to the Station's review of Information Notice (IN)-06-021 "Operating Experience Regarding Entrainment of Air into Emergency Core Cooling and Containment Spray Systems". The condition was discovered on January 15, 2007 and the CR was initiated on the same day. The CR was classified as Significance Level "B" (Adverse condition classified as non-significant) with an apparent cause evaluation required. Based on the classification 'B' no root cause was required or requested. This is in accordance with procedure EN-LI-102 Corrective Action Process. There was an Operability Evaluation performed and the systems were determined to be operable. Review of the Operability Evaluation found 2 immediate corrective actions. The first was to maintain Condensate Storage tank level greater than 38 per cent and to ensure that the HPCI/RCIC swap over to the Torus prior to reaching the 38 per cent level. Review of procedure OP 2120 High Pressure Coolant Injection System verified that the changes had been made. The issue was verified as correctly annotated as an Operator Work-Around (OWA) in accordance with procedure DP 0166 Operations Department Standards and verified tracked in the OWA performance indicator. Review of Operation Standing Orders verified that Operations communicated the relevant information.

Based on interviews, review of procedures, and review of the response to unanticipated plant conditions and events for the HPCI System, the processes, procedures and the actions taken were assessed to meet industry standards.

2.2.5 Criterion 5 - Testing

Assessment Response

Multiple interviews with the System Engineer and supervisor were conducted to determine the operating performance of the High Pressure Coolant Injection System over the past three years, with a focus on the past twelve months. As stated in the system description section, the High Pressure Coolant Injection (HPCI) System is an Emergency Core Cooling System (ECCS) and is designed to supply high pressure coolant to the reactor core, to prevent excessive fuel clad temperatures, in the event of a small-break loss of coolant accident (LOCA). As a result, the HPCI System remains in a standby condition during normal plant operations with its suction and discharge water lines filled. Based on being a standby system the assessment focused on surveillance testing results, preventive maintenance and predictive maintenance activities over the past three years.

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As part of the assessment the HPCI System Monitoring Plan (issued September 9, 2008 Rev 3) was reviewed and a component was selected to determine if what the plan identified was in fact what was being tested. HPCI V23-14 (motor operated valve) was identified as requiring a quarterly surveillance test in accordance e with the IST program ENN-SEP-IST-001 and OP 4120. It was concluded that the testing was completed successfully and the testing requirements were consistent with industry standards.

2.2.6 Criterion 6 – Inspection

Assessment Response

HPCI System Inspections consisted of a wide range of activities including the System Health/Component Health Reporting process (EN-DC-143), System Engineering Walk-downs (EN-DC-178) and the System Monitoring Program (EN-DC-159). The output of these core system engineering processes, provide the basis for this assessment.

A review of the past 6 months of system walk-down reports as well as participating and observing a System Engineer during the October 2008 walk-down provided an opportunity to see the actual implementation process. In general, the walk-down was consistent with industry practices based on reviewing the implementation procedure and past reports. Condition Reports were initiated for appropriate material condition issues. It should be noted that with the system in standby mode, there are limited parameters to monitor/observe.

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Review of HPCI System inspection documentation e.g., PM Basis and results revealed that ENVY was scheduling and performing inspections as planned. These inspections are in-line with industry standards.

2.2.7 Criterion 7 – Maintenance

Assessment Response

On-line maintenance of the HPCI System is based on the Limited Condition of Operation (LCO) process, meaning any scheduled on-line maintenance is completed within a predefined time limit prescribed by Technical Specifications. When an LCO is entered, HPCI System unavailability is sometimes impacted. This practice of performing a majority of maintenance on-line is an approved management decision and is consistent with a number of operating plants in the US.

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In addition, no HPCI work activities or PM's were removed from RFO-27 scope.

Based on discussions with the System Engineer, a review of EN-DC-153 (Preventive Maintenance Component Classification) was conducted to determine the basis for how the HPCI system is maintained. It was concluded that the procedural processes utilized by ENVY are consistent with current industry standards and practices**.**

In addition a review of a specific PM basis document (ME032 Rev. 5), was conducted for HPCI V23-14 PM (Motor Operated Valve) and it was concluded that the overall approach and requirements were consistent with industry standards. START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION All data and test results were within the acceptance criteria specified.

2.2.8 Criterion 8 – Repairs

Assessment Response

Repairs to the HPCI System were based on a review of maintenance activities conducted during RFO-26 (refer to Section 2.3.7) and selected CRs which documented specific repair activities that could affect system reliability:

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Review of the information provided indicates that repairs to the HPCI System have been sufficiently in-depth

2.2.9 Criterion 9 – Modifications

Assessment Response

All design changes are controlled by procedure EN-DC-115, Engineering Change Development. That procedure mandates reviews to ensure compliance with regulatory, code, and industry requirements. It also requires that plant documents be updated to reflect any modification changes prior to modification closure and system return to service. The prior procedure, AP 6008, Rev. 3, Vermont Yankee Design Change, contained similar controls. AP 0020, Rev. 78 was also utilized to control temporary and minor modifications to ensure conformance with design intent and to maintain plant configuration and operability requirements.

Interviews with plant engineers and managers confirmed their knowledge of these procedures as well as the system design codes and standards and the design basis for the system. Review of selected design documents, including modification packages, has determined that these procedures are being effectively implemented and the original design requirements and bases are being adequately maintained. Quality Assurance audits are also used to verify procedure compliance.

A number of NRC Component Design Basis Inspection (CDBI) Reports, including one this year, concluded that sufficient design controls are in place to assure the components would meet their intended safety functions. The CDBI Report this year inspected numerous plant components and found three findings of very low risk significance. None of the findings were related to compliance with the original design basis. Details can be found in NRC Component Design Basis Inspection Report 05000271/2008008, dated September 26, 2008. A CDBI Report in 2007 looked at selected components in the HPCI System to verify the HPCI System's ability to meet its design basis. The inspection included HPCI motor operated valves, turbine and pump. There were no findings relative to maintenance of design bases. Details can be found in NRC Component Design Basis Inspection Report 05000271/2006007, dated September 29, 2006.

The EPU Program included a comprehensive analysis of the effect of EPU on the design and operating basis of the plant system and components and evaluated the acceptability of any changes. The review looked at margin changes at both the component and system level. The review analyzed the EPU impact on margins to ensure there were no unacceptable reductions. The analysis was reviewed and approved by every affected organization at the site. The analysis is comprehensive, thorough, and consistent with industry standards. The analysis showed that there are no impacts on the HPCI System as a result of EPU.

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END CONFIDENTIAL INFORMATION

There have been 24 modifications to the HPCI System since original plant construction. Modification Package for Modification 2001-051, HPCI Turbine Exhaust Check Valve Replacement, was reviewed to ensure compliance with the original design bases. Turbine exhaust check valves V23-3 and 4 have affected outage durations due to leak test unreliability resulting from sensitivity of the valves to flange loading. The modification replaced the valves with new valves with thicker walls and soft seats. The review determined that the need to comply with the original design basis was considered and the modification maintained the original design bases.

Based on the document reviews and interviews noted above, all modifications to the HPCI System comply with the system's original design basis. There should be no negative effects on the future reliability of the HPCI system based on design modifications.

2.2.10 Criterion 10 - Redesign

Assessment Response

Engineering Change Closure Procedure EN-DC-118 requires review of all safety significant analysis prior to closure of any plant modification to ensure safety margins have not been reduced. Engineering Margin Management Procedure EN-DC-195 also requires that maintenance of design and operating margins be considered in engineering changes. Discussions with plant engineers and managers confirmed their familiarity with these procedure requirements. Review of plant records confirmed that these requirements are being implemented in a timely fashion. Review of modification packages also confirmed implementation of these requirements. Quality assurance audits are also used to verify procedure compliance.

ENVY has instituted a formal program to track and disposition any potential changes to design and operating margins. The program assigns responsibilities and due dates and is reviewed on a periodic basis at engineering management meetings.

The EPU Program included a comprehensive analysis of the effect of EPU on the design and operating basis of the plant system and components and evaluated the acceptability of any changes. The review looked at margin changes at both the component and system level. The review analyzed the EPU impact on margins to ensure there were no unacceptable reductions. The analysis was reviewed and approved by every affected organization at the site. The analysis is comprehensive, thorough, and consistent with industry standards. The analysis determined that there was no impact on the HPCI System as a result of EPU.

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The NRC conducted a comprehensive inspection of the EPU Program and determined that the components and systems reviewed would be capable of performing their intended safety functions. The NRC also concluded that sufficient design controls had been implemented for design and engineering work, including that related to EPU. Details can be found in NRC Inspection Report 05000271/2004008, dated December 2, 2004.

Modification package for Modification 2001-051, HPCI Turbine Exhaust Check Valve Replacement was reviewed to ensure that safety margins had not been reduced. Turbine exhaust check valves V23-3 and 4 have affected outage durations due to leak test unreliability resulting from sensitivity of the valves to flange loading. The modification replaced the valves with new valves with thicker walls and soft seats. The review determined that appropriate consideration was given to safety margins and no reduction took place as a result of the modification.

Based on the document reviews and interviews noted above, changes made to the HPCI System since it's original construction have been reviewed to ensure that safety margins have not been reduced. There were no HPCI System components modified for Power Up-rate. Repairs, maintenance, or modifications have not impacted the original design of the HPCI Systems. The HPCI System is still "single failure proof". The practices at ENVY pertaining to redesign applied to the HPCI System are meeting industry standard expectations.

2.2.11 Criterion 11 - Seismic Analysis:

Assessment Response

This audit includes the investigation and assessment of ENVY's seismic analysis program. The following scope is proposed to fulfill the intended requirements of the seismic analysis investigation and assessment. The scope of the seismic analysis investigation and assessment includes the following:

1. Review of ENVY's seismic analysis program to determine its approach to seismic design requirements
2. Review of the modification process to determine how seismic design considerations are addressed
3. Review of selected modification packages to determine if appropriate consideration was given to seismic analysis requirements

For selected modifications, NSA will verify that seismic analysis screening criteria were properly applied and that seismic calculations were performed as required. For the selected systems, NSA will review the DBDs and UFSAR to identify their seismic design basis and verify its maintenance in the modification reviews noted above.

The seismic design basis for the plant is documented in Topical Design Basis Document for External Events, Rev 2, dated August 15, 2005. This basis derives from Draft AEC Criterion 2, Performance Standards. The seismic design criteria are based on ground horizontal acceleration of 0.07g for and Operating Basis Earthquake (OBE) and 0.14g for a Safe Shutdown Earthquake (SSE). The vertical acceleration assumed is equal to 2/3 of the horizontal ground acceleration.

The current design of the High Pressure Coolant Injection System is in keeping with its original seismic design basis, including OBE and SSE. All design changes are being controlled by procedure EN-DC-115, Engineering Change Development. That procedure mandates reviews to ensure compliance with regulatory, code, and industry requirements, including seismic requirements.

Interviews with plant engineers and managers confirmed their knowledge of these procedures as well as the system design codes and standards and the original seismic design basis for the system. Review of selected design documents, including modification packages, has determined that these procedures are being effectively implemented and the original seismic design bases are being adequately maintained.

UFSAR Section A.9 provides a description, scope, and design methodology used for the reanalysis of seismic class I piping subsequent to initial operation. HPCI System class I piping was reanalyzed using computer dynamic analyses to evaluate seismic loadings. Ground spectra based Regulatory Guide 1.60 and Regulatory Guide 1.61 damping or floor spectra with ASME code case N-411 damping defined the seismic loading for these piping systems. Seismic analyses were performed for the Safe Shutdown Earthquake scenario and the piping was evaluated to ANSI B31.1 - 1977 code allowables. HPCI torus attached piping was evaluated in accordance with Mark I Containment program Structural Acceptance Criteria Plant Unique Analysis Application Guide. The piping was evaluated to ASME Section III 1977 code allowable.

There have been 24 modifications to the HPCI System since original plant construction. The modification package for Modification 2001-051, HPCI Turbine Exhaust Check Valve replacement, was reviewed to ensure compliance with the original seismic design basis. Turbine exhaust check valves V23-3 and 4 have affected outage durations due to leak test unreliability resulting from sensitivity of the valves to flange loading. The modification replaced the valves with new valves with thicker walls and soft seats. The review determined that the need to comply with the original seismic design basis was considered, was appropriately factored into the design and the original seismic basis was maintained. For additional discussion on maintenance of the seismic design basis see Cooling Tower Assessment. As noted in that assessment, ENVY has utilized finite element analysis computer programs to perform seismic analyses.

Therefore, based on this review, the current design of the HPCI System is in keeping with its original seismic design basis, including OBE and SSE.

2.4.12 Criterion 12 – Training

Assessment Response

The training organization has a process for evaluating all engineering changes to determine whether training is required. This evaluation was performed by conducting interviews with training personnel and by reviewing training processes, governing procedures and training materials. Training is notified of all engineering changes which are entered on the Modification Training Matrix.

The matrix is routinely evaluated to determine if the change requires the training materials for any training program to be modified. If the evaluation determines training may be required a Training Evaluation/Action Request (TEAR) is initiated. The TEAR assigns actions to training personnel to evaluate specific training materials to determine whether the materials require revision. Actions are also assigned to develop training materials and to conduct training as required. The engineering change is also reviewed to determine if any changes to the simulator are required. If a simulator change is required a Discrepancy Report (DR) is generated to implement required changes. All actions are tracked to completion.

Training also has a process to evaluate whether site and industry operating experience (OE) is to be incorporated into related instructor guides (IG) for classroom and simulator training. Interview with the Operations Training Superintendent revealed an expectation that operating experience is to be included in the development and revision of training material. There is a general understanding of the Entergy Operating Experience procedure (EN-OE-100) including interface with the Site OE Coordinator. There is also a proceduralized requirement for use in the development for training material. Systematic Approach to Training Process procedure (EN-TQ-201) reinforces the need to consider using Operating Experience to re-enforce learning objectives.

Modification 04-1273 (High Pressure Coolant Injection (HPCI) Suction Valve Modification) was selected based on review of the modification. Review of the modification found that training would be required. This modification provided a manual switch to allow the operator to manually initiate the HPCI Auto Isolation if required. The Operations Training Department reviewed this modification and initiated Training Evaluation Action Request (TEAR) VTY 2006-116. This initiated a NEEDs analysis which identifies if training is required and if so, what Training Department Information Guides (IG) and/or changes to the Simulator would be required. There are then actions assigned to address the changes. The following Instructor Guides were verified complete and reflect the changes made under modification ER 04-1273: Licensed Operator Re-qualification (LOR) LOR-00-605 for modification training, Licensed Operator Training (LOT) LOT-00-206, LOT-00206H, Auxiliary Operator Re-qualification AOR-25-605 and Licensed Operator Training LOT-01-223 Primary Containment Isolation System (PCIS).

Discrepancy Report (DR) 06-0069 was initiated by the Training Department to review the modification documents for any required changes to the Simulator. Review of DR 06-0069 actions found that remote isolation capability on Condensate Storage Tank low level documented that the changes to the Simulator had been completed and the modification acceptance test had been completed. DR 06-0069 was closed February 15, 2007. Modification training was verified through Training department tracking documentation to have been completed during the September through November 2006 time frame.

Systems Training is required every 2 years (per interview with Operations Training Manager). Operating Experience references were validated to have been referenced in each Instructor Guide. Licensed Operator Training Instructor Guide LOT-00 206 High Pressure Coolant System referenced the following Operating Experience documents: Significant Operations Event Records 83-3 and 82-8, Operating Experience 3107 and one internal Condition Report. Procedures OP 2120 High pressure Coolant Injection System, OP 2115 Primary Containment and OE 3107 Emergency Operating Procedure were reviewed and the appropriate modification changes were verified to have been incorporated.

Operations Training performance in the review and implementation of modification ER 04-1273 (HPCI Suction Valve Modification) meets expectations.

As a result of review of the process and interviews with Training department management, along with review of the actions taken in response to this particular modification, it is concluded the Training department meets industry standards with respect to evaluating modifications and taking the appropriate actions to incorporate the required changes into training material.

2.4.13 Criterion 13 - Corrective Action Program

Assessment Response

An overall assessment of the corrective action program and its effectiveness was completed and is documented in section 1.2.5 that addresses the Criterion 13 (Corrective Action Program) questions for the overall Corrective Action Program.

A list of CRs back to the year 2000 was reviewed. Certain CRs from this list are discussed in detail in the sections above.

Based on these reviews it was determined that issues related to the HPCI system are being identified and entered into the Corrective Action Process. Corrective actions were assigned to appropriately address these issues.

The HPCI System CR review supports the conclusion of Section 1.2.

2.2.14 *Criterion 14 – Standard Review Plan*

A special assessment was also performed of the design basis of the HPCI System against the design bases of new reactive plants. As noted previously, the design basis of the HPCI System is contained in the HPCI DBD. The design basis requirements for current plants are contained in NUREG-0800, U. S. Nuclear Regulatory Commission Standard review Plan, Revision 3, March 2007. This assessment was performed by comparing the General Design Criteria (GDC) that was applied to ENVY during its licensing period against the GDC that apply today.

The results of that assessment show that all of the GDC that are applicable today for the HPCI System were applicable to ENVY with two exceptions. These two exceptions are GDC 5 and 27. GDC 5 applies to multiple unit sites and so is not applicable to VY. Likewise, GDC 27 applies to PWRs and so is also not applicable to ENVY.

HPCI System Conclusion

Based on the document reviews and interviews noted above the following is a summary of HPCI System assessment conclusions. More detail is provided at the end of each individual section above.

The ENVY engineering design process is well documented, controlled and consistent with industry practice. The current design of the HPCI System is consistent with its original design basis and is adequately reflected in plant records and procedures.

ENVY has formal processes for identifying and correcting unanticipated operations outcomes. The corrective action system tracks the implementation of corrective and compensatory measures for the HPCI System when issues arise, including the revision of appropriate documents. These processes are consistent with industry practice.

ENVY has formal processes to ensure that when engineering changes are implemented for the HPCI System that reviews are conducted and changes made as required to operating procedures, training materials and the simulator to accurately reflect plant conditions. These processes are consistent with industry practices.

With respect to the focus on effective equipment reliability, the process and procedures that support the HPCI System are consistent with current commercial nuclear power's standards for safe and reliable operation. In addition, these processes/procedures are effectively applied to the HPCI System. In particular, the System Engineer is a qualified an experienced individual who has many years of HPCI specific experience and effectively applies the applicable process/procedures.

HPCI System issues, as described in the individual sections above, are being entered into the Corrective Action Process. Corrective actions were appropriately assigned to address those issues

References

Procedures and Specifications:

1. EN-DC-115, Rev. 5, Engineering Change Development
2. EN-DC-118, Rev. 2, Engineering Change Closure
3. EN-DC-126, Rev. 1, Engineering Calculation Process
4. EN-DC-141, Rev. 5, Design Inputs
5. EN-DC-143, System Health
6. EN-DC-195, Rev. 2, Margin Management
7. EN-DC-313, Rev. 2, Procurement Engineering Process
8. EN-LI-113, Rev. 3, Licensing Basis Document Change Process
9. EN-DC-105, Rev. 2, Configuration Management Program
10. AP 6008, Rev. 3, Vermont Yankee Design Change
11. AP 0020, Rev. 78, Control of Temporary and Minor Modifications

Documents

1. Vermont Yankee UFSAR, Rev. 20
2. License Renewal Application, dated January 25, 2006
3. Power Uprate Licensing Application dated, September 10, 2003
4. NRC Safety Evaluation for License Renewal, March, 2007
5. NUREG-0800, U. S. Nuclear Regulatory Commission Standard Review Plan, Rev. 3, March 2007.
6. Modification Package for Modification 2001-051, HPCI Turbine Exhaust Check Valve Replacement
7. Alternate Cooling Systems, Rev. 29, July 20, 2007
8. DBD for High Pressure Coolant Injection System, Rev. 32, May 6, 2008
9. Topical DBD for External Events, Rev. 2, August 15, 2008
10. Topical DBD for Safety Analysis, Rev. 4, September 29, 2005
11. DBD for Accident - Event Combinations Topical, Rev. 6, June 29, 2007
12. Task Report T0404, High Pressure Coolant Injection
13. Vermont Yankee Response to Request for Information Pursuant to 10 CFR50.54(f)Regarding Adequacy and Availability of Design Bases Information, BVY 97-23, dated February 14, 1997.
14. Validation Report Design Basis Document HPCI, dated November 27, 1998

Drawings

1. Flow Diagram High Pressure Coolant Injection System, Sheet 1, Rev. 50, November 30, 2007
2. Flow Diagram High Pressure Coolant Injection System, Sheet 2, Rev. 43, December 6, 2004

NRC Inspection Reports

1. NRC Inspection Report 05000271/2004008, dated December 2, 2004
2. NRC Component Design Basis Inspection Report 05000271/2008008, dated September 26, 2008
3. NRC Component Design Basis Inspection Report 05000271/2006007, dated September 29, 2006
4. NRC Inspection Report 05000271/2007006

Condition Reports

1. CR-VTY-2007-00556, Dated February 21, 2007
2. CR-VTY-2007-03425, Dated September 5, 2007
3. CR-VTY-2007-00132, Dated January 15, 2007

2.3 Residual Heat Removal (RHR) System

System Description

The RHR System consists of two closed loops, each loop containing two pumps in parallel, one heat exchanger, and the necessary valves and instrumentation. The RHR heat exchanger in each loop is cooled by the RHR Service Water (RHRSW) System which consists of two pumps in parallel that take suction from the Station Service Water System or the deep basin beneath the west-cooling tower. The RHR System is designed to remove decay heat energy from the reactor under both operational and accident conditions.

The modes of operation of the RHR System are the following:

- Low Pressure Coolant Injection (LPCI) Mode
- Containment Spray Cooling (Drywell Spray and Torus Spray) Mode
- Suppression Pool Cooling Mode
- Shutdown Cooling Mode
- Augmented Fuel Pool Cooling Mode
- Emergency Reactor Vessel Fill (RHRSW Intertie) Mode
- Alternate Shutdown Mode

The LPCI Mode of RHR takes suction from the suppression pool and injects flow into the core region of the reactor vessel through one of the two reactor recirculation loops. The LPCI Mode is typically identified as ‘short-term’, which is approximately the first ten minutes from LPCI initiation. Any time after this is typically identified as ‘long-term’. The RHR heat exchanger bypass valves are open to permit full injection flow during initial LPCI mode. They can be closed after a time delay to permit suppression pool cooling. This mode of operation is designed to restore and maintain the water level of the reactor vessel following a loss of coolant accident.

The Containment Spray Cooling Mode of RHR takes suction from the suppression pool and injects flow into spray headers located in the drywell and suppression chamber. This mode of operation is designed to reduce containment pressure and temperature following a loss of coolant accident by cooling any non-condensables and condensing any steam which may be present.

The Suppression Pool Cooling Mode of RHR takes suction from the suppression pool, passes it through the RHR heat exchangers, and returns flow to the suppression pool. This mode of operation is designed to remove heat, which has been added to the suppression pool.

The Shutdown Cooling Mode of RHR takes suction from the reactor vessel via the Reactor Recirculation "A" loop suction piping, passes it through the RHR heat- exchangers, and returns flow to the reactor through the recirculation lines. This mode of operation is designed to remove sensible and decay heat from the reactor during shutdown.

The Alternate Shutdown Cooling Mode provides a cooling path if the normal Shutdown Cooling path is inoperable. The RHR pumps take suction from the suppression pool, pass it through the RHR heat exchangers and inject into the vessel via the RHR injection valves. The Safety Relief Valves (SRVs) on the reactor vessel are open to allow overflow to the suppression pool. This mode can be performed from the Control Room.

The Augmented Fuel Pool Cooling Mode takes suction from the Fuel Pool Cooling System, passes it through the RHR heat exchangers, and returns flow to the Fuel Pool Cooling System. This mode of operation is designed to assist in fuel pool cooling during reactor shutdown periods and Alternate Cooling System operation. The Emergency Reactor Vessel Fill mode of RHR provides a crosstie between the RHR Service Water System and the A loop of RHR piping. The RHRSW pumps take suction from the Service Water System and inject flow into the reactor vessel through the RHR piping. This mode of operation is designed to provide a source of water to keep the reactor core covered (and fill containment) in the event that Core Standby Cooling Systems (CSCS) pumps are lost due to loss of containment pressure or adequate core cooling cannot be assured. This mode is considered a "beyond" design basis mode of operation.

The Alternate Shutdown Mode of RHR uses the RHR alternate shutdown panel to control the minimum valving required for vessel injection, torus cooling and shutdown cooling modes. This mode of operation is designed to achieve and maintain cold shutdown conditions during a postulated fire in the Control Room or Cable Vault, which eliminates the normal means of control of the system.

2.3.1 Criterion 1 - Initial Conditions:

Assessment Response

ENVY has verified and documented the original design codes and standards as well as the current design bases for the RHR System in 'Vermont Yankee Nuclear Power Station Design Basis Document (DBD) For Residual Heat Removal System', Revision 22, dated July 10, 2007. The DBD verification also included an as-built walk down of the RHR System to confirm compliance with the design basis. Details can be found in "Validation Report RHR", dated June 19, 1998. The design bases for the RHR System are documented in the ENVY UFSAR. As part of ENVY's response to NRC letter 10 CFR 50.54(f), the station committed to a complete FSAR verification program to ensure consistency between the FSAR and plant documentation.

Those codes and standards are as follows:

- IEEE 279-1968, Proposed IEEE Criteria for Nuclear Power Plant protection systems
- USAS B31.1.0, Power Piping Code 1967
- ASME Boiler and Pressure Vessel Code Section III, Nuclear Power Plant Components
- Standard of the Hydraulic Institute
- ASME Boiler and Pressure Vessel Code section XI, Rules for the In-service Inspection of Nuclear Power Plant Components

The design basis for the RHR system is also documented in the DBD. The design basis consists of system design bases and major component design bases. The system design bases include the following:

- System functions
- Regulatory requirements
- System design requirements
- Operating requirements
- Electrical and instrument and control requirements

Major component design bases are typically performance requirements such as pump flow rate with developed head. They are too numerous to list here but are discussed in detail in the DBD.

As other NRC regulations and regulatory requirements and requests evolved ENVY evaluated its compliance with these regulations or requirements to assure the NRC of its compliance with their intent. Those that were applicable to the RHR System ranged from NRC regulations in 10 CFR 50, Regulatory Guides, to Generic Letters and I.E. Bulletins. All of these are also discussed in the RHR System Design Basis Document without any exemptions or deviations noted. As part of the DBD process, the RHR System was walked down to verify its conformance with the current design basis.

It should be noted that the RHR System is a safety-related system and as such is subject to strict controls by the NRC. Any deviations or exemptions from the original design basis are not permitted without prior approval by the NRC. Likewise, the system is subject to Technical Specification limits and failure to comply with those limits could cause the unit to be shut down. Thus, non-compliance with these requirements could have an impact on unit reliability and for this reason is subject to tight controls by the plant.

It should also be noted that the RHR System is an Emergency Core Cooling System. It does not operate during normal unit operation but rather is in a standby mode in case of an accident. There are, therefore, few challenges to the operability of the system.

Design changes are being controlled by procedure EN-DC-115, Engineering Change Development. That procedure mandates reviews to ensure compliance with regulatory, code, and industry requirements. It also requires initiation and processing of updates to plant configuration documents and plant operating and design margins. Procedure EN-DC-118 requires that all design basis documents be updated prior to modification closure.

ENVY utilizes a fleet-wide Configuration Management Program, EN-DC-105, Rev. 2. The purpose of the procedure is to ensure consistency between the design requirements, physical configuration, and plant configuration information. This program is also used for primary suppliers of equipment and software for ENVY.

Interviews with plant engineers and managers confirmed their knowledge of these procedures as well as the system design codes and standards and the design basis for the system. Review of selected design documents has determined that these procedures are being effectively implemented and the original design requirements and bases maintained.

Various NRC inspection reports support this assessment. A number of NRC CDBI Reports, including one this year, concluded that sufficient design controls are in place to assure the components will meet their intended safety functions. The CDBI Report for this year covered numerous plant components and contains 3 findings of very low risk significance. None of the findings were related to maintenance of the original design bases. Details can be found in NRC Component Design Basis Inspection Report 05000271/2008008, dated September 26, 2008. A comprehensive CDBI Report conducted in 2007 looked at selected components in the RHR System to verify the RHR System's ability to meet its design basis. The inspection looked at the RHR pump, heat exchanger, injection valve and heat exchanger bypass valve. There were no findings in the inspection report. Details of this inspection can be found in NRC Component Design Basis Inspection Report 05000271/2006007.

The EPU Program included a comprehensive analysis of the effect of Power Uprate on the design and operating basis of the plant systems and components and evaluated the acceptability of any changes. The review looked at margin changes at both the component and system level. The review analyzed the EPU impact on margins to ensure there were no unacceptable reductions. The analysis was reviewed and approved by every affected organization at the site. The review is comprehensive, thorough, and consistent with industry standards. The review determined that there were three items impacting RHR System margins as a result of EPU. They are discussed below.

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END CONFIDENTIAL INFORMATION

The NRC reviewed selected components in the RHR System for acceptable operation under up-rated conditions. The details can be found in NRC Inspection Report 05000271/2004008, dated December 2, 2004. There were no findings.

Based on the document reviews and interviews noted above, the design of the RHR System is in keeping with the expected initial conditions and current design basis. There should be no negative effects on the future reliability of the RHR system based on design modifications.

2.3.2 Criterion 2 – Procurement

Assessment Response

The ENVY processes governing engineering changes, calculations, procurement etc., referenced below, require that applicable calculations be performed to support the engineering changes, that the changes be compared against the original design requirements and design bases, and that they be verified as built prior to return to service of the implemented engineering change. The procedure for performance of calculations, EN-DC-126, prescribes the requirements and the format for the preparation of the calculations. Discussions with plant engineers and managers confirmed their familiarity with these procedure requirements. It has been verified that this procedure was used for RHR System procurement changes. Quality Assurance audits are also used to verify procedure compliance.

There have been 30 modifications to the RHR System since original plant construction. None of those modifications are procurement changes. However, a review of Modification ENVYDC 2003-016, Alternate Source Term, verified that new calculations were performed and those calculations were compared against the original design basis. This modification did impact the RHR design bases and is considered a procurement change.

Based on the document reviews and interviews noted above, new sets of review calculations were completed for the respective procurement changes and the procurement changes were compared against the original design and all of its calculations. There should be no negative effects on the future reliability of the RHR system based on the process for maintaining procurement records.

2.3.3 Criterion 3 – Installation

Assessment Response

The Engineering Change Closure procedure sets forth the requirements of updating pertinent engineering, operations, training, maintenance, program, and licensing documentation. The procedure specifies which updates are required prior to return to service and which are required to be tracked and completed after return to service. Discussions with plant engineers and managers confirmed their familiarity with these procedure requirements. The procedure specifically requires that plant calculation changes be verified as-built prior to returning the system to service. Quality Assurance audits are also used to verify procedure compliance. Reviews of plant records indicate these procedures are being implemented and in a timely fashion.

Inspection of the UFSAR, DBD, and flow diagrams (P&ID) of the RHR System found that the system changes required for EPU were properly reflected in these documents. Modification package reviews for the RHR System also verified that system changes are being properly reflected in these design documents.

The NRC has also performed an inspection of the EPU program and concluded that sufficient design controls are in place and are being implemented and therefore plant records should adequately reflect the as-built condition of the plant. Details can be found in NRC Inspection Report 05000271/2004008, dated December 2, 2004.

In February 1997, ENVY responded under oath to the adequacy and availability of design bases. In its response ENVY stated it has reasonable assurance that the ENVY design bases have been adequately translated to the plant design and procedures and that plant configuration is in an appropriate manner. ENVY went on to state that it was committing to provide improved configuration management including completion of a Design Basis Documentation program, improved Technical Specifications program and a FSAR Verification program. The subsequent development of the DBDs, which took place in 1997 and 1998, included a DBD validation process in which validation teams verified both the design basis information and its application to design documents, and its application to operations, maintenance, surveillance, testing functions and physical configuration. The physical configuration validation included as-built walk downs. Walk downs were performed to verify that the system configuration reflects the design basis, is in agreement with plant drawings, and the component labeling is adequate.

Based on the document reviews and interviews noted above, RHR System plant records adequately represent the as-built condition of the plant. RHR System changes are reflected in documents from the design basis through as-built and through current operation. The RHR System at ENVY meets industry expectations as it pertains to the installation criteria for the RHR System.

2.3.4 Criterion 4 – Operation

Assessment Response

Unanticipated operations outcomes for this assessment considered plant operating or equipment conditions that did not result in the desired or expected outcome. Interviews with operations management were conducted in order to review the processes used to identify and correct each unexpected operational issue. How Operations implements the governing procedures and processes was discussed with operations management. An example was selected of an event within the Residual Heat Removal (RHR) system, and reviewed to ensure appropriate compensatory actions were taken, and that plant procedure deficiencies were addressed as appropriate. Additionally, it was evaluated to determine if the appropriate level of analysis was performed in accordance with the corrective action system.

As part of the CR evaluation it is determined whether an Operability Evaluation (in accordance with EN-OP-104, Operability Determination procedure) is required for degraded safety-related equipment.

The CR evaluation also determines whether an Operational Decision Making Instruction is required for degraded equipment.

An Operability Evaluation is a formal process that is performed in two phases. The first step is performed by Operations as an interim evaluation to determine if there is reasonable assurance that a degraded safety-related system or component can perform its intended safety function. The second step is a more comprehensive evaluation by Engineering that justifies operability for the system or component. In some cases the Operability Evaluation specifies compensatory actions that must be completed to ensure operability.

The compensatory actions that must be completed by Operations on an interim basis can be included in operator equipment inspection instructions, operator logs, standing orders, turnover sheets and operating procedures. These compensatory actions are discussed by the operating crew at the beginning of each shift.

Condition Report CR-VTY-2005-03586 Unexpected Primary Containment Isolation System (PCIS) Group 1, 2,3,4, and 5 Isolations was selected as an event driven action taken in response to an unanticipated plant event. On November 4, 2005 the plant experienced a loss of Shutdown Cooling when both logic trains were down-powered resulting in multiple system isolations. Following recovery and restart of the Shutdown Cooling System the condition report was initiated. The CR was classified as Significance Level "A" (Significant and requires a root cause). This is in accordance with procedure EN-LI-102 Corrective Action Process. The initiation of the group isolations resulted in the loss of the in-service Shutdown Cooling Loop. The immediate actions were reviewed and verified as appropriate. The resultant Root Cause Analysis was completed and reviewed/accepted by the Corrective Actions Review Board (CARB) December 5, 2005 and the Operations Manager. The completed Root Cause was sent to the Entergy Fleet Operation Experience Coordinators on December 5, 2005. Corrective action to prevent recurrence involved the incorporation of the requirement to place Shutdown Cooling Critical Plant Equipment signs and Robust Barriers in place for the Shutdown Cooling Loop to be placed in service to ensure that the equipment remains protected from a similar occurrence. Procedure OP 2124 Residual Heat Removal System was reviewed and found that the recommended changes have been appropriately incorporated. Review of Information Guide Licensed Operator Training (LOT) 01-300 Pre-outage Shutdown and Startup Overview Lesson Plan validated that review of CR-VTY-2008-03425 and lessons learned have been incorporated. The performance review of documentation finds that the actions taken meet expectations.

Based on interviews, review of procedures, and review of the response to unanticipated plant conditions and events for the RHR systems it was evaluated that the processes, procedures, and the actions taken meet industry standards.

2.3.5 Criterion 5 - Testing

Assessment Response

Samples of documents were reviewed and interviews were conducted relating to testing associated with the Residual Heat Removal System. This testing includes tests in the System Trending and Monitoring Plan such as those required by the In Service Testing (IST) program which requires periodically testing certain pumps/motors and valves. START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION

The RHR System Engineer has over 11 years of experience. During interviews with the System Engineer, it was observed that the System Engineer was familiar with items on the System Monitoring Plan. The System Engineer was able to electronically display the basis for testing in the Preventive Maintenance (PM) Basis Document as well as other information. This included an explanation of how components are classified on their system. This is important since component classifications drive which components are tested. As an example, the System Engineer explained how the P-8-1A pump was classified. The classification was HLM, 'H' for high critical because if this pump is not operable the plant would enter a shutdown Limited Condition for Operation (LCO). 'L' represents a low duty cycle since this pump runs only during quarterly tests and during outages. 'M' represents mild operating conditions since each of the RHR rooms have temperatures maintained by coolers.

The System Engineer exhibited an understanding of testing, and his responsibilities relating to testing of the system which included periodic review of testing results. This was validated by review of test results discussed earlier in this section.

The review of documentation relating to testing, including results and discussions with the current System Engineer did not reveal any issues that would indicate ENVY was not scheduling and carrying out planned testing on the RHR System. The types of testing performed are inline with industry practices and standards. This includes component classification, system-monitoring plans, predictive maintenance tasks etc. discussed above.

One item relating to testing that could be improved is the interface between the System Engineer and other areas in Engineering.

2.3.6 Criterion 6 – Inspection

Assessment Response

Inspections on the RHR system include those specified in the System Trending & Monitoring Plan, PM Program, System Walk Downs, System Health Reporting, and Component Health Reporting processes. These inspections include physical inspection of certain components and system data review for reports, e.g. System Health.

START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION

The RHR System Engineer is required to perform periodic walk-downs of the system per Entergy Procedure EN-DC-178. These walk-downs include monitoring system parameters, looking for material degradation, and reviewing data such as outstanding work orders. The RHR System Engineer was requested to perform a system walk-down by NSA, and members of the NSA team accompanied the System Engineer on this walk down. EN-DC-178 includes a checklist for preparing for and conducting a system walk down. The RHR System Engineer did not use or refer to the checklist. When questioned he mentioned that he is well aware of what was included on the checklist since he has been performing them for many years. During the walk-down, the System Engineer looked for material degradation. However, when questioned concerning the use of EPRI Aging Assessment Field Guide, which other System Engineers use or reference during walk-downs, he was not familiar with the Guide. Complete monitoring of system parameters was not possible since the system was not operating. Additional monitoring of system parameters is performed during quarterly tests of the system. The one area that was not discussed by the System Engineer is the review of outstanding work orders and operator rounds. The NSA inspection team noted during the walk-down that overall the housekeeping in RHR rooms was acceptable.

During the interview with the RHR System Engineer he exhibited a good understanding of inspections on his system. This was revealed by his explanations of the basis for certain inspections and during discussions relating to RHR System Health. One area he did not have a full understanding was the PM Program. As an example, he was not familiar with condition codes that are entered on completed work orders, which are used for revising PM tasks. The responsibility for entering these codes and trending them is the PM Program Manager. However, the System Engineer is required to approve any PM task changes.

Also, during interviews the System Engineer was requested to discuss his System Notebook. The purpose of this notebook per Entergy Engineering Standard EN-MS-S-001-Multi is to provide a summary source of system information used by the System Engineer to better support peers. Another intention of the notebook is to provide new System Engineers with a useful source of information. Review of the RHR System Engineer notebook revealed that not all documents were up-to-date, i.e. not the latest revision. The System Engineer indicated they were for reference only and should it be required he would access the latest revision. Comparison to the Entergy Standard revealed inconsistencies in content. This was the case with other system notebooks reviewed. The expectation for content does not appear to be clear among System Engineers.

Review of RHR System inspection documentation (e.g. PM Basis and results) revealed that ENVY was scheduling and performing inspections as planned. The good performance documented on the RHR System Health Reports also indicates ENVY is scheduling and performing inspections as planned. These inspections are in-line with industry standards, which is indicated by the discussions above relating to PM Basis documents, preventive maintenance tasks, system and component health reporting, etc.

Areas relating to inspections that could be improved were mentioned above. The use of the system walk down checklist would help ensure consistency in walk down performance. Consistency in content and updating of notebooks should ensure the intent of the Engineering standard is met. Improving both areas will require clarity from management on expectations and monitoring of these items. Taking these actions will be important in the future should a new less experienced RHR System Engineer be performing these tasks.

2.3.7 Criterion 7 – Maintenance

Assessment Response

Several programs, processes, and initiatives contribute to ENVY managing assets to ensure reliability. This includes managing aging components and parts obsolescence.

Entergy Nuclear Management Manual EN PL 170 Nuclear Asset Management Planning describes the process for addressing aging components. This manual references INPO AP 913, which is an industry standard that includes long term asset management recommendations. START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION Also planned is the refurbishment of one of the replaced motors to be available as a spare in the future. This is an example of ensuring critical spares are available. These actions contribute to a sound long-range plan for RHR reliability.

System Engineers have access to an EPRI Aging Assessment Field Guide 1007933. This Guide is in a booklet form that can easily be carried and referenced during walk downs. The Guide provides information on how to detect and evaluate aging-related degradation. Metal, Concrete, Coating, Lubricant, and Mechanical and Electrical Component degradation are covered. Interviews with System Engineers indicate that use of this guide is inconsistent. As mentioned previously the RHR System Engineer does not use this guide.

Actions to address obsolete parts are described in Entergy Procedure EN-DC-320 Identification and Processing of Obsolete Items. These actions include the quarterly review of the industry obsolete inventory database (OIRD). This database lists obsolete parts and possible replacements. Another industry source for identifying obsolete parts is Operating Experience (OE). Entergy Procedure EN-OE-100 governs the OE Program. OE from the Institute of Nuclear Power Operations (INPO), Nuclear Regulatory Commission (NRC), Owners Groups, and Vendors is evaluated per the actions described in the OE Program procedure. Obsolete parts identification also occurs during system single point failure reviews and during work planning. An RHR system example of identifying and resolving an obsolete part was the replacement of analog flow indicators with digital indicators since indicators were no longer available in the market.

Certain ENVY initiatives also contribute to addressing aging components. The Extended Power Uprate (EPU) in 2003 included a detailed evaluation of the RHR system operation to ensure its operability at higher power levels. This included review of existing components. Additionally, as part of ENVY's application to the NRC for License Renewal, ENVY committed to implementing a comprehensive Aging Management Program by 2012. This program has been reviewed and approved by the NRC and is consistent with industry standards.

Review of RHR System CRs back to the year 2000 did not reveal any issues identified relating to aging of components or parts obsolescence.

The review described above indicates that ENVY has the programs and processes in place to address the management of aging components as demonstrated by the RHR System pump and motor replacements. Another section of this report discusses the management of aging components further.

At the System level, one item that could improve the management of aging components is the consistent use of the EPRI Aging Assessment Field Guide. During System Engineer interviews it was mentioned that training in the use of this guide was provided. However, the expectation for use of the Guide during walk-downs is not clear, and the use of it during walk-downs is inconsistent among System Engineers.

2.3.8 Criterion 8 – Repairs

Assessment Response

Programs that direct testing and inspections can positively impact the amount and type of repairs required for system reliability. Previous sections detailed what testing and inspections are performed on the RHR System.

Repairs at ENVY are prioritized and scheduled per the Work Management Process. This includes work while the unit is on-line and during outages. Numerous procedures govern the work management process. All areas needed for typical industry programs are addressed. Review of ENVY Work Management Process is included in another section of this report.

Overall, ENVY has a low corrective maintenance backlog, which indicates that repairs are completed in a timely manor. START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION

ENVY should consider in its planning for future Refuel Outages identifying valves, through trending, that have the potential for failing their local leak rate test. Contingency work orders can then be planned to repair valves during the Refuel Outage instead of deferring them for future outages.

2.3.9 Criterion 9 – Modifications

Assessment Response

All design changes are controlled by procedure EN-DC-115, Engineering Change Development. That procedure mandates reviews to ensure compliance with regulatory, code, and industry requirements. It also requires that plant documents be updated to reflect any modification changes prior to modification closure and system return to service. The prior procedure, AP 6008, Rev. 3, Vermont Yankee Design Change, contained similar controls. AP 0020, Rev. 78 was also utilized to control temporary and minor modifications to ensure conformance with design intent and to maintain plant configuration and operability requirements. Interviews with plant engineers and managers confirmed their knowledge of these procedures as well as the system design codes and standards and the design basis for the system. Review of selected design documents, including modification packages, has determined that these procedures are being effectively implemented and the original design requirements and bases are being adequately maintained. Quality Assurance audits are also used to verify procedure compliance.

Various NRC inspection reports, such as those referenced previously, have confirmed this conclusion. A number of NRC CDBI Reports, including one this year, concluded that the sufficient design controls are in place to assure the components will meet their intended safety functions. The CDBI Report for this year addressed numerous plant components and contained 3 findings of very low risk significance. None of the findings were related to compliance with the original design bases. Details can be found in NRC Component Design Basis Inspection Report 05000271/2008008, dated September 26, 2008. A comprehensive CDBI Report conducted in 2007 looked at selected components in the RHR System to verify the RHR System's ability to meet its design basis. The inspection looked at the RHR pump, heat exchanger, injection valve and heat exchanger bypass valve. There were no findings. Details of this inspection can be found in NRC Component Design Basis Inspection Report 05000271/2006007.

The EPU Program included a comprehensive analysis of the effect of EPU on the design and operating basis of the plant system and components and evaluated the acceptability of any changes. The review looked at margin changes at both the component and system level. The review analyzed the EPU impact on margins to ensure there were no unacceptable reductions. The analysis was reviewed and approved by every affected organization at the site. The analysis was comprehensive, thorough and consistent, with industry standards. The analysis determined there was no impact on the RHR System as a result of EPU.

START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION

There have been approximately 30 modifications to the RHR System since the original design. A review of one of those modifications, Modification ENVYDC 2003-016, Alternate Source Term, verified that the design was compared against the original design basis and that original basis has been maintained.

Based on the document reviews and interviews noted above, all modifications to the RHR System comply with the system's original design basis. There should be no negative effects on the future reliability of the RHR system based on design modifications.

2.3.10 Criterion 10 – Redesign

Assessment Response

Engineering Change Closure Procedure EN-DC-118 requires review of all safety significant analysis prior to closure of any plant modification to ensure safety margins have not been reduced. Engineering Margin Management Procedure EN-DC-195 also requires that maintenance of design and operating margins be considered in engineering changes. Discussions with plant engineers and managers confirmed their familiarity with these procedure requirements. Review of plant records confirmed that these requirements are being implemented in a timely fashion. Review of modification packages also confirmed implementation of these requirements. Quality Assurance audits are also used to verify procedure compliance.

ENVY has instituted a formal program to track and disposition any potential changes to design and operating margins. The program assigns responsibilities and due dates and is reviewed on a periodic basis at engineering management meetings.

The EPU Program included a comprehensive analysis of the effect of EPU on the design and operating basis of the plant system and components and evaluated the acceptability of any changes. The review looked at margin changes at both the component and system level. The review analyzed the EPU impact on margins to ensure there were no unacceptable reductions. The analysis was reviewed and approved by every affected organization at the site. The analysis is comprehensive, thorough, and consistent with industry standards. START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION The third impact identified was the need to take credit for containment over-pressurization in order to assure adequate NPSH for the RHR pumps in their emergency core cooling mode. Credit for over-pressurization of containment has been approved by the NRC with net result of an increase in pump NPSH margins. Thus the original design requirements and basis of the system has been maintained for the EPU conditions.

The NRC conducted a comprehensive inspection of the EPU Program and determined that the components and systems reviewed would be capable of performing their intended safety functions. They also concluded that sufficient design controls had been implemented for design and engineering work, including that related to EPU. Details can be found in NRC Inspection Report 05000271/2004008, dated December 2, 2004.

A review of Modification ENVYDC 2003-016, Alternate Source Term, verified that safety margins were considered in the design and they were not reduced as a result of the modification.

Based on the document reviews and interviews noted above, changes made to the RHR System since its original construction have been reviewed to ensure that safety margins have not been reduced. Each RHR System component modified for uprate has been reviewed to assure that operational margins have not been reduced and to assure that design basis redundancy has not been compromised. Repairs, maintenance, or modifications have not impacted the original design of the redundant safety systems.

All systems that are required to be 'single failure proof' remain 'single failure proof'. The practices at ENVY pertaining to redesign applied to the RHR System meet industry expectations.

2.3.11 Criterion 11 - Seismic Analysis

Assessment Response

The agreement with the Vermont Department of Public Service on the scope of work for this assessment states the following regarding seismic analysis:

This audit includes the investigation and assessment of ENVY's seismic analysis program. The following scope is proposed to fulfill the intended requirements of the seismic analysis investigation and assessment. The scope of the seismic analysis investigation and assessment includes the review of the following:

- Review ENVY's seismic analysis program to determine the approach to seismic design requirements
- Review the modification process to determine how seismic design considerations are addressed
- Review selected modification packages to determine if appropriate consideration was given to seismic analysis requirements

For selected modifications, NSA will verify that seismic analysis screening criteria were properly applied and that seismic calculations were performed as required. For the selected systems NSA will review the DBDs and UFSAR to identify their seismic design basis and verify its maintenance in the modification reviews noted above.

The seismic design basis for the plant is documented in Topical Design Basis Document For External Events, Rev 2, dated 8/15/05. This basis derives from Draft AEC Criterion 2, Performance standards. The seismic design criteria are based on ground horizontal acceleration of 0.07g for and Operating Basis Earthquake (OBE) and 0.14g for a Safe Shutdown Earthquake (SSE). The vertical acceleration assumed is equal to 2/3 of the horizontal ground acceleration.

The current design of the Residual Heat removal System is in keeping with its original seismic design basis, both for OBE and SSE. All design changes are being controlled by procedure EN-DC-115, Engineering Change Development. That procedure mandates reviews to ensure compliance with regulatory, code, and industry requirements, including seismic requirements.

UFSAR Section A.9 provides a description, scope, and design methodology used for the reanalysis of seismic class I piping subsequent to initial operation. RHR System class I piping was reanalyzed using computer dynamic analyses to evaluate seismic loadings. Ground spectra based Regulatory Guide 1.60 and Regulatory Guide 1.61 damping or floor spectra with ASME code case N-411damping defined the seismic loading for these piping systems. Seismic analyses were performed for the Safe Shutdown Earthquake scenario and the piping was evaluated to ANSI B31.1 - 1977 code allowables.

RHR torus attached piping was evaluated in accordance with Mark I Containment program Structural Acceptance Criteria Plant Unique Analysis Application Guide. The piping was evaluated to ASME Section III 1977 code allowables.

Interviews with plant engineers and managers confirmed their knowledge of these procedures as well as the system design codes and standards and the original seismic design basis for the system. Review of selected design documents has determined that these procedures are being effectively implemented and the original seismic design bases were appropriately factored into the design, and the original seismic design basis has been adequately maintained. For additional discussion on maintenance of the seismic design basis see Cooling Tower Assessment. As noted in that assessment, ENVY has utilized finite element analysis computer programs to perform seismic analysis.

Therefore, based on this review, the current design of the RHR System is in keeping with its original seismic design basis, including OBE and SSE.

2.3.12 Criterion 12 – Training

Assessment Response

The training organization has a process for evaluating all engineering changes to determine whether training is required. This evaluation was performed by conducting interviews with training personnel and by reviewing training processes, governing procedures and training materials. Training is notified of all engineering changes which are entered on the Modification Training Matrix.

The matrix is routinely evaluated to determine if the change requires the training materials for any training program to be modified. If the evaluation determines training may be required a Training Evaluation/Action Request (TEAR) is initiated. The TEAR assigns actions to training personnel to evaluate specific training materials to determine whether the materials require revision. Actions are also assigned to develop training materials and to conduct training as required. The engineering change is also reviewed to determine if any changes to the simulator are required. If a simulator change is required a Discrepancy Report (DR) is generated to implement required changes. All actions are tracked to completion.

Training also has a process to evaluate whether site and industry operating experience (OE) is to be incorporated into related instructor guides (IG) for classroom and simulator training. Interviews with the Operations Training Superintendent revealed an expectation that operating experience is to be included in the development and revision of training material. There is a general understanding of the Entergy Operating Experience procedure (EN-OE-100) including interface with the Site OE Coordinator. There is also a proceduralized requirement for use in the development for training material. Systematic Approach to Training Process procedure (EN-TQ-201) reinforces the need to consider using Operating Experience to re-enforce learning objectives.

Modification ER 05-0534 (Residual Heat Removal Service Water Pump flow restrictors) was selected based on review of the modification. Review of the modification found that training would be required. This modification provided two flow restricting orifice plates to each pump motor bearing cooler Service Water Supply piping. The Operations Training Department reviewed this modification and initiated Training Evaluation Action Request (TEAR) VTY 2006-69. This initiated a NEEDs analysis which identifies if training is required and if so, what Training Department Information Guides (IG) and/or changes to the Simulator would be required. There are then actions assigned to address the changes. The following Instructor Guides were verified complete and reflect the changes made under modification ER 05-0534: Licensed Operator Re-qualification (LOR) LOR-25-305 and Auxiliary Operator RE-qualification (AOR) AOR-25-305 for modification training. Licensed Operator Training (LOT) LOT-00-205 Residual Heat Removal System and LOT-00-276 Service Water System were reviewed by Training and no changes were required. There were no changes required for the Simulator. Modification training was verified through Training Department tracking documentation to have been completed during the March-April 2006 time frame. Operating Experience review for the modification training was reviewed by the Training Department and no Operating Experience documentation was found to be relevant to this modification. Operating Experience references were validated to have been referenced in LOT-00-205 Residual Heat Removal System and included the following Operating Experience documents; Significant Operations Event Records (SOER) 82-2, 85-4, 87-2 and 88-3, and Significant Event Notification (SEN) 224. Operating procedures OP 2181 Service Water/Alternate Cooling Operation Procedure and OP 2124 Residual Heat Removal System Procedure were reviewed and the appropriate modification changes were verified to have been incorporated.

Training performance in the review and implementation of modification ER 05-0534 (Residual Heat Removal Service Water Pump flow restrictors) meets expectations.

As a result of review of the process and interviews with Training Department management, along with review of the actions taken in response to this particular modification, it is concluded the Training Department meets industry standards with respect to evaluating modifications and taking the appropriate actions to incorporate the required changes into training material.

2.3.13 Criterion 13 – Corrective Action Process

Assessment Response

An overall assessment of the corrective action program and its effectiveness was completed and is documented in section 1.2.5 that addresses the Criterion 13 questions for the overall Corrective Action Program.

A list of CRs back to the year 2000 was reviewed. Certain CRs from this list are discussed in detail in the sections above.

Based on these reviews it was determined that issues related to the RHR system have been identified and entered into the Corrective Action Process. Corrective actions were assigned to appropriately address these issues.

The RHR System CR review supports the conclusion of Section 1.2.

RHR System Conclusions

Based on the document reviews and interviews noted above the following is a summary of RHR System assessment conclusions. More detail is provided at the end of each individual section above.

The ENVY engineering design process is well documented, controlled and consistent with industry practice. The current design of the RHR System is consistent with its original design basis and is adequately reflected in plant records and procedures.

ENVY has formal processes for identifying and correcting unanticipated operations outcomes. The corrective action system tracks the implementation of corrective and compensatory measures for the RHR System when issues arise, including the revision of appropriate documents. These processes are consistent with industry practice.

ENVY has formal processes to ensure that when engineering changes are implemented for the RHR System that reviews are conducted and required changes are made to operating procedures, training materials and the simulator to accurately reflect plant conditions. These processes are consistent with industry practices.

The data reviewed and interviews conducted indicate that the RHR system is tested, inspected, maintained, and repaired to industry standards such as AP-913 Equipment Reliability Process. Component classification was performed, PM Basis documents created, and testing and inspections were scheduled and carried out.

RHR system issues, as described in individual sections above, are being entered into the Corrective Action Process. Corrective actions were appropriately assigned to address issues.

Based on the data review and interviews described above, the RHR system poses no significant risk to the ability of the plant to maintain reliable operation in the future.

Barriers to continuing this positive performance, relating to Equipment Reliability, include the inconsistent use of procedures, which can be attributed to a knowledge based versus process based approach. Examples of this are the content of system notebooks, the use of system walk-down checklists, and the use of EPRI Field Guide are not consistent between System Engineers. During future turnover of System Engineers in particular from experienced to less experienced engineers, the expectations for procedure use is essential to ensure continued performance.

References

Procedures and Specifications:

1. EN-DC-115, Rev. 5, Engineering Change Development
2. EN-DC-118, Rev. 2, Engineering Change Closure
3. EN-DC-126, Rev. 1, Engineering Calculation Process
4. EN-DC-141, Rev. 5, Design Inputs
5. EN-DC-195, Rev. 2, Margin Management

6. EN-DC-313, Rev. 2, Procurement Engineering Process
7. EN-LI-113, Rev. 3, Licensing Basis Document Change Process
8. EN-DC-105, Rev. 2, Configuration Management Program
9. EN-OP-104, Operability Determination
10. EN-LI-102, Corrective Action Process
11. EN-MS-S-001, System Notebooks
12. EN-DC-143, System Health
13. EN-DC-178, System Walk Downs
14. EN-PL-170, Nuclear Asset Management Planning
15. EN-DC-320, Identification and Processing of Obsolete Items
16. EN-DC-100, Operating Experience Program
17. AP 6008, Rev. 3, Vermont Yankee Design Change
18. AP 0020, Rev. 78, Control of Temporary and Minor Modifications

Documents

1. Vermont Yankee UFSAR, Rev. 20
2. License Renewal Application, dated January 25, 2006
3. Power Uprate Licensing Application dated, September 10, 2003
4. NRC Safety Evaluation for License Renewal, March, 2007
5. Modification Package for VYDC 2003-016, Alternate Source Term
6. DBD for Residual Heat Removal System, Rev. 22, July 10, 2007
7. Topical DBD for External Events, Rev. 2, August 15, 2008
8. Topical DBD for Safety Analysis, Rev. 4, September 29, 2005
9. DBD for Accident - Event Combinations Topical, Rev. 6, June 29, 2007
10. Task Report T0310, Residual Heat Removal
11. Vermont Yankee Response to Request for Information Pursuant to 10 CFR50.54(f) Regarding Adequacy and Availability of Design Bases Information, BVY 97-23, dated February 14, 1997.
12. Validation Report Design Basis Document RHR, dated May 19, 1998
13. (LOT) 01-300 Pre-outage Shutdown and Startup Overview Lesson Plan
14. Preventive Maintenance Basis Documents; ME030 and M030
15. INPO AP-913, Equipment Reliability
16. NRC Generic Letter GL-2008-01, Air Intrusion

Drawings

1. Flow Diagram residual Heat removal system, Rev. 65, November 10, 2005

NRC Inspection Reports

1. NRC Inspection Report 05000271/2004008, dated December 2, 2004
2. NRC Component Design Basis Inspection Report 05000271/2008008, dated September 26, 2008
3. NRC Component Design Basis Inspection Report 05000271/2006007, dated September 29, 2006

Condition Reports

1. CR-VTY-2007-00556
2. CR-VTY-2007-03425
3. CR-VTY-2007-00132
4. CR-VTY-2005-03586
5. CR-VTY-2008-03425
6. CR-VTY-2008-03163
7. CR-VTY-2004-03600
8. CR-VTY-2006-01427

2.4 Condensate and Reactor Feedwater System

System Description

The following is a summary level description of the Condensate and Reactor Feedwater System. The detailed description of the system is provided in Section 11.8 of the VYNPS UFSAR and the Vermont Yankee Condensate and Reactor Feedwater System DBD.

The Condensate and Reactor Feedwater system takes suction from the hot-wells of the main condensers and delivers demineralized water to the reactor vessel at an elevated pressure and temperature. The main condenser is a carbon steel, deaerating, single pass, radial flow surface condenser located directly below the low pressure turbine. A steam cross-over connection between the condensers is provided to minimize differential pressure. The main condenser hot-well is a carbon steel, divided water-box. It is an integral part of the Westinghouse-supplied main condensers and is located directly below the tube bundle. The hot-well is bailed and anti-vortex bars are welded to the condensate outlets to prevent a vortex from forming at the entrance of the outlets.

Three condensate pumps and three feedwater pumps provide the required flow to the reactor vessel at full load. During full power operation, if a condensate pumps trips, the 'B' feedwater pump will trip to ensure sufficient margin to prevent a low feedwater pump suction pressure trip. A signal is also generated to run-back reactor power to avoid a high reactor vessel water level trip. With 2 condensate pumps running the plant can be operated at about 80% capacity.

Before entering the low pressure feedwater heaters the condensate flow is directed to the full flow filter/demineralized subsystem to ensure that the required water quality to the reactor vessel is maintained.

Two parallel strings of heaters, each consisting of three low pressure and two high pressure feedwater heaters, is provided. If a string of heaters is unavailable, the plant can be operated at reduced capacity. The turbine extraction steam entering each heater is condensed, thereby heating the feedwater and the cooled condensate drains to the next lower pressure heater. Condensate drainage from the lowest pressure feedwater heater is returned to the condenser.

The Feedwater Control System maintains a pre-established water level in the reactor vessel during planned operation by regulating the flow of feedwater into the reactor vessel. In addition to supplying condensate to the reactor feedwater system the condensate pumps provide required flows to the following:

- Air Ejector Condensers
- Steam Packing Exhauster
- RHR, Core Spray, HPCI and RCIC pressurizing (keep fill) piping
- CRD Pumps
- Advanced Off-Gas System Recombiner Condenser
- Low Pressure Turbine Exhaust Hood Spray

- Condenser Bypass Quench Spray
- Seal Water to Heater Drain Valve packing

2.4.1 Criterion 1 - Initial Conditions

Assessment Response

The ENVY design basis for this system is documented in its UFSAR Section 11.8 and in more detail in *Vermont Yankee Nuclear Power Station Design Basis Document for Condensate and Reactor Feedwater System*, Rev. 16, dated July 24, 2007. The codes and standards utilized in the design of the system are as follows:

- ANSI (USAS) B3 1.1.0, 1967, Power Piping Code
- ASME Boiler and Pressure Vessel Code
- National Electrical Manufacturers Association (NEMA), MG- 1
- Heat Exchange Institute, Standards for Steam Surface Condensers
- Heat Exchange Institute, Standards for Closed Feedwater Heaters, Second Edition, 1974

The original design heat load for the main condenser during normal operation was 3.605 x 10^9 Btu/hr. The heat load at 1912 Mwt operation is 4.444 x 10^9 Btu/hr. The condensers are rated to accept 8.846 x 10^9 Btu/hr during emergency steam dump operation without exceeding the main turbine low vacuum trip point or 175°F exhaust hood temperature.

The detailed design basis for the Condensate and Reactor Feedwater system, including the main condenser, is described in the Design Basis Document and includes the system functions, regulatory requirements, system design requirements, operating requirements, and electrical and instrument and control requirements. The verification of the original issue of the Design Basis Document also included an as-built walk down of the system to confirm compliance of the physical configuration with the design basis documents. Details of this validation are documented in *Validation Report, Design Basis Document CFW-1 (Condensate and Reactor Feedwater System)* dated October 21, 1998. Subsequent system modifications were incorporated in the system Design Basis Document in accordance with the requirements of ENVY design change and configuration management procedures.

The most important design basis function of the system from a generation perspective is having the capability to provide sufficient water at an elevated pressure and temperature to maintain a level in the reactor vessel within a predetermined range during all modes of station operation. In the original license of Vermont Yankee Nuclear Power Station, the 100% power level was 1593 Mwt and system flow requirements were 3 condensate pumps at a total flow of 8,048,160 lb/hr and 2 reactor feedwater pumps at a total flow of 7,000,000 lb/hr. At the currently licensed power level of 1912 Mwt the corresponding flow requirements are three condensate pumps at 7,907,000 lb/hr and three reactor feedwater pumps at 7,878,600 lb/hr.

The component design bases are typically performance requirements such as flow rate and developed pressure for pumps and flow rates, pressures and heat transfer requirements for heat exchangers. They too are detailed in the Design Basis Document.

The system and component requirements to support up-rated operation have been evaluated and, where required, the system and/or components have been modified to provide the required capability. The evaluation of the Condensate and Feedwater System is documented in VY-RPT-05-00047, *WBS 1.4.1.3 Condensate and Feedwater EPU Task Report for ER 04-1409* and the modifications are listed in the Design Basis Document. Thus the design basis of the system has been upgraded for the EPU conditions.

Vermont Yankee was issued an Operating License in March 1972 following issuance of a Construction Permit in December 1967. During the licensing process the Vermont Yankee unit was evaluated against the 70 proposed General Design Criteria. Appendix F of the UFSAR contains the original evaluation of the design bases of the facility relative to each of the nine groups of the 70 proposed General Design Criteria. Five of these criteria apply specifically to the Condensate and Reactor Feedwater System and are discussed in the Condensate and Reactor Feedwater System Design Basis Document. These 5 proposed criteria cover Performance Standards, Instrumentation and Control, Reactor Coolant Pressure Boundary outside Containment, Containment Isolation Valves, and Provisions for Testing of Isolation Valves. The Condensate and Reactor Feedwater System complies with all of these criteria. As other NRC regulations and regulatory requirements and requests evolved the Vermont Yankee evaluated its compliance with these regulations or requirements to assure the NRC of its compliance. Regulations that were applicable to the Condensate and Reactor Feedwater System ranged from several NRC regulations in 10 CFR 50, to several NRC Regulatory Guides, and several NRC Generic Letters and I.E. Bulletins. All of these are also discussed in the Condensate and Reactor Feedwater System Design Basis Document without any exemptions or deviations noted.

Based on these document reviews and interviews it was concluded that the design of the Condensate and Reactor Feedwater System, including the Main Condenser, is in keeping with its initial conditions and its current design basis. There should be no negative effects on the future reliability of the Condensate and Reactor Feedwater System based on the design of modifications.

2.4.2 Criterion 2- Procurement

Assessment Response

The ENVY processes governing Engineering Changes require that applicable calculations be performed to support the engineering change and that they be verified as-built prior to return to service of the implemented engineering change. The procedure for performance of calculations prescribes the requirements and the format for the preparation of the calculations. The return to service form of the Engineering Change Closure Procedure requires a sign-off that calculation changes be verified as-built.

An inspection of the High Pressure Feedwater Heater Replacement project which involved the replacement of 4 heat exchangers in the Condensate and Feedwater System, found that the supporting calculations were performed and verified as-built.

There were no procurement changes requiring review calculations associated with the Main Condenser.

Based on the document reviews of the High Pressure Feedwater Heater Replacement project, it was concluded that review calculations were completed for the procurement change to provide new High Pressure Feedwater Heaters with the required capacity to support the EPU.

2.4.3 Criterion 3 – Installation

Assessment Response

In February 1997, Vermont Yankee responded under oath to the USNRC 10 CFR 50.54(f) letter to Licensees requesting information regarding adequacy and availability of design bases. In its response Vermont Yankee stated it has reasonable assurance that the Vermont Yankee design bases have been adequately translated to the plant design and procedures and that plant configuration is being maintained in an appropriate manner. Vermont Yankee went on to state that it was committing to provide improved configuration management including completion of a Design Basis Documentation program, improved Technical Specifications program and a FSAR Verification program. The subsequent development of the DBDs, which took place in 1997 and 1998, included a DBD validation process in which validation teams verified both the design basis information and its application to design documents, and its application to operations, maintenance, surveillance, testing functions and physical configuration. The physical configuration validation included as-built walk-downs. Walk-downs were performed to verify that the system configuration reflects the design basis, is in agreement with plant drawings, and the component labeling is adequate.

Since the completion of the DBD, the engineering procedures for modification development and their closure included and emphasized the requirements of as-built updating of pertinent engineering, operations, training, maintenance, program, and licensing documentation. The closure procedure specifies which updates are required prior to return-to-service and which are required to be tracked and completed after return-to-service. The closure procedure includes a check list with departmental responsibilities for updating the required documentation.

Interviews with Engineering Managers and Engineers demonstrated that they were knowledgeable in these procedural requirements and that procedure compliance was good.

Inspection of the UFSAR, DBD, and flow diagrams (P&ID) of the Condensate and Feedwater system found that the system modifications required for EPU were properly reflected in these documents. Similarly, inspection of the changes to the UFSAR, affected sheets of the Control Wiring Diagram and operating procedures found that changes resulting from implementation of the Reactor Recirculation System Runback and the Reactor Feed Pump Suction Pressure Trip Changes were also properly reflected in these documents.

Based on the document reviews and interviews it was concluded that the ENVY configuration management processes meet industry expectations and that the design of the Condensate and Reactor Feedwater System, including the Main Condenser, is properly documented in the plant records and procedures and the records properly reflect the as-built condition of the plant.

2.4.4 Criterion 4 - Operation

Assessment Response

Unanticipated operations outcomes for this assessment considered plant operating or equipment conditions that did not result in the desired or expected outcome. Interviews with operations management were conducted in order to review the processes used to identify and correct each unexpected operational issue. How Operations implements the governing procedures and processes was discussed with operations management. An example was selected of an event within the condensate/feed water system and reviewed to ensure appropriate compensatory actions were taken, and that plant procedure deficiencies were addressed as appropriate. Additionally it was evaluated to determine if the appropriate level of analysis was performed in accordance with the corrective action system.

As part of the CR evaluation it is determined whether an Operability Evaluation (in accordance with EN-OP-104, Operability Determination procedure) is required for degraded safety-related equipment. The CR evaluation also determines whether an Operational Decision Making Instruction is required for degraded equipment.

An Operability Evaluation is a formal process that is performed in 2 phases. The first step is performed by operations as an interim evaluation to determine if there is reasonable assurance that a degraded safety-related system or component can perform its intended safety function. The second step is a more comprehensive evaluation by engineering that justifies operability for the system or component. In some cases the Operability Evaluation specifies compensatory actions that must be completed to ensure operability.

An Operational Decision Making Instruction is a formal process that justifies continued operation of a degraded system or component. It is developed by operations and/or engineering and reviewed by CRG. In some cases the Operational Decision Making Instruction specifies compensatory actions that must be completed to continue to operate the system.

The compensatory actions that must be completed by Operations on an interim basis can be included in operator equipment inspection instructions, operator logs, standing orders, turnover sheets, and operating procedures. These compensatory actions are discussed by the operating crew at the beginning of each shift.

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Based on the process review, interviews with Operations management and the actions taken to address this specific plant issue, it is concluded that the ENVY response to unanticipated plant operational outcomes meets expectations consistent with industry good practices with the minor noted exceptions:

1. While the ODMI review found the instructions to be adequate, there was a 10 day delay from the date of discovery until the ODMI was initiated. While there was no adverse impact to operations, the initial classification of the CR as level D caused an unnecessary delay in providing clear expectations to the operating crews.
2. Although the condition was tracked in the rounds per the ODMI, there was no Control Room log entry made until September 6, 2008. The threshold for logging of adverse conditions to plant equipment needs improvement.

2.4.5 Criterion 5 - Testing

Assessment Response

Multiple interviews with the Condensate and Reactor Feedwater System Engineer and Supervisor were conducted to assess the operating performance of the systems over the past 3 years with a focus on activities performed during the recent fall 2008 Refuel Outage (RFO-27).

As stated in the system description section, the Condensate and Reactor Feedwater systems take suction from the hot-wells of the main condensers and deliver demineralized water to the reactor vessel at an elevated pressure and temperature. The Condensate and Reactor Feedwater systems are required for plant operation.

START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION However, it has been concluded that there are ongoing chronic issues associated with the Condensate System and in particular the main condenser. This issue has been well documented over the past few years and resolution of issues associated with the Condensate system are based on the License renewal decision.

START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION

The Feedwater System has operated effectively over the past year and a review of testing requirements in accordance with the Feedwater Performance Monitoring Plan did not indicate any significant anomalies.

2.4.6 Criterion 6 – Inspection

Assessment Response

A review of the Condensate and Reactor Feedwater System Monitoring Plan (EN-DC-159) consisting of predictive maintenance activities, quarterly surveillance testing, valve stroke times, pressure, temperatures with a comprehensive listing of tasks were consistent with industry standards.

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END CONFIDENTIAL INFORMATION

Review of the Condensate/Feedwater System inspection documentation, PM Basis and results, revealed that ENVY was scheduling and performing inspections as planned. These inspections are in line with industry standards, which is indicated by the discussions above relating to PM Basis documents, preventive maintenance tasks, system and component health reporting, etc.

2.4.7 Criterion 7 – Maintenance

Assessment Response

START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION

Based on discussions with the System Engineer a review of EN-DC-153 (Preventive Maintenance Component Classification) was conducted to determine the basis for how the Condensate/Feedwater System is maintained. It was concluded that the procedural processes utilized by ENVY are consistent with current industry standards and practices**.** In addition, a review of a specific PM basis document was conducted for Condensate Pumps and it was concluded that the overall approach and requirements were consistent with industry standards. This document establishes functional importance determinations that include Risk and Safety Considerations consisting of questions such as: Safety Related, Safe Shutdown (App. R) PRA/IPE, Required by Tech Specifications, EEQ Program and

Emergency Operating procedures. In addition, the PM 190 was reviewed and concluded that there were no PM deferrals on Condensate/Feed water components processed during the RFO 27

From an overall operational and reliability perspective maintaining feedwater chemistry within industry accepted limits is important to the long term reliability of a nuclear power plant's systems and components. Feedwater chemistry was identified in the June 24, 2008 Management Review Meeting as an issue in the Senior Leadership Concerns section of the report. Feedwater chemistry has been a long-standing concern at ENVY. The WANO chemistry index is an industry standard used to monitor the effectiveness of overall chemistry control based on the concentration of impurities and corrosion products. A lower index number indicates better water chemistry. The WANO index awards maximum credit for an index of 10 or lower and awards no points for an index of 35 or higher. ENVY's index for May 2008 was approximately START CONFIDENTIAL INFORMATION 7

END CONFIDENTIAL INFORMATION A long-term strategy needs to be developed and implemented to improve overall chemistry performance.

2.4.8 Criterion 8 – Repairs

Assessment Response

Overall, it generally appeared that the appropriate scheduled Condensate/Reactor feedwater repairs were planned and performed within reason. This observation of the Condensate/Reactor FW system is based on not identifying failures that caused plant shutdown. The process includes the Design Engineering organization which is responsible for developing the Engineering Changes and it was observed to be appropriate. The question of "timeliness" with respect to responsiveness and implementation is an area that station management must be more cognizant of going forward. There were 3 examples.

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2.4.9 Criterion 9 – Modifications

Assessment Response

Modifications are controlled by procedure EN-DC-115, Engineering Change Development. This procedure mandates reviews to ensure compliance with regulatory, code, and industry requirements. It also requires initiation and processing of updates to plant configuration documents and plant design, licensing and operating margins. Procedure EN-DC-118, Engineering Change Closure, requires that all plant design, licensing, operation, training and maintenance documents be updated prior to modification closure.

START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION

The Reactor Feed Pump suction pressure trip changes and the Reactor Recirculation System Runback were also implemented to support the EPU and to minimize the potential for reactor scrams resulting from Condensate and Feedwater system pump trips and associated transients. The system functions remain the same as the original design basis. All documents including UFSAR, DBD, drawings and procedures were updated as required to reflect the implemented modifications.

Based on the review of the modifications described above, it was concluded that the Condensate and Reactor Feedwater System complies with the original design basis functions and its current design basis parameters. The plant procedures have been updated to reflect the modifications. The ENVY modification processes are consistent with industry expectations. There should be no negative effects on the future reliability of the Condensate and Reactor Feedwater System based on design of modifications

2.4.10 Criterion 10 – Redesign

Assessment Response

Modifications are controlled by procedure EN-DC-115, Engineering Change Development. That procedure mandates reviews to ensure compliance with regulatory, code, and industry requirements. It also requires initiation and processing of updates to plant configuration documents and plant design, licensing and operating margins. EN-DC-195, Margin Management procedure requires that maintenance of design and operating margins be considered in development of engineering changes.

As part of the EPU Program, ER No. 04-1409 provided a detailed review of the effects of EPU on design and operating margins. The review looked at margin changes at both the component and system level. The review analyzed the EPU impact on design and operating margins to ensure there were no unacceptable reductions. There were no unacceptable margin changes associated with the Condensate and Reactor Feedwater System.

START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION

Based on the document reviews described above, there were no margin reductions regarding the safety functions associated with the Condensate and Reactor Feedwater System. There were no unacceptable margin reductions, although the partial Reactor Feedwater Pump redundancy was eliminated. ENVY processes and procedures pertaining to redesign are consistent with industry standards.

2.4.11 Criterion 11 - Seismic Analysis

Assessment Response

The scope of the seismic analysis investigation and assessment includes the following:

1. Review of ENVY's seismic analysis program to determine their approach to seismic design requirements
2. Review of the modification process to determine how seismic design considerations are addressed
3. Review of selected modification packages to determine if appropriate consideration was given to seismic analysis requirements

For selected modifications, NSA will verify that seismic analysis screening criteria were properly applied and that seismic calculations were performed as required. For the selected systems, NSA will review the DBDs and UFSAR to identify their seismic design basis and verify its maintenance in the modification reviews noted above.

The seismic design basis for the plant is documented in Topical Design Basis Document for External Events. The seismic design criteria are based on ground horizontal acceleration of 0.07g for an Operating Basis Earthquake (OBE) and 0.14g for a Safe Shutdown Earthquake (SSE). The vertical acceleration assumed is equal to 2/3 of the horizontal ground acceleration.

Reactor feedwater piping inside containment and the piping outside containment up to the containment isolation valves, is designed to seismic criteria consistent with its safety function, i.e., it has been designed to withstand a Safe Shutdown Earthquake. The remainder of the Condensate and Reactor Feedwater System is not safety related and is designed to withstand an Operating Basis Earthquake.

UFSAR Section A.9 provides a description, scope, and design methodology used for the reanalysis of seismic class I piping subsequent to initial operation. Class I portions of Feedwater piping were reanalyzed using computer dynamic analyses to evaluate seismic loadings. Ground spectra based Regulatory Guide 1.60 and floor spectra with ASME code case N-411 damping defined the seismic loading for this piping. Seismic analyses were performed for the Safe Shutdown Earthquake scenario and the piping was evaluated to ANSI B31.1 - 1977 code allowables.

The Condensate and Reactor Feedwater System modifications reviewed for this assessment did not involve class I seismic systems and, therefore, there were no Class I seismic analyses required or performed.

Based on this review the current design of the Condensate and Reactor Feedwater System, including the Main Condensers, is in keeping with its original seismic design basis, including Operating Basis Earthquake and Safe Shutdown Earthquake.

2.4.12 Criteria 12 – Training

Assessment Response

The training organization has a process for evaluating all engineering changes to determine whether training is required. This evaluation was performed by conducting interviews with training personnel and by reviewing training processes, governing procedures and training materials. Training is notified of all engineering changes which are entered on the Modification Training Matrix.

The matrix is routinely evaluated to determine if the change requires the training materials for any training program to be modified. If the evaluation determines training may be required a Training Evaluation/Action Request (TEAR) is initiated. This is the process in effect at this time. Prior to the TEAR process a Training Change Request (TCR) was utilized. The TEAR/TCR assigns actions to training personnel to evaluate specific training materials to determine whether the materials require revision. Actions are also assigned to develop training materials and to conduct training as required. The engineering change is also reviewed to determine if any changes to the simulator are required. If a simulator change is required a Discrepancy Report (DR) is generated to implement required changes. All actions are tracked to completion.

Training also has a process to evaluate whether site and industry Operating Experience (OE) is to be incorporated into related Instructor Guides (IG) for classroom and simulator training. Interviews with the Operations Training Superintendent revealed an expectation that operating experience is to be included in the development and revision of training material.

There is a general understanding of the Entergy Operating Experience procedure (EN-OE-100) including interface with the Site OE Coordinator. There is also a proceduralized requirement for use in the development for training material. Systematic Approach to Training Process procedure (EN-TQ-201) reinforces the need to consider using Operating Experience to re-enforce learning objectives.

Modification MM 03-016 (Reactor Recirculation System Runback for Feedwater and Condensate System Transients) was selected based on review of the modification, which involved the loss of a Condensate and/or Feed pump. Review of the modification found that training would be required. This modification adjusts the scoop tube positioner and provides a Recirculation System Runback if any condensate pump or Reactor Feed Pump is stopped at high power.

Operations Training Department initiated Training Change Request (TCR) 03-0376 to perform a NEEDs (screening) analysis to evaluate the need to make changes to the Simulator and/or the Operator training material. Review of the TCR actions taken found that the resultant evaluation of the modification led to changes to Training Department Information Guides (IG) and changes to the simulator that properly reflected the modification.

Instructor Guides; Licensed Operator Re-qualification (LOR) LOR-23-905-2 Plant Modifications, Licensed Operator Training LOT-00-202, were reviewed and the appropriate changes were verified to have been completed. Training on LOR-23-905-2 Plant Modifications was verified through Training Department tracking documentation to have been completed in the February through March 2006 timeframe. The simulator fidelity review and incorporation of the appropriate change to the Recirculation function, along with acceptance testing was verified by review of Discrepancy Report 03-0121 as completed on February 6, 2006. Operations procedures, OP 2110 Reactor Recirculation System and OP 2429 Recirculation Flow System Calibration were reviewed and the required changes were verified to have been implemented. These changes were reflected in the Modification Instructor Guide LOR-23-905-2.

Instructional Guide LOT-00-202 Reactor Recirculation was reviewed for the use of Operating Experience. The Reference Section included 6 Service Industry Letters, 2 Significant Event Reports and one Condition Report had been referenced.

Operations Training performance in the review and implementation of modification MM 03-016 RECIRC PUMP RUNBACK meets expectations.

As a result of review of the process and interviews with Training Department management, along with review of the actions taken in response to this particular modification, it is concluded the Training Department meets industry standards with respect to evaluating modifications and taking the appropriate actions to incorporate the required changes into training material.

2.4.13 Criteria 13 - Corrective Action Program

Assessment Response

An overall assessment of the corrective action program and its effectiveness was completed and is documented in section 1.2.5 that addresses the Criterion 13 (Corrective Action Program) questions for the overall Corrective Action Program.

A list of CRs back to the year 2000 was reviewed. Certain CRs from this list are discussed in detail in the sections above.

Based on these reviews it was determined that issues related to the Condensate/Feedwater System are being identified and entered into the Corrective Action Process. Corrective Actions were assigned to appropriately address these issues.

The Condensate/Feedwater System CR review supports the conclusion of Section 1.2.

Condensate/Reactor Feedwater System Conclusion

The ENVY engineering design process is well documented, controlled and consistent with industry practice. The current design of the Condensate/Feedwater System is consistent with its original design basis and is adequately reflected in plant records and procedures.

ENVY has formal processes for identifying and correcting unanticipated operations outcomes. The corrective action system tracks the implementation of corrective and compensatory measures for the Condensate/Feedwater System when issues arise, including the revision of appropriate documents. These processes are consistent with industry practice.

ENVY has formal processes to ensure that when engineering changes are implemented for the Condensate/Feedwater System that reviews are conducted and changes made as required to operating procedures, training materials and the simulator to accurately reflect plant conditions. These processes are consistent with industry practices.

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END CONFIDENTIAL INFORMATION The NSA team believes that this is a challenge to both near term (down-powers to manage tube leaks) and long-term (potential adverse impact on fuel cladding) reliability.

The data reviewed and interviews indicate that the Condensate/Feedwater System is tested and inspected to industry standards such as AP-913 Equipment Reliability Process. Component classification was performed; PM Basis documents created, and testing and inspections were scheduled and carried out. CRs were created to resolve issues found during testing and inspections.

With respect to the focus on effective equipment reliability, the process and procedures that support the Condensate/Feedwater Systems are consistent with current commercial nuclear power's standards for reliable operation. However, the effective application of these processes is in question based on:

- The need to reach a conclusion regarding the License Renewal Program
- The need to shift from 'informal compliance' to 'formal compliance' associated with approved process and procedures. This issue is discussed in detail in the management overview section
- The need to effectively train and develop the new Condensate and Reactor Feedwater System Engineer

Condensate/Feedwater System issues, as described in the individual sections above, are being entered into the Corrective Action Process. Corrective actions were appropriately assigned to address these issues.

References

1. VYNPS, Condensate and Reactor Feedwater System DBD
2. VYNPS, External Events Topical DBD
3. Validation Report, Design Basis Document CFW-1 (Condensate and Reactor Feedwater System) October 21, 1998
4. VYNPS UFSAR Section 11.8, Condensate and Reactor Feedwater Systems
5. VYNPS UFSAR Section A.9, Description, Scope, and Design Methodology Used for the Reanalysis of Seismic Class I Piping subsequent to Initial Operation
6. VY-RPT-05-00047, WBS 1.4.1.3 Condensate and Feedwater EPU Task Report for ER 04-1409
7. VY-RPT-05-00049, WBS 1.4.1.5 Circulating Water/Condenser/Cooling Tower EPU Task Report for ER 04-1409
8. VYDC-2003-02, #1A, #1B, #2A and #2B Feedwater Heater Replacement Project
9. MM 2003-015, Reactor Feed Pump Suction Pressure Trip Changes for EPU
10. MM 2003-016, Reactor Recirculation System Runback for Feedwater and Condensate System Transients
11. ER 06-1099, Reactor Recirculation Runback Termination Point Change
12. VYNPS, Extended Power Up-rate Application
13. NRC SER, License Renewal of VYNPS
14. NRC Inspection Report 05000271/2004008
15. NRC License Renewal Inspection Report 05000271/2007006
16. NRC Component Design Basis Inspection Report 05000271/2008008
17. EN-DC-105 Rev. 2, Configuration Management Program

18. EN-DC-115 Rev. 5, Engineering Change Development
19. EN-DC-118 Rev. 2, Engineering Change Closure
20. EN-DC-126 Rev. 1, Engineering Calculation Process
21. EN-DC-141 Rev. 5, Design Inputs
22. EN-DC-195 Rev. 2, Margin Management
23. EN-DC-313 Rev. 2, Procurement Engineering Process
24. EN-LI-113 Rev. 3, Licensing Basis Document Change Process
25. AP 6008 Rev. 3, Vermont Yankee Design Change
26. AP 0020 Rev. 78, Control of Temporary and Minor Modifications
27. G-191157 SH. 1, 2, and3, Flow Diagram, Condensate, Feedwater and Air Evacuation Systems

2.5 Cooling Tower (CT) Structure (Including part of the Circulating Water system)

System Description

ENVY Cooling Towers (CT) provide cooling for the Circulating Water System (CW) and the Alternate Cooling System (ACS). Cooling tower construction is an Ecodyne Model 70. There are 2 towers; the east tower is CT 1 and west tower CT 2. Each tower has 11 cells, which are divided on north, and south ends by partition walls.

The cooling towers are braced wood frame structures constructed of treated Douglas fir with bolted and steel bracket type connections. More recently, fiberglass reinforced plastic has been used to replace some of the wooden members of each cell except for cells CT 2-1 and CT 2-2, the safety and seismic cells. The towers are modular construction with the 11 cells in the longitudinal direction. The towers are constructed of timber columns, beams, girders and diagonal bracing. The wood frames support fans, fan motors, circulating water distribution piping and other components. The original construction of the towers was of treated Douglas fir with plastic fill and drift eliminators. Each tower's 11 cells are separated by fireproof partitions. Fire-resistant materials are used throughout. A distance of 300 feet, center-to-center, separates the towers. The fan deck is covered with fire resistant materials. A concrete cooling water basin below grade supports the tower frames.

The cooling towers have a 366,000 gpm and 2,600 times x 10^6 btu/hr original design capacity. Discharge from the towers was originally designed to be 87 degrees F with a 101.2 degree F inlet and 75 degree F ambient wet bulb temperature. The actual cooling tower heat load and temperature parameters vary with operating mode, river conditions, ambient wet bulb, and actual tower performance.

The cooling towers are designed to withstand seismic ground motion of Class II intensity except those parts required for the Alternate Cooling System, which are designed to withstand Class I, seismic intensity loads. The cooling towers are designed to withstand wind loading of 30 pounds per square foot and 40 pounds per square foot snow and ice loadings.

There are 3 modes of operation for the Cooling Towers. In the recirculation mode, circulating water is supplied by the circulating water pumps to the discharge structure where the bypass gates are closed, preventing discharge of circulating water to the river. There are three circulating water booster pumps of 122,000 gpm capacity each, supplying the required head necessary to pump the water to the towers. The cooling towers are capable of 100% heat rejection, i.e., 100% of decay heat removal. The cooled circulating water then flows to the circulating water pumps where operation is the same as for the circulating water system.

In the hybrid mode, circulating water from the condenser is pumped from the discharge structure by the circulating water booster pumps, through the cooling towers, and part of the flow is directed to the river.

The third mode is the discharge mode. The discharge mode is the same as the recirculation mode, except after the cooled water leaves the cooling towers it is discharged to the river. The mode in which the system operates is a function of river temperature. The switching of the different modes is accomplished manually.

The Circulating Water (CW) system provides the power generation heat sink for plant operations. Three CW pumps provide a total of 360,000 GPM of river water to pass through the condenser and cool steam that has exited the main turbine.

A portion of the Cooling Tower System is part of the Alternate Cooling System (ACS). That portion consists of a cooling tower cell (CT 2-1) and the CT 2 deep-water basin. The deep water basin provides flow for ACS even with CT 2 out-of-service. The deep basin remains filled to support ACS when it is required to be operable. Cooling water is supplied by gravity flow from the cooling tower basin to the RHRSW pumps. Water is then pumped to the RHR heat exchangers, RHRSW pump motor bearing oil coolers, emergency diesel generator coolers, ECCS room coolers, SFPCS heat exchangers, RHR pump seal coolers and auxiliary piping, valves and instrumentation necessary to provide flow through the RHR heat exchangers, then returned to the cooling tower where the latent heat is transferred to the atmosphere. This system is used for those events where the Service Water pumps are not available, which occurs if the Intake Structure is submerged during maximum flood conditions or failure of various upstream dams, if the Vernon Dam fails and the river level falls, or if a fire at the intake Structure disables the SW pumps. Cooling tower cell CT 2-2 is also designed to withstand Class I seismic loads so its failure would not impact safety-related cell CT 2-1.

The assessment of the ACS is addressed in the Service Water Section.

2.5.1 Criterion 1 - Initial Conditions

Assessment Response

Cooling Tower cell CT 2-1 is the only cooling tower cell that is safety-related. For that cell, ENVY has verified and documented the original design codes and standards as well as the current design bases for the cooling towers in *Vermont Yankee Nuclear Power Station Design Basis Document (DBD) For Service Water, Residual Heat Removal Service Water, Alternate Cooling System, Rev. 29,* dated August 20, 2007. The DBD verification also included an as-built walk-down of the safety-related portion of the cooling towers to confirm compliance with the design basis. Details can be found in *Validation Report Design Basis Document SW-1, Service Water System*s, dated November 27, 1998.

The design basis for the cooling towers is also documented in the ENVY UFSAR. As part of ENVY's response to NRC letter 10 CFR 50.54(f), ENVY committed to a complete FSAR verification program to ensure consistency between the FSAR and plant documentation.

The codes and standards used in the design of the entire cooling towers are as follows: The basic design of the cooling towers is in accordance with the Cooling Tower Institute *Standard Specification for The Design of Cooling Towers with Douglas Fir Lumber*. It is also designed in accordance with

EBASCO Specification 54-63, *Mechanical Draft Cooling Towers, Rev 1*, dated April 23, 1969. As will be discussed later, cooling towers cells CT 2-1 and CT 2-2 are designed to Seismic Class I standards.

Cooling tower cell CT 2-1 is the only safety-related cell and is, therefore, the only cell subject to direct NRC regulations. As those NRC regulations and regulatory requirements and requests evolved over the years, the ENVY plant evaluated its compliance with these regulations or requirements to assure the NRC of its compliance with its intent. Those that were applicable to the safety-related portion of the cooling tower ranged from NRC regulations in 10 CFR 50, Regulatory Guides, to Generic Letters and I.E. Bulletins. All of these are also noted in the Design Basis Document referenced above without any exemptions or deviations noted. It should also be noted that for the safety-related cooling tower cell, any deviations or exemptions from the original design basis must be identified to and approved by the NRC.

ENVY utilizes a fleet-wide Configuration Management Program, EN-DC-105, Rev. 2. The purpose of the procedure is to ensure consistency between the design requirements, physical configuration, and plant configuration information. This program is also used for primary suppliers of equipment and software for ENVY.

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Based on the document reviews and interviews noted above, the design of the Cooling Tower System is in keeping with the expected initial conditions and its current design basis. There should be no negative effects on the future reliability of the Cooling Tower System based on design modifications.

2.5.2 Criterion 2 – Procurement

Assessment Response

The ENVY processes governing engineering changes, calculations, procurement etc., referenced below, require that applicable calculations be performed to support the engineering changes, that the changes be compared against the original design requirements and design bases, and that they be verified as built prior to return to service of the implemented engineering change. The procedure for performance of calculations, EN-DC-126, prescribes the requirements and the format for the preparation of the calculations. Discussions with plant engineers and managers confirmed their familiarity with these procedure requirements. It was verified that this procedure was utilized for cooling tower procurement changes. Quality Assurance audits are also utilized to verify procedure compliance.

There have been modifications to the Cooling Tower System since the original plant construction. Two modification packages were reviewed as part of this assessment. The modification package to replace the cooling tower fans and motors and the modification package to replace the cooling tower fill and structural members with fiberglass components, EC-4721 were reviewed. For both modifications, appropriate calculations were performed for procurement changes and the changes were compared against the original design.

Based on the document reviews and interviews noted above, for the Cooling Tower System, new sets of review calculations were completed for the respective procurement changes and the procurement changes were compared against the original design and all of its calculations. There should be no negative effects on the future reliability of the Cooling Tower System based on the process for maintaining procurement records.

2.5.3 Criterion 3 – Installation

Assessment Response

The Engineering Change Closure procedure sets forth the requirements of updating pertinent engineering, operations, training, maintenance, program, and licensing documentation. The procedure specifies which updates are required prior to return to service and which are required to be tracked and completed after return to service. The procedure specifically requires that calculation changes be verified as-built prior to returning the system to service. Discussions with plant engineers and managers confirmed their familiarity with these procedure requirements. Quality assurance audits are also utilized to verify procedure compliance. Reviews of plant records indicate these procedures are being implemented and in a timely fashion.

Modification package reviews for the Cooling Tower System also verified that system changes are being properly reflected in these design documents. Two modification packages were reviewed as part of this assessment. The modification package to replace the cooling tower fans and motors, ER-04-0705, and the modification package to replace the cooling tower fill and structural members were reviewed. For both modifications the review confirmed that appropriate updates to plant design documents were performed to reflect the as-built condition of the plant and the changes were made to all affected design documents.

In February 1997, ENVY responded under oath to the USNRC 10 CFR 50.54(f) letter to Licensees requesting information regarding adequacy and availability of design bases. In its response ENVY stated it has reasonable assurance that the ENVY design bases have been adequately translated to the plant design and procedures and that 'plant configuration is maintained in an appropriate manner'. ENVY went on to state that it was committing to provide improved configuration management including completion of a Design Basis Documentation program, improved Technical Specifications program and a FSAR Verification program. The subsequent development of the DBDs, which took place in 1997 and 1998, included a DBD validation process in which validation teams verified both the design basis information and its application to design documents, and its application to operations, maintenance, surveillance, testing functions and physical configuration. The physical configuration validation included as-built walk downs. Walk downs were performed to verify that the system configuration reflects the design basis, is in agreement with plant drawings, and the component labeling is adequate.

The NRC has also performed an inspection of the extended EPU Program and concluded that sufficient design controls are in place and being implemented and, therefore, plant records should adequately represent the as-built condition of the plant. Details can be found in NRC Inspection Report 05000271/2004008, dated December 2, 2004.

Based on the document reviews and interviews noted above, Cooling Tower plant records do adequately represent the as-built condition of the plant. All Cooling Tower System changes are reflected in all documents from the design basis through as-built and through current operation. ENVY meets industry expectations as it pertains to the installation criteria for the Cooling Tower System.

2.5.4 Criterion 4 – Operation

Assessment Response

Unanticipated operations outcomes for this assessment considered plant operating or equipment conditions that did not result in the desired or expected outcome. Interviews with operations management were conducted in order to review the processes used to identify and correct each unexpected operational issue. How Operations implements the governing procedures and processes was discussed with operations management. An example was selected of an event within the Cooling Tower System and reviewed to ensure appropriate compensatory actions were taken, and that plant procedure deficiencies were addressed as appropriate. Additionally it was evaluated to determine if the appropriate level of analysis was performed in accordance with the corrective action system.

As part of the CR evaluation it is determined whether an Operability Evaluation (in accordance with EN-OP-104, Operability Determination procedure) is required for degraded safety-related equipment. The CR evaluation also determines whether an Operational Decision Making Instruction is required for degraded equipment.

An Operational Decision Making Instruction is a formal process that justifies continued operation of a degraded system or component. It is developed by operations and/or engineering and reviewed by CRG. In some cases the Operational Decision Making Instruction specifies compensatory actions that must be completed to continue to operate the system.

Condition Report CR-VTY-2007-03571 *Cooling Tower Leak* was selected as an event driven action taken in response to an unanticipated plant event. As part of a Cooling Tower inspection being performed on September 17, 2007, a leak was reported coming from a sag in the water deck at Bent 3 west side of Cooling Tower 2-3. This had previously been identified as a minor leak. The repairs were made and the condition report was initiated. The CR was classified Level 'C' (condition adverse to quality requiring correction). It was noted by the Condition Reporting Group that CR-VTY-2007-03234, required a root cause analysis; therefore, no apparent cause would be required under this CR. This is according to EN-LI-102 Corrective Action Process. No Operability Determination was required. Corrective Action 2 resulted in creation of Corrective Action type- Operations Decision Making Instruction (ODMI). ODMI is a plant process which involves formulation of a peer group or identification of an individual to evaluate the condition and determine if interim actions would be required pending repairs. The ODMI was initiated to assist Operations personnel in determining conditions which may indicate new problems with regard to the structural integrity of Cooling Tower 2. This ODMI resulted in determination of what is acceptable or excessive leakage conditions and defined trigger points based on several observable indications (increased leakage and/or relevant observations of Cooling Tower 2 conditions) and corresponding action(s) by Operations (including removing the Cooling Tower from service and reducing Reactor Power). The Specific Actions involved operator Training. This was verified to be a comprehensive document that was disseminated as on-the-job training to the shifts. The requirement to perform a monthly walk down was verified to be incorporated into procedure OP 2180 Circulating Water/Cooling Tower Operation. The ODMI was verified to have been incorporated into the ODMI list and the night order was reviewed. Operating Experience was not specifically referenced in the ODMI; however the training material was an example of the utilization of internal Operating Experience.

Based on interviews, review of procedures, and review of the response to unanticipated plant conditions and events for the Cooling Tower it was evaluated that the processes, procedures, and the actions taken meet industry standards.

2.5.5 Criterion 5 – Testing

Assessment Response

A sample of documents was reviewed and interviews were conducted relating to testing associated with Cooling Towers and portions of the Circulating Water System (e.g. CW & Booster Pumps). The testing of Cooling Tower components and items listed on the Circulating Water System Trending and Monitoring Plan was reviewed.

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The first step in determining testing is component classification. Component classification has been performed at ENVY. The basis of the classification of a sample of components (e.g. CW Pump) was reviewed and verified 'live time' from a data base during an interview with the CW System Engineer. The classification process is governed by Entergy procedure EN-DC-153 which references industry standard INPO AP 913.

Examples of testing results were also reviewed "live time" from the work order database. This verified what was expected by review of ENVY Performance Indicators that PMs are being performed within their time frame. Testing results and discussions with System Engineers also showed that when issues are identified they are entered into the Corrective Action Program.

Two System Engineers were interviewed. They were the Circulating Water Engineer who is responsible for Cooling Tower mechanical and electrical components and the Building/Structure System Engineer who is responsible for CT structural components.

The testing discussion primarily involved the CW Engineer since it relates to components. The CW System Engineer has experience as a System Engineer at ENVY on other systems and at other nuclear sites. During interviews with the System Engineer, the System Engineer was familiar with items on his System Monitoring Plan. The System Engineer was able to electronically display the basis for testing in the PM Basis Document as well as other information. This included an explanation of how components are classified on his system. As an example, the System Engineer explained how the CW pumps were classified.

The CW System Engineer exhibited an understanding of testing, and his responsibilities relating to testing on his system which included periodic review of testing results. He did mention that performing these reviews was impacted by the current shortage of System Engineers at ENVY. The trending of results was mentioned as being impacted. Not meeting testing criteria will drive action. However, trending of data can lead to identifying issues earlier. Per discussions at ENVY, a replacement CW Engineer has been selected. The current CW Engineer will be assigned different systems; this should reduce some of the burden on the new CW System Engineer. An overall discussion on the System Engineer organization is included in other sections of this report.

Review of a CR list back to 2000 did not reveal any issues identified relating to testing not being performed as planned.

The review of documentation relating to testing, including results and discussions with the current System Engineers did not reveal any issues that would indicate ENVY was not scheduling and carrying out planned testing on the Cooling Towers and Circulating Water System. The testing is in-line with industry practices and standards. Component classifications, system monitoring reports, predictive maintenance tasks discussed above are the basis for this conclusion.

As was the case with other System Engineers interviewed at ENVY, one item relating to testing that could be improved was the interface between the System Engineer and other areas in Engineering. As an example, the CW System Engineer does not routinely review vibration test results or receive feedback from the Component Engineer on his/her review of test results. The System Engineer should be aware of all test results on his system. This would assist in evaluating the condition of his/her system.

2.5.6 Criterion 6 – Inspection

Assessment Response

Inspections on the Cooling Tower (CT) include those specified in the PM Program, System Walk-Downs, System Health Reporting, and Component Health Reporting. These inspections include physical inspection of certain components and system data review for reports, such as System Health reports.

Additional inspections of the CT are also performed per corrective actions related to CT Condition Reports.

Per PM basis document ME076 for CT 2-1 and ME077 for the remaining cells, numerous inspections are performed on CT fans and motors. These include gearbox, fan blade pitch angles and clearances, and motor inspections. All are performed annually prior to placing towers in operation in the spring.

Per PM basis document M307 the West Cooling Tower Deep Basin, associated with the Alternate Cooling System, is inspected periodically. These inspections include measuring the depth of the silt in the basin. The results of this inspection determine if basin cleaning is required. During basin cleaning the cooling tower column sections below the water line are inspected. The last inspection was in 2007 and no degradation was noted.

Corrective actions from 2007 and 2008 CT event Condition Reports have resulted in weekly inspections of the CT structure being performed. Three of these weekly inspections each month are performed by Engineering. The remaining inspections performed by Operations. Guidance for inspections is included in ENVY procedure OP 52114 Cooling Tower Structural Inspection and repair. This procedure is for inspections when the CTs are out of service, however; the guidance is helpful for online inspections. A draft for on-line inspections has been prepared and is in use. The plan is to have this approved by the end of 2008 and included in a PM task for inspections. The Structural System Engineer also uses EPRI Aging Assessment Guideline 1007933 during weekly inspections. The installation of new cooling tower fill, per CT project plan, provides a better view of the tower structure, which enhances the ability of visual inspections to identify issues.

Results of these weekly inspections are entered into a work order LO-WTENVY-2008-0125. Additionally, data from the inspections is entered into a matrix controlled by the Structural System Engineer and the Design Engineer responsible for the CTs. The work order and matrix are used for trending data to monitor conditions.

There are numerous examples of items being identified during these inspections in the list of CRs relating to the Cooling Towers, which have resulted in repairs being performed. As an example CR-VTY-2008-03198 was created as a result of a weekly inspection by engineering July 31,2008. Other examples (CR-VTY-2008-158 thru 165) resulted from inspections performed during 2008 spring maintenance of non-safety-related CT cells.

Routine operator rounds also are performed on the CT. These rounds have resulted in identifying issues. As an example, an Auxiliary Operator identified some structural degradation to the distribution piping on CT 1-1 (reference CR-VTY-2008-02904).

There is evidence that personnel on site in organizations, other than Engineering and Operations, have sensitivity toward identifying CT issues. CR-VTY-2008-3775 describes a situation where an employee contacted operations after viewing a condition he was not used to seeing. Investigation revealed a minor leak that was addressed.

During the recent fall 2008 outage, the work scope on CT 2-1 was increased due to inspections performed at the beginning of the outage. The inspections found that items for replacement were in a degraded condition (e.g., 2 West ‘B’ columns) and additional columns (e.g. ‘C’ columns on Bent West) not scheduled for replacement were also replaced due to conditions identified in the inspection. ENVY created CR-VTY-2008-4391 to document these and other inspection findings.

Circulating Water Pump inspections per ME080 include pump disassembly and inspections and motor inspections every 6 years. Per ME129 CW Booster pumps and motors are visually inspected each year prior to being put into service. The difference between the types of inspections is noted on the basis document. CW Booster pumps do not operate continuously and are only during summer months.

Entergy Procedure EN-DC-143 governs System Heath reporting. Several areas contribute to an overall System Health rating, which is represented by a color; 'Green' acceptable, 'White' acceptable with additional monitoring, 'Yellow' degraded with action plan, and, 'Red' unacceptable for reliable plant operation.

Cooling Tower System Health is covered by Buildings System Health Reports. Building Systems covers 16 subsystems and is described as a catch all for identifying work not associated with other systems.

Review of Building System Health Reports revealed that the overall rating of Building Systems was 'Red' for 3rd Qtr. 2007. One cause for this rating was CT issues. The Maintenance Rule (MR) monitored performance area was rated red due to CT being MR catgory (a) (1) and recovery plan under development. The 4th Qtr. 2007 report overall color was 'Red', however, MR parameter was rated 'Yellow' since the MR recovery plan had been developed. The 1st Qtr. 2008 Buildings System Health report had an overall color of 'Yellow'. This change was explained as being due to overall improvements in performance areas. Contributing to this, it was noted that the CT MR recovery plan was in progress and working. The overall color returned to 'Red' in the 2nd Qtr. 2008 report. The cause indicated on the report was the Operational Impact performance area being rated 'Red'. Because of other Building Systems sub-systems (e.g., Control room Annunicators) not related to the cooling towers. This was due to 7 open alarm issues, 4 above the acceptance criteria of 3. Alarm issues are covered in another section of this report.

The ENVY Component Engineers are responsible for Component Health Reporting. The ENVY Component Health Program is discussed in another portion of this report. Cooling tower fan motors are not included in the Component Health Program due to their lower horsepower. CW Pump Motors are monitored as part of the Component Health Program.

The Motor Component Health report dated September 2008 was reviewed. There are 10 performance parameters monitored. Nine of 10 for CW pumps A, B, and C were either 'Green' or 'White'. The equipment status parameter on all 3 pumps was rated 'Yellow' which drove the overall color to be 'Yellow;' for these pumps. The reason for the 'Yellow' rating was a small oil leak on all 3 motors. The action until the leaks are repaired is to monitor winding temperature to ensure oil buildup does not occur. CW Booster Pumps A and B are rated overall 'Yellow'. Eight of the 10 performance parameters on both pumps are either 'Green' or 'White' with the remaining 2 are rated 'Yellow'. The reason for the 'Yellow' rating is that the motors amperages are at a level that requires monitoring. The overall ratings of both the CW and CW Booster pumps is appropriate and at a conservative threshold.

The CW and Structure System Engineers are required to perform periodic walk downs of their systems per Entergy Procedure EN-DC-178. These walk downs include monitoring system parameters, looking for material degradation, and reviewing data such as outstanding work orders. The System Engineers were requested to perform a system walk-down, members of the inspection team accompanied the System Engineer on this walk-down. EN-DC-178 includes a checklist for preparing for and conducting a system walk-down. Both System Engineers were familiar with the checklist. However, they did not use or refer to the checklist during the walk-downs. The Structure System Engineer was observed using the EPRI Aging Assessment Field Guide during his walk-down.

During walk-down of the CTs, members of the assessment team noted that the housekeeping on the top deck of the west cooling tower could be improved. Small pieces of material from CT repairs, e.g. nails, were still on the deck. The assessment team did not note any housekeeping issues during the walk-down of the CW Pump house.

Review of a list of CRs since 2000 revealed that CR-VTY-2008-2136 was created in 2008 due to FME concerns on the Cooling Tower. This can be attributed to poor housekeeping. ENVY Quality Assurance initiated this CR.

During interviews, both System Engineers exhibited a good understanding of inspections on their systems. Relating to Cooling Towers both discussed in detail inspection specifics. The Design Engineer responsible for CTs was also interviewed during which his understanding of inspection and repair of CTs was evident. The CW Engineer was familiar with the PM program and how changes to tasks are accomplished. The Structural System Engineer has fewer interfaces with the PM program since it is primarily a component driven process.

Review of Cooling Tower and CW inspection documentation (e.g. PM Basis and results) revealed that ENVY was scheduling and performing inspections as planned. This review, which included Condition Reports, also revealed that ENVY is entering inspection findings into its Corrective Action Program. The PM Bases documents, Preventive Maintenance tasks, System and Component health reports discussed above are in-line with industry practices and standards.

Several areas require focus. The items found during the inspection of CT 2-1 at the beginning of the fall Refuel Outage (reference CR-VTY-2008-4391) are an indication that degraded columns can go undetected since not all areas are fully accessible to visual online inspections. A review of the items found per the referenced CR for potential improvements to the on-line inspection process is recommended. This issue is also discussed in 2.7.8 Repairs.

Both the CW and Structure System Engineers will be changed out shortly. The CW Engineer will be assigned different systems, the Structure Engineer is retiring. The turnover plan for both Engineers needs to have the detail and amount of time necessary to ensure CT inspections are thorough enough to ensure issues are identified early. The fact that the CW Engineer and the Design Engineer for CTs will remain onsite should help since they can mentor the new System Engineers.

Cooling Tower housekeeping needs more focus. The inspection team noted this and Quality Assurance created a CR in 2008 relating to FME concerns. A lower threshold for identifying housekeeping items during weekly Cooling Tower inspections needs to be established.

2.5.7 Criterion 7 – Maintenance

Assessment Response

Several programs, processes, and initiatives contribute to ENVY managing assets to ensure reliability. This includes managing aging components and parts obsolescence. Entergy Management Manual EN PL 170 Nuclear Asset Management Planning governs the process for addressing aging components. Entergy Procedure EN-DC-320 Identification and Processing of Obsolete Items governs the process for parts obsolescence. ENVY per procedure performs quarterly review of the industry obsolete inventory database (OIRD). This database lists obsolete parts and possible replacements. Another industry source for identifying obsolete parts is Operating Experience (OE). Entergy Procedure EN-OE-100 governs the OE Program. OE from INPO, NRC, Owners Groups, and Vendors is evaluated per the actions described in the OE Program procedure.

One example of ENVY addressing obsolete parts is CT brass anchor brackets that are no longer available. ENVY did have some spares but during future CT maintenance will need additional brackets. ENVY has developed a process that refurbishes the replaced brackets for installation during future CT maintenance.

Identifying and obtaining parts for the fall 2008 Refuel Outage CT work was an issue for ENVY. Parts were still not available at the start of work. This impacted site resources. The CT Project Team needed to spend time expediting parts, creating code numbers, etc.

Certain ENVY initiatives have had an added advantage of contributing to addressing aging components. The EPU in 2003 included the replacement of cooling tower motors and fans. The fans and motors were replaced to reduce the height of CT mist plumes. The Cell CT 2-1 fan and motor were not replaced per an order from the Vermont Public Service Board

The replacement of cooling tower columns, louvers, fill, partition walls, etc. are included in the project plan for ENVY Cooling Tower Upgrades EC-4721. This work is included in the site's long-term plan and budget.

Relating to the Circulating Water System, ENVY's long-term plan includes an item to replace CW Pump Motors. One of the replaced motors will be rebuilt and kept as a spare. Another example of managing aging components is the scheduled inspections of CW underground piping performed during the recent fall 2008 Refuel Outage. One issue with a previously repaired section was revealed. A portion of the grout had flaked off. An engineering evaluation determined that repairs could be deferred until the next scheduled Refuel Outage.

ENVY Preventive Maintenance tasks also contribute to managing the aging of components. As an example on the CTs this includes the periodic change out of distribution nozzles and the flexible element between fan motor and fan. These items have been determined to have a service life. The periodic application of fungicide to certain areas of CTs to prevent mold is another example of ENVY taking action to manage aging components. The basis for these tasks is listed in Preventive

Maintenance Basis document M307. On CW the rebuilding of CW pumps every six years per basis in ME080 mentioned in the inspection section is another example of PM tasks contributing to managing aging components.

As mentioned previously, ENVY System Engineers have access to an EPRI Aging Assessment Field Guide 1007933. The Structural System Engineer discussed the value of this Guide during interviews and was observed using the Guide during a walk-down of the CTs. Continued use of this Guide should aide in managing aging of the CT structure and components. The Guide provides information on how to detect and evaluate aging related degradation. Metal, Concrete, Coating, Lubricant, Mechanical and Electrical Component degradation are covered.

Review of CRs since 2000 relating to CT and CW system did not reveal any issues relating to aging components or obsolete parts.

The review described above indicates that ENVY has the programs and processes in place to address the management of aging components. ENVY Cooling Tower Upgrades EC-4721 and CW motor replacements are examples of this. However, during interviews a better understanding of these programs and processes as well as more examples of their implementation would be expected as part an aggressive program. Another section of this report discusses the management of aging components further.

During interviews, System Engineers mentioned that certain plans for aging component replacements/upgrades are dependent upon license renewal. Should license renewal be granted, a thorough review of these plans is recommended. To ensure they are detailed enough since; there may have been a tendency to not spend time on details when completing the work was in question.

2.5.8 Criterion 8 – Repairs

Assessment Response

System and component testing and inspections can positively impact the amount and type of repairs required for system component reliability. Previous sections detailed what testing and inspections are performed on the CT components and structures and CW components. The type of testing and inspections described are in line with industry standards, such as INPO AP 913.

Due to past events, CT repairs are a priority at ENVY. The site budget for the next few years provides funds to replace CT columns, louvers, and other items. There is a Management focus on CT repairs. One example of this was the involvement of ENVY Management when work was performed on CT 2-1 during the Fall 2008 Outage RFO-27. During a daily coordination meeting, observed by an inspection team member, Management was present. During the meeting the Director of Engineering coached and questioned the CT team. This included questions not related to Engineering such as manpower and parts issues. The Manager of Project Management was also present and provided coaching to team members.

Overall ENVY has a low corrective maintenance backlog, which indicates that repairs are completed as soon as possible. The non-outage corrective maintenance backlog indicator provided by ENVY, which is a key industry indicator, has been below the goal of 10 for all the months reported. Being below the goal is a positive indicator for this measurement.

Review of the Non-Outage CM indicator found none of the open corrective maintenance items are relating to CW or CT components. Review of Building System Health reports found open Elective Maintenance work orders, however none related to CW or CT components.

Circulating Water (CW) System performance and no open non-outage CMs on CW are indications that CW system component repairs are sufficiently in depth to support reliable operation. Additionally, review of CW CRs from the years 2000 to 2008 did not reveal any issues relating to reworking items previously repaired.

Results of the review of the information provided to determine if repairs have been sufficiently in-depth for CTs is mixed. The plans to complete repairs are in place. However, the performance when carrying out these repairs could be improved.

There has been a large amount of repairs completed and planned per the ENVY Cooling Tower Upgrade Project. This includes the replacement of wood columns with of Fiberglass Reinforced Plastic (FRP) columns that support the distribution piping in non-safety related cells. Safety related columns are replaced in kind with Douglass Fir. The reasons provided, by ENVY personnel for in-kind replacement in the safety cell CT 2-1, are that this cell has additional bracing that makes it more structurally sound than non-safety related cells and that this cell has been inspected and maintained more aggressively than other cells.

Column replacement for both cooling towers also differs in one other way. Cooling Tower 2-1 columns are fully replaced, above and below water line. While not all Cooling Tower 2-2 columns are replaced below the water line. Discussions with ENVY personnel revealed that the CT 2-2 basin was drained during the 2007 Refuel Outage and column sections below the water line were inspected and found to be sound. Currently the plan is to continue to inspect CT 2-2 column sections below the water line when the basin is drained during future Refuel Outages and take actions per the inspection results.

Review of CRs relating to the CTs prior to the August 2007 failure, revealed several issues relating to structural degradation. Not fully investigating this degradation was a missed opportunity to identify the potential for the collapse that occurred. A full review of events leading up to and the August 2007 event itself are included in another section of this report.

Review of CRs related to the CTs since the failure of CT 2-4 structural members in August 2007 revealed issues with how the repairs were carried out. As an example, CR-VTY-2008-02904 was initiated in July 2008 when an operator observed structural degradation in CT 1-1. This included sagging of the distribution piping and leakage on one slip joint. This was in an area repaired during Spring 2008 work. Connections made during this work were not installed correctly. Similar issues with connections were found in CT 2-4 and CT 2-3.

Certain causes relating to this event could be classifiled as lack of oversight of contractors. These included issues with the contract for the vendor selected, Midwest Towers Incorporated (MTI). The contract did not include requirements for drawings or descriptions of various connections. As a result, work was performed in the field without design drawings. Additionally the contract was awarded to MTI based on its performance at other locations, particularly Palisades (a nuclear plant with similarly constructed cooling towers). However, it was later learned MTI field personnel were not experienced with wood-to-FRP construction.

Per this CR, ENVY Engineering assumed that MTI Engineering accounted for the construction of the CW water header during its analysis which would have developed a bolting pattern to distribute loads from CW pipe to the header and support columns. Per the CR, this incorrect assumption was the single element that would have prevented failure of the header supports. ENVY engineering oversight of MTI Engineering was not adequate to identify this issue.

The CR describes that MTI field personnel performed work without detailed work orders. They contained mostly industrial safety information. There was an over reliance on the skill and expertise of MTI personnel. ENVY Maintenance personnel monitored fieldwork primarily from an industrial safety perspective.

Lack of contractor oversight of CT work was identified as a cause of CT 2-4 collapse in CR-VTY-2007-3243. A full review of this CR including the effectiveness of its evaluation is discussed in Section 1.2.5 of this report. The wording in this CR is very similar to the July 2008 event. There was a different vendor, Tower Performance Incorporated (TPI), responsible for inspections and repairs. However, the CR states that there was an over relianne of the skill of TPI. TPI often worked from memory and did not use detailed work orders or drawings. Also there was excessive delegation and lack of 'trust but verify' behavior by ENVY.

The issues with Contractor oversight identified in the July 2008 event indicate that this issue had not been fully resolved after the 2007 event, which contributed to the July 2008 event. As a follow, up the inspection team observed work being performed by MTI on CT 2-1 during the fall 2008 outage. This included attending a daily work coordination meeting, a morning pre-job brief, and a walk down of the work area. ENVY had several individuals involved in this work. They included the Project Manager (PM), Construction Manager (CM), Project Coordinator, Design Engineer, and System Engineer.

The PM led the coordination meeting. He provided an overview of the work status and a look ahead. He clearly was in charge of the meeting. The MTI representative spoke to specifics such as manpower and schedule progress, which was appropriate.

The NSA assessment team member walked down the work area with the Construction Manager. The CM was a contractor working directly for ENVY Project Management and was very experienced in fieldwork. His responses to questions during the walk down indicated he was aware of his responsibilities he stated he was the single point of contact for MTI and was responsible for MTI performing work as planned. This was inline with the Entergy Procedure for Control of Supplemental Personnel EN-MA-126, which applies to contractors. During the walk-down all personnel were engaged and all were wearing appropriate personnel protective equipment. Overall the area housekeeping was acceptable. The lower deck of CT 2-1 was covered with plywood to eliminate a tripping hazard and to prevent foreign material from entering the CT basin.

The CM led the morning pre-job brief and discussed the following were mentioned:

- Housekeeping is important for safety reasons and must be reinforced staying on top of it. This means piling removing material neatly and organizing it in specific areas until it is removed from site
- Safety incidents in the plant that occurred the day before and discussed how they could happen on CT work
- Reinforcement of configuration management. He explained they are not necessarily replacing "like with like", but are ensuring that what's in the field matches what's in the design, that means they may not put back what was removed, but design drawings will match what's in the field. He told them that a questioning attitude is both helpful and necessary

The PM also addressed the craft during the meeting. He stressed the importance of following the plan and that changes needed to be discussed and approved.

The MTI lead and safety representative also spoke during the meeting. The MTI lead discussed work assignments for the day and the safety representative shared his observations of the previous day relating to safety.

The organizational structure and ENVY personnel involvement, observed during 2F027 CT work was inline with the Entergy procedure EN-MA-126 Control of Supplemental Personnel and standard industry practices.

After the 2008 fall outage, members of the project team were interviewed. Project team members stated that all of the planned scope for the outage and increase scope resulting from inspections was completed. Further discussions relating to contractor oversight revealed that the Construction Manager (CM) and Project Coordinator (PM) who were contractors working for ENVY received orientation prior to the outage per checklists in EN-MA-126 Control of Supplemental Personnel. Additionally the CM was qualified as an upgrade Maintenance Supervisor per the requirements of the procedure. Prior to the outage MTI Managers and Supervisors received orientation per checklists in EN-MA-126. During the interview an inspection team member reviewed copies of completed checklists described above. They were filled out completely and signed and dated.

One area identified as needing improvement was obtaining parts. The inspection team noted this in its observations both prior to and during the fall 2008 outage. All parts were not available at the start of work and this impacted site resources. The CT Project Team needed to spend time expediting parts, creating code numbers, etc. The CT Project Team indicated that there is now a more detailed bill of materials (BOM) for CTs that should help address this issue. Per the fall 2008 outage CT Project Team a critique meeting will be held to identify CT work areas needing improvement and to create actions to address them.

As was mentioned previously, the results of the review to determine if CT repairs have been sufficiently in depth has been mixed. The plans to complete repairs are in place. However, the performance when carrying out these repairs could be improved.

To ensure consistent contractor oversight, the expectations for use of Entergy Procedure for Control of Supplemental Personnel EN-MA-126, has to be made clear and reinforced periodically by ENVY Management. This should include verifying that personnel are using checklists in the procedure. Additionally, at other sites in the industry, individuals are qualified and re-qualified yearly to provide contractor oversight. The training is minimal it focuses on ensuring individuals understand the procedure and their responsibilities. Examples of good and poor practices relating to contractor oversight are also discussed during training.

Relating to cooling tower work planned for the spring over the next few years. The organization set up to provide oversight of the work during RF027 should be established for this work also. This requires committing resources for a longer period than during a Refuel Outage. However, it should be beneficial in ensuring work is completed as designed and planned. Ensuring this work is performed as planned also requires continued management focus since 2008 spring work required rework in July 2008 due to lack of oversight per CR-VTY-2008-2904.

The content of the contract for CT work should be reviewed to ensure issues identified in previous events, such as not understanding the expertise of contractors, providing design drawings, etc. are not repeated.

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2.5.9 Criterion 9 – Modifications

Assessment Response

All design changes are controlled by procedure EN-DC-115, Engineering Change Development. The procedure mandates reviews to ensure compliance with regulatory, code, and industry requirements. It also requires that plant documents be updated to reflect any modification changes prior to modification closure and system return to service. The prior procedure, AP 6008, Rev. 3, *Vermont Yankee Design Change*, contained similar controls. AP 0020 Rev. 78, was also utilized to control temporary and minor modifications to ensure conformance with design intent and to maintain plant configuration and operability requirements. Interviews with plant engineers and managers confirmed their knowledge of these procedures as well as the system design codes and standards and the design basis for the system. Review of selected design documents, including modification packages, has determined that these procedures are being effectively implemented and the original design requirements and bases are being adequately maintained. Quality Assurance audits are also utilized to verify procedure compliance.

The EPU Program included a comprehensive analysis of the effect of EPU on the design and operating basis of the plant system and components and evaluated the acceptability of any changes. The review looked at margin changes at both the component and system level. The review analyzed the EPU impact on margins to ensure there were no unacceptable reductions. The analysis was reviewed and approved by every affected organization at the site. Details can be found in ER No. 04-1409. The analysis is comprehensive, thorough, and consistent with industry practices. It determined that no modifications to the cooling towers were required as a result of EPU.

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END CONFIDENTIAL INFORMATION The cooling tower modification noted below to replace the fans and motors, although done in support of uprate, was not required for uprate, but rather it was done to comply with a Public Service Board order. Thus the original design requirements and bases of the system have been maintained for the EPU conditions.

Two modification packages were reviewed as part of this assessment. The modification package to replace the cooling tower fans and motors, ER-04-0705, and the modification package to replace the cooling tower fill and structural members were reviewed. For both modifications, the changes were compared against the original design bases and determined to be in accordance with those bases. It was also determined that appropriate design documents had been updated as part of the modification closure process.

Based on the document reviews and interviews noted above, all Cooling Tower System modifications comply with the system's original design basis. There should be no negative effects on the future reliability of the Cooling Tower System based on design modifications.

2.5.10 Criterion 10 - Redesign

Assessment Response

Engineering Change Closure Procedure EN-DC-118 requires review of all safety significant analysis prior to closure of any plant modification to ensure safety margins have not been reduced or the reductions were well within safety or operating margins. Engineering Margin Management Procedure EN-DC-195 also requires that maintenance of design and operating margins be considered in engineering changes. Interviews with plant engineers and managers confirmed their familiarity with these procedure requirements. Review of plant records confirmed that these requirements are being implemented in a timely fashion. Review of modification packages also confirmed implementation of these requirements. Quality Assurance audits are also utilized to verify procedure compliance.

ENVY has instituted a formal program to track and disposition any potential changes to design and operating margins. The program assigns responsibilities and due dates and is reviewed on a periodic basis at engineering management meetings.

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Two modification packages were reviewed as part of this assessment. The modification package to replace the cooling tower fans and motors, ER-04-0705, and the modification package to replace the cooling tower fill and structural members were reviewed. For both modifications, appropriate evaluations were done to ensure safety margins were not reduced.

No part of the cooling towers, including cells CT 2-1 and CT2-2, are classified as engineered safeguard systems. There is therefore no single failure criteria requirement applied to the cooling towers.

Based on the document reviews and interviews noted above, changes made to the Cooling Tower System since it's original construction have been reviewed to ensure that safety margins have not been reduced. Each Cooling Tower component modified for up-rate has been reviewed to assure that operational margins have not been reduced. The Cooling Towers are not required to be 'single failure proof'. The practices at ENVY pertaining to redesign applied to the Cooling Tower System meet industry standard expectations.

2.5.11 Criterion 11 - Seismic Analysis

Assessment Response

The agreement with Vermont Department of Public Service on the scope of work for this assessment requires the investigation and assessment of ENVY's seismic analysis program. The following scope fulfills the intended requirements of the seismic analysis investigation and assessment. The scope of the seismic analysis investigation and assessment includes the following:

- Review of ENVY's seismic analysis program to determine their approach to seismic design requirements
- Review of the modification process to determine how seismic design considerations are addressed.
- Review of selected modification packages to determine if appropriate consideration was given to seismic analysis requirements

For selected modifications, NSA verifyied that seismic analysis screening criteria were properly applied and that seismic calculations were performed as required. For the selected systems, NSA reviewed the DBDs and UFSAR to identify their seismic design basis and verify its maintenance in the modification reviews noted above.

The seismic design basis for the plant is documented in Topical Design Basis Document for External Events, Rev 2, dated August 15, 2005. It derives from Draft AEC Criterion 2, Performance Standards. The seismic design criteria are based on ground horizontal acceleration of 0.07g for and Operating Basis Earthquake (OBE) and 0.14g for a Safe Shutdown Earthquake (SSE). The vertical acceleration assumed is equal to 2/3 of the horizontal ground acceleration.

There are two cells in the cooling towers that are Class I seismically designed, CT 2-1 and C T 2-2. The remaining cells are designed for Seismic Class II loads and are structurally separated from CT 2-1 and CT 2-2 by cut-away ties designed to isolate the safety-related cells from the non-safety related cells.

All design changes are being controlled by procedure EN-DC-115, Engineering Change Development. That procedure mandates reviews to ensure compliance with regulatory, code, and industry requirements, including seismic requirements. Interviews with plant engineers and managers confirmed their knowledge of these procedures as well as the system design codes and standards and the original seismic design basis for the system. Review of selected design documents, including modification packages, has determined that these procedures are being effectively implemented and the original seismic design bases is being appropriately factored into the design and is being adequately maintained.

Two modification packages were reviewed as part of this assessment. The modification package to replace the cooling tower fans and motors, ER-04-0705, and the modification package to replace the cooling tower fill and structural members were reviewed. Each modification package contained the appropriate seismic analysis, utilizing the appropriate seismic input, verifying the suitability of the seismic design of CT 2-1 and CT 2-2 for the subject modification.

In 1987, ENVY had a seismic analysis performed for cooling tower cells CT 2-1 and CT 2-2. The analysis was in support of a modification to replace the existing PVC fill with a new improved PVC fill. The analysis was performed by Engineering decision Analysis Company and utilized a computer program called EDAC/MSAP. It is stated that EDAC/MSAP is a general-purpose program and is based on SAP, originally developed by Professor E. L. Wilson at the University of California at Berkeley.

In 2005 ENVY had another seismic analysis performed in support of the cooling tower fan replacement modification. At that time they utilized ABS Consulting for the analysis. ABS Consulting utilized the finite element computer program SAP2000, Version 7.40 for the analysis.

In 2007 ENVY again had cooling towers seismic analyses performed in support of the cooling tower collapse incident. At that time they utilized ARES Corporation for the analysis. ARES Corporation utilized a finite element computer program SAP2000, Version 11.0.4 for the analysis.

Therefore, based on this review, the current design of the seismically designed cells of the Cooling Tower System is in keeping with its original seismic design basis, including OBE and SSE.

2.5.12 Criterion 12 – Training

Assessment Response

The training organization has a process for evaluating all engineering changes to determine whether training is required. This evaluation was performed by conducting interviews with training personnel and by reviewing training processes, governing procedures and training materials. Training is notified of all engineering changes, which are entered on the Modification Training Matrix.

The matrix is routinely evaluated to determine if the change requires the training materials for any training program to be modified. If the evaluation determines training may be required a Training Evaluation/Action Request (TEAR) is initiated. The TEAR assigns actions to training personnel to evaluate specific training materials to determine whether the materials require revision. Actions are also assigned to develop training materials and to conduct training as required. The engineering change is also reviewed to determine if any changes to the simulator are required. If a simulator change is required a Discrepancy Report (DR) is generated to implement required changes. All actions are tracked to completion.

Training also has a process to evaluate whether site and industry operating experience (OE) is to be incorporated into related instructor guides (IG) for classroom and simulator training. Interview with the Operations Training Superintendent revealed an expectation that operating experience is to be included in the development and revision of training material. There is a general understanding of the Entergy Operating Experience procedure (EN-OE-100) including interface with the Site OE Coordinator. There is also a proceduralized requirement for use in the development for training material. Systematic Approach to Training Process procedure (EN-TQ-201) reinforces the need to consider using Operating Experience to re-enforce learning objectives.

Modification ER 04-705 (Cooling Tower Upgrade) was selected based on review of the modification, which involved replacing 21 of the 22 Cooling Tower 125 HP fans with higher 200 HP fans. Review of the modification found that training would be required.

The Operations Training Department reviewed this modification and initiated Training Evaluation Action Request (TEAR) VTY 2005 81. This initiated a NEEDs analysis, which identifies if training is required and if so, what Training Department Information Guides (IG) and/or changes to the Simulator would be required. There were then actions assigned to address the changes. The following Instructor Guides were verified complete and reflect the changes made under modification ER 04-705: Licensed Operator Training (LOT) LOT-00-275 Circulating Water System/Cooling Towers for initial training, Licensed Operator Re-qualification training (LOR) LOR-24-805-2, LOR-24-805, Auxiliary Operator Re-qualification AOR-24-805 for modification training. Discrepancy Report (DR) 05-0600 Cooling Tower Fan/Motor upgrade to 200 HP documented that the changes to the Simulator had been completed and the modification acceptance test had been competed. DR 05-0060 was closed 6/22/2005. Modification training was verified as completed during the July-August 2005 time frame. Systems Training is required every two years (through interview with Operations Training Manager). Operating Experience references were validated to have been referenced in each Instructor Guide. Licensed Operator Training Instructor Guide LOT-00 275 Circulating Water System, Cooling Towers referenced the following Operating Experience documents; Significant Operations Event Record (SOER) 84-1, Licensee Event Report (LER) 93-014 and 9 different internal Condition Reports. Procedures OP 2180 Circulating Water/Cooling Tower Operation and OP4100 Emergency Core Cooling System Integrated Automatic Initiation Test were reviewed and the appropriate modification changes were verified to have been incorporated. Operations Training performance in the review and implementation of modification ER 04-705 (Cooling Tower Upgrade) meets expectations.

As a result of review of the process and interviews with Training Department management, along with review of the actions taken in response to this particular modification, it is concluded the Training Department meets industry standards with respect to evaluating modifications and taking the appropriate actions to incorporate the required changes into training material.

2.5.13 Criterion 13 – Corrective Action Program

Assessment Response

An overall assessment of the corrective action program and its effectiveness was completed and is documented in section 1.2.5 that addresses the Criterion 13 (Corrective Action Program) questions for the overall Corrective Action Program.

A list of CRs back to the year 2000 was reviewed. Certain CRs from this list are discussed in detail in the sections above.

Based on these reviews it was determined that issues related to the Cooling Tower Structure and portions of the Circulating Water system reviewed are being identified and entered into the Corrective Action Process. As an example numerous CRs were created to resolve issues found during inspections. Corrective actions were assigned to appropriately address these issues.

One exception noted relates to effectiveness of corrective actions. Contractor oversight was identified as a contributing cause in CRs relating to the CT collapse event in 2007 and after that event in 2008. A full review of this CR including the effectiveness of its evaluation is discussed in another section of this report.

The CT Structure and CW System CR review supports the conclusion of Section 1.2.5 with the exception noted above.

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References

1. Procedures and Specifications:
2. EN-DC-115, Rev. 5, Engineering Change Development
3. EN-DC-118, Rev. 2, Engineering Change Closure
4. EN-DC-126, Rev. 1, Engineering Calculation Process
5. EN-DC-141, Rev. 5, Design Inputs
6. EN-DC-195, Rev. 2, Margin Management
7. EN-DC-313, Rev. 2, Procurement Engineering Process
8. EN-LI-113, Rev. 3, Licensing Basis Document Change Process
9. AP 6008, Rev. 3, Vermont Yankee Design Change
10. AP 0020, Rev. 78, Control of Temporary and Minor Modifications
11. "Standard Specification for the Design of Cooling Towers with Douglas Fir Lumber"
12. "Mechanical Draft Cooling Towers", Rev 1, dated April 23, 1969.
13. EN-OP-104, Operability Determination
14. EN-LI-102 Corrective Action Process

15. EN-MS-S-001 System Notebooks

16. EN-DC-143 System Health
17. EN-DC-178 System Walk Downs
18. EN-PL-170 Nuclear Asset Management Planning
19. EN-DC-320 Identification and Processing of Obsolete Items
20. EN-DC-100 Operating Experience Program
21. EN-DC-105, Rev. 2, Configuration Management Program
22. EN-MA-126 Control of Supplemental Personnel
23. EN-DC-153 Component Classification
24. OP-52114 CT Structure Inspection and Repairs
25. EN-TQ-201 Systematic Approach to Training Process
26. EN-DC-105, Rev. 2, Configuration Management Program
27. OP-52114 CT Structure Inspection and Repairs
28. OP-2180 CW/CT Operation

Documents

1. Vermont Yankee UFSAR, Rev. 20
2. License Renewal Application, dated January 25, 2006
3. Power Uprate Licensing Application dated, September 10, 2003
4. NRC Safety Evaluation for License Renewal, March 2007
5. Modification Package ER-04-0705, Cooling Tower Fan and Motor Replacement Modification Package, Cooling Tower Fill and Structural Member Replacement
6. DBD for Service Water, Residual Heat Removal Service Water, Alternate Cooling systems, Rev. 29, 7/20/07
7. Topical DBD for External Events, Rev. 2, 8/15/08
8. Task Report W1.4.3.1, Service Water
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12. INPO AP-913 Equipment Reliability
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14. LO-WTENVY-2008-0125 Work Order for CT Inspection Results

15. EC-4721 Cooling Tower Upgrade
16. VTY-2005-81 Training Evaluation Request
17. LOT-00-275 CW/CT Operator Initial Training
18. LOR-24-805 AUX Operator Requalification Training
19. LOR-24-805-2 License Operator Requalification Training
20. AOR-24-805 Operator Modification Training
21. Significant Operations Event Report 84-1
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2. CR-VTY-2007-03425
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2.6 Service Water System

System Description

The ENVY Service Water System requirements are supplied by four vertical pumps which are located in the intake structure. The two pumps for each train take suction from separate bays in the intake structure with all four pumps physically located in the same space. The pumps are normally started and stopped by controls on the main control board. With a design temperature, i.e., the maximum expected river water temperature, of 85^{o} F, the design of the SW system requires three of the four pumps to supply station cooling demands. An additional pump will start automatically if the operating pumps cannot maintain the required system pressure. Check valves are provided on the discharge of each pump to prevent backflow in the event a pump is not operating. The two Service Water pumps in each train are connected to separate division 4160 volt buses that can be supplied by the standby diesel generators in the event of a loss of all off-site power. Under a condition of concurrent Loss of Coolant Accident and loss of off-site power, any two pumps, utilizing emergency power, are capable of supplying the required cooling capacity.

The Service Water System consists of two parallel headers to supply the turbine and the reactor auxiliaries. In each of the two trains, two pumps and a self-cleaning strainer serve each header. Each header supplies cooling water to a reactor building closed cooling water heat exchanger, the Residual Heat Removal system room ventilation coolers, a Diesel Generator cooler and a Residual Heat Removal heat exchanger via the Residual Heat Removal Service Water pumps. The Standby Fuel Pool Cooling Subsystem is supplied only from the Service Water Train B header. Other turbine and reactor auxiliary equipment are supplied from a line tied into both Service Water System headers.

The Circulating Water System pumps and the chlorinator are supplied from the common Service Water header downstream of the Service Water strainers. The Intake Screen Wash is supplied from the Service Water B header.

The Service Water System may be lined up to supply the main condenser hotwell emergency fill line via the turbine building supply header, the Fire System through its connection with the Service Water System, the Nuclear Boiler System through the Residual Heat Removal System intertie, and the spent fuel pool for emergency fill through a branch line from Train B. The Service Water System can also supply the station make-up water treatment system.

In normal operation the Service Water System discharge is routed to the Circulating Water System discharge to the river; however when the river water temperature is below a specified limit, the Service Water discharge is aligned to flow through the cooling tower basin to preclude the basin water from freezing. A process radiation monitor is located on the SW discharge header. There are no automatic actions associated with this monitor. It is alarm-only, to alert the operator of abnormally high radiation levels.

Residual Heat Removal Service Water System

The Residual Heat Removal Service Water System consists of four vertical, four stage, turbine type pumps, piping, valves, and instrumentation necessary to provide cooling water flow through the Residual Heat Removal heat exchangers. Check valves are provided on the discharge of the pumps to prevent back flow through an idle pump. The system is divided into two distinct trains, A and B, each provided with two pumps. The Residual Heat Removal Service Water pumps take suction from the Service Water supply header and the system return is to the Service Water System discharge header. A motor operated valve is provided at the discharge of each Residual Heat Removal heat exchanger to regulate pressure and flow. A Residual Heat Removal Service Water System cross-tie is provided if any post accident flooding of the primary containment is required.

Alternate Cooling Water System

The Alternate Cooling Water System consists of a cooling tower cell (CT 2-1) and fan, cooling tower water basin, Residual Heat Removal Service Water pump and piping, valves, and instrumentation. Cooling water is supplied by gravity flow from the cooling tower basin to the Residual Heat Removal Service Water pumps. Water is then pumped to the RHR heat exchangers, Residual Heat Removal Service Water pump motor bearing oil coolers, emergency diesel generator coolers, Emergency Core Cooling System room coolers, Spent Fuel Pool Cooling System heat exchangers, Residual Heat Removal pump seal coolers and auxiliary piping, valves and instrumentation necessary to provide flow through the Residual Heat Removal heat exchangers, then returned to the cooling tower where the latent heat is transferred to the atmosphere.

This system is used for those events where the Service Water pumps are not available, which may occur if the Intake Structure is submerged during maximum flood conditions or failure of various upstream dams, if the Vernon Dam fails and the river level falls, or if a fire at the Intake Structure disables the SW pumps.

The cooling tower basin is sized such that the system can be operated for seven days before make-up water is required from off-site sources.

2.6.1 Criterion 1 - Initial Conditions

Assessment Response

The ENVY design basis for this system is documented in its UFSAR sections 10.6, 10.7, and 10.8 and in more detail in *Vermont Yankee Nuclear Power Station Design Basis Document for Service Water, Residual Heat Removal Service Water, Alternate Cooling Systems*, Rev. 29, dated July 20, 2007. The verification of the original issue of the Design Basis Document also included an as-built walk-down of the system to confirm compliance of the physical configuration with the design basis documents. Details of this validation are documented in *Validation Report, Design Basis Document SW-1 (Service Water, Residual Heat Removal Service Water, and Alternate Cooling Systems)* dated October 27 1998. Subsequent system modifications were incorporated in the system Design Basis Document in accordance with the requirements of ENVY design change and configuration management procedures.

The codes utilized in the design of the system are as follows:

- ANSI (USAS) B3 1.1.0, Power Piping Code
- ASME Boiler and Pressure Vessel Code (applicable to certain valves and inspections)

The detailed design basis for the Service Water System is described in the Design Basis Document and includes the system functions, regulatory requirements, system design requirements, operating requirements, and electrical and instrument and control requirements.

The power generation design basis functions of the service water system can be summarized as providing water for turbine and reactor auxiliary equipment cooling during normal operation and providing cooling water together with the Residual Heat Removal Service Water pumps for reactor shutdown cooling.

The safety design basis functions of the system are to provide cooling water, both individually and together with the Residual Heat Removal Service Water pumps, for Core Standby Cooling system equipment required during accident conditions, including events when normal power is lost. Further, to supply cooling water for the standby diesel generators, the Standby Fuel Pool Cooling subsystem, and to minimize the probability of a release of radioactive contaminants to the environment by monitoring system discharge and maintaining sufficient pressure at specific areas of the system.

The component design bases are typically performance requirements for pumps, valves, strainers, restricting orifices and the safety related cooling tower cell. These design bases are detailed in the Design Basis Document.

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ENVY was issued an Operating License in March 1972 following issuance of a Construction Permit in December 1967. During the licensing process the ENVY unit was evaluated against the 70 proposed General Design Criteria. Appendix F of the UFSAR contains the original evaluation of the design bases of the facility relative to each of the nine groups of the 70 proposed General Design Criteria. Nine of these criteria apply specifically to the Service Water System and are discussed in the Service Water System Design Basis Document. These nine proposed criteria cover Performance Standards, Fire Protection, Instrumentation and Control, Monitoring Radioactivity Releases, Single Failure Definition, Reliability and Testability of Engineered Safety Features, Emergency Power for Engineered Safety Features, Missile Protection, Engineered Safety Features Performance Capability, Engineered Safety Features, and Accident Aggravation Prevention. The Service Water System complies with all of these criteria. As other NRC regulations and regulatory requirements and requests evolved the Vermont Yankee plant evaluated its compliance with these regulations or requirements to assure the NRC of its compliance with their intent. Those that were applicable to Service Water System ranged from several NRC regulations in 10CFR50, several Regulatory Guides, several Generic Letters and I.E. Bulletins to the Discharge Permit required by state regulatory agencies. All of these are also discussed in the Service Water System Design Basis Document without any exemptions or deviations noted.

Based on the document reviews and interviews it was concluded that the Service Water system design is in keeping with its initial conditions and its current design basis. There should be no negative effects on the future reliability of the Service Water System based on the design of modifications.

2.6.2 Criterion 2 - Procurement

Assessment Response

The ENVY processes governing Engineering Changes require that applicable calculations be performed to support the engineering change and that they be verified as built prior to return to service of the implemented engineering change. The procedure for performance of calculations prescribes the requirements and the format for the preparation of the calculations. The return to service form of the Engineering Change Closure Procedure requires a sign-off that calculation changes be verified as-built.

There were no procurement changes associated with the modifications to the Service Water System for EPU or were there any procurement changes associated with the modification that was reviewed for this assessment which was not related to the EPU.

2.6.3 Criterion 3 - Installation – "as-built"

Assessment Response

In February 1997, ENVY responded under oath to the USNRC 10 CFR 50.54(f) letter to Licensees requesting information regarding adequacy and availability of design bases. In its response ENVY stated it has 'reasonable assurance' that the ENVY design bases have been adequately translated to the plant design and procedures and that plant configuration is in an appropriate manner. ENVY went on to state that it was committing to provide improved configuration management including completion of a Design Basis Documentation program, improved Technical Specifications program and a FSAR Verification program. The subsequent development of the DBDs, which took place in 1997 and 1998, included a DBD validation process in which validation teams verified both the design basis information and its application to design documents, and its application to operations, maintenance, surveillance, testing functions and physical configuration. The physical configuration validation included as-built walk-downs. Walk-downs were performed to verify that the system configuration reflects the design basis, is in agreement with plant drawings, and the component labeling is adequate.

Since the completion of the DBD the engineering procedures for modification development and their closure included and emphasized the requirements of as-built updating of pertinent engineering, operations, training, maintenance, program, and licensing documentation. The closure procedure specifies which updates are required prior to return to service and which are required to be tracked and completed after return to service. The closure procedure includes a check list with departmental responsibilities for updating the required documentation.

Interviews with engineering managers and engineers demonstrated that they were knowledgeable in these procedural requirements and that procedure compliance was good.

Inspection of the UFSAR, DBD, and flow diagrams (P&ID) of the Service Water System found that the system modifications required for EPU were as-built and properly reflected in these documents.

As part of this inspection the team conducted an as-built walk-down of the Service Water pump intake structure and concluded that the physical configuration of the pumps, associated auxiliaries, piping and valves were consistent with their presentation in the physical drawings of the facility.

Based on the document reviews, interviews and the team walk down, it was concluded that the ENVY configuration management processes meet industry expectations and that the design of the Service Water System is properly documented in the plant records and procedures and the records properly reflect the as-built condition of the plant.

2.6.4 Criterion 4- Operation

Assessment Response

Unanticipated operations outcomes for this assessment considered plant operating or equipment conditions that did not result in the desired or expected outcome. Interviews with operations management were conducted in order to review the processes used to identify and correct each unexpected operational issue. How Operations implements the governing procedures and processes was discussed with operations management. An example was selected of an event within the service water system and reviewed to ensure appropriate compensatory actions were taken, and that plant procedure deficiencies were addressed as appropriate. Additionally it was evaluated to determine if the appropriate level of analysis was performed in accordance with the corrective action system.

As part of the CR evaluation it is determined whether an Operability Evaluation (in accordance with EN-OP-104, Operability Determination procedure) is required for degraded safety related equipment. The CR evaluation also determines whether an Operational Decision Making Instruction is required for degraded equipment.

An Operability Evaluation is a formal process that is performed in two phases. The first step is performed by operations as an interim evaluation to determine if there is reasonable assurance that a degraded safety related system or component can perform its intended safety function. The second step is a more comprehensive evaluation by engineering that justifies operability for the system or component. In some cases the Operability Evaluation specifies compensatory actions that must be completed to ensure operability.

An Operational Decision Making Instruction is a formal process that justifies continued operation of a degraded system or component. It is developed by operations and/or engineering and reviewed by CRG. In some cases the Operational Decision Making Instruction specifies compensatory actions that must be completed to continue to operate the system.

The compensatory actions that must be completed by operations on an interim basis can be included in operator equipment inspection instructions, operator logs, standing orders, turnover sheets and operating procedures. These compensatory actions are discussed by the operating crew at the beginning of each shift.

Condition Report VTY-2006-02184-'A' Service Water pump high winding temperature was selected as an event driven action taken in response to an unanticipated plant event. As part of normal operations rounds performed by Operations Auxiliary Operators (AO) inspections are performed twice daily. On October 17, 2007 the AO reported the 'A' Service Water pump winding temperature had exceeded its maximum design temperature of 266 degrees. The pump was removed from service and the Condition report was initiated. The CR was classified as Significance Level "C" (condition adverse to quality requiring correction). Based on this classification "C" no root cause was required or requested. This is in accordance with procedure EN-LI-102 Corrective Action Process. An Operability Evaluation was performed and determined the Service Water pump to be Operable. An Operations Decision Making Instruction (ODMI) was assigned. ODMI is a plant process which involves formulation of a peer group or individual in order to evaluate the condition and determine if interim actions would be required pending repairs. The ODMI was initiated to assist Operations personnel in responding to high Service Water pump winding temperatures including the need to maintain Service Water pressure greater than 90 psig. The ODMI Action plan established trigger points (based on winding temperatures) and corresponding actions to be taken should they be exceeded. The ODMI was verified logged in the ODMI log. The requirement for each shift to review the new ODMIs satisfied the communications requirement (there is no formal verification method). Based upon review the appropriate compensatory actions were implemented.

Based on the process review, interviews with Operations management and the actions taken to address this specific plant issue, it is concluded that the ENVY response to unanticipated Plant operational outcomes meets expectations consistent with industry good practices.

2.6.5 Criterion 5- Testing

Assessment Response

Multiple interviews with the System Engineer were conducted to determine the operating performance of the Service Water (SW) System over the past three years, with a focus on the past twelve months. As stated in the system description section, the Service Water System consists of two parallel headers to supply the turbine and the reactor auxiliaries. In each of the two trains, two pumps and a self-cleaning strainer serve each header. Each header supplies cooling water to a reactor building closed cooling water heat exchanger, the Residual Heat Removal system room ventilation coolers, a Diesel Generator cooler and a Residual Heat Removal heat exchanger via the Residual Heat Removal Service Water pumps. The assessment focused on surveillance testing results, preventive maintenance and predictive maintenance activities over the past three years.

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ENVY has formal processes for identifying testing requirements and methods as well as the actual testing of the Service Water system. The corrective action system tracks the implementation of corrective and compensatory measures for the Service Water System when issues arise. These processes are consistent with industry practice.

2.6.6 Criterion 6 – Inspection

Assessment Response

Service Water System Inspections consisted of a wide range of activities including the System Health/Component Health Reporting process (EN-DC-143), System Engineering Walk-downs (EN-DC-178) and the System Monitoring Program (EN-DC-159). The output of these core system engineering processes, provide the basis for this assessment.

A review of the past 6 months of system walk-down reports as well as participating and observing the System Engineer during an October 2008 walk-down provided an opportunity to see the actual implementation process. The walk-down was consistent with industry practices based on reviewing the implementation procedure and past reports. Condition Reports were initiated for appropriate material condition issues.

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ENVY has formal processes for identifying inspection requirements and methods as well as the actual inspection of the Service Water System. The corrective action system tracks the implementation of corrective and compensatory measures for the Service Water System when issues arise. These processes are consistent with industry practice.

2.6.7 Criterion 7 – Maintenance

Assessment Response

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Based on discussions with the System Engineer a review of EN-DC-153 (Preventive Maintenance Component Classification) was conducted to determine the basis for how the Service Water System is maintained. It was concluded that the procedural processes utilized by ENVY are consistent with current industry standards and practices**.**

In addition, a review of a specific PM basis document (ME057 Rev.8), was conducted for Service Water P-7-1B (Pump/Motor) and it was concluded that the overall approach and requirements were consistent with industry standards. This document establishes functional importance determinations that include Risk and Safety Considerations consisting of questions such as: Safety Related, Safe Shutdown (App. R) PRA/IPE, Required by Tech Specifications, EEQ Program and Emergency Operating procedures.

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ENVY has formal processes for identifying maintenance of the Service Water System. These processes are consistent with industry practice.

2.6.8 Criterion 8 – Repairs

Assessment Response

Repairs to the Service Water System were based on a review of maintenance activities, Predictive Maintenance watch list items and selected maintenance activities which could affect system reliability:

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Overall, repairs at ENVY are prioritized and scheduled per the Work Management Process. This includes work while the unit is on-line and during outages. Numerous procedures govern the work management process.

Overall, ENVY has a low corrective maintenance backlog, which indicates that repairs are completed in a timely manner. Another indication that issues are being identified and addressed is that CRs are created to address minor issues identified during testing, walk-downs, etc.

Review of the information provided indicates that repairs have been sufficiently in-depth for the Service Water System. START CONFIDENTIAL INFORMATION

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Based on the review of the documents, and interviews described above, it was concluded that the Service Water System repairs are identified and address utilizing industry accepted practices.

ENVY has formal processes for identifying repair requirements and methods as well as the actual repair of the Service Water System. The corrective action system tracks the implementation of corrective and compensatory measures for the Service Water System when issues arise. These processes are consistent with industry practice.

2.6.9 Criteria 9 – Modifications

Assessment Response

All modifications are controlled by procedure EN-DC-115, Engineering Change Development. That procedure mandates reviews to ensure compliance with regulatory, code, and industry requirements. It also requires initiation and processing of updates to plant configuration documents and plant design, licensing and operating margins. Procedure EN-DC-118, Engineering Change Closure, requires that all plant design, licensing, operation, training and maintenance documents be updated prior to modification closure.

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This modification provided an additional flow path from the Service Water System to the Spent Fuel Pool Cooling System but did not change the system functions as contained in the original design basis. All documents including UFSAR, DBD, drawings, and procedures were updated as required to reflect the implemented modifications.

Based on the review of the documents and the modification described above, it was concluded that the Service Water System complies with the original design basis functions and its current design basis. The plant procedures have been updated to reflect the modifications. The ENVY modification processes are consistent with industry expectations. There should be no negative effects on the future reliability of the Service Water System based on design of modifications.

2.6.10 Criteria 10 - Redesign

Assessment Response

Modifications are controlled by procedure EN-DC-115, Engineering Change Development. That procedure mandates reviews to ensure compliance with regulatory, code, and industry requirements. It also requires initiation and processing of updates to plant configuration documents and plant design, licensing and operating margins. EN-DC-195, Margin Management procedure, requires that maintenance of design and operating margins be considered in engineering changes.

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The Service Water System review concluded that there were no changes to the redundancy or single failure characteristics of the system associated with the EPU.

Based on the document reviews described above there were no margin reductions regarding the safety functions associated with the Service Water System. There were no unacceptable margin reductions although the EPU review did identify the need to operate two Residual Heat Removal Service Water pumps through one heat exchanger instead of one pump as had been required previously. ENVY processes and procedures pertaining to redesign are consistent with industry expectations

2.6.11 Criteria 11- Seismic Analysis:

Assessment Response

The agreement with the Vermont Department of Public Service on the scope of work for this assessment states the following regarding seismic analysis.

This audit includes the investigation and assessment of ENVY's seismic analysis program. The following scope is proposed to fulfill the intended requirements of the seismic analysis investigation and assessment.

The scope of the seismic analysis investigation and assessment includes the review of the following:

- Review ENVY's seismic analysis program to determine their approach to seismic design requirements
- Review the modification process to determine how seismic design considerations are addressed
- Review selected modification packages to determine if appropriate consideration was given to seismic analysis requirements

For selected modifications, NSA will verify that seismic analysis screening criteria were properly applied and that seismic calculations were performed as required. For the selected systems NSA will review the DBDs and UFSAR to identify their seismic design basis and verify its maintenance in the modification reviews noted above.

The seismic design basis for the plant is documented in Topical Design Basis Document for External Events. The seismic design criteria are based on ground horizontal acceleration of 0.07g for an Operating Basis Earthquake (OBE) and 0.14g for a Safe Shutdown Earthquake (SSE). The vertical acceleration assumed is equal to 2/3 of the horizontal ground acceleration.

The safety-related portions of the Service Water/Residual Heat Removal Service Water/Alternate Cooling subsystems designated as Seismic Class I and are designed to withstand the effects of a Safe Shutdown Earthquake and perform their intended safety function. Those portions of the Service Water System supplying safeguard equipment and the non-isolatable portions of the service water supply are of Class I seismic design. The Residual Heat Removal Service Water System is also of Class I seismic design.

That portion of the intake structure which houses the Service Water System equipment is of Class I seismic design. The remainder of the Service Water System is Seismic Class II, i.e., designed to withstand the effects of an Operating Basis Earthquake and perform its function.

UFSAR Section A.9 provides a description, scope, and design methodology used for the reanalysis of seismic class I piping subsequent to initial operation. Service Water System class I piping was reanalyzed using computer dynamic analyses to evaluate seismic loadings. Ground spectra based Regulatory Guide 1.60 and Regulatory Guide 1.61 damping or floor spectra with ASME code case N-411 damping defined the seismic loading for these piping systems. Seismic analyses were performed for the Safe Shutdown Earthquake scenario and the piping was evaluated to ANSI B31.1 - 1977 code allowables.

The Service Water System modification, VYDC 2000-024, providing the cross-tie to the Spent Fuel Pool cooling was designed and analyzed in accordance with class I seismic requirements including the performance of supporting calculations.

Based on this review the current design of the Service Water System is in keeping with its original seismic design basis, including Operating Basis Earthquake and Safe Shutdown Earthquake.

2.6.12 Criteria 12 – Training

Assessment Response

The training organization has a process for evaluating all engineering changes to determine whether training is required. This evaluation was performed by conducting interviews with training personnel and by reviewing training processes, governing procedures and training materials. Training is notified of all engineering changes which are entered on the Modification Training Matrix. The matrix is routinely evaluated to determine if the change requires the training materials for any training program to be modified. If the evaluation determines training may be required a Training Evaluation/Action Request (TEAR) is initiated. The TEAR assigns actions to training personnel to evaluate specific training materials to determine whether the materials require revision. Actions are also assigned to develop training materials and to conduct training as required. The engineering change is also reviewed to determine if any changes to the simulator are required. If a simulator change is required a

Discrepancy Report (DR) is generated to implement required changes. All actions are tracked to completion. Training also has a process to evaluate whether site and industry operating experience (OE) is to be incorporated into related Instructor Guides (IG) for classroom and simulator training. Interview with the Operations Training Superintendent revealed an expectation that operating experience is to be included in the development and revision of training material. There is a general understanding of the Entergy Operating Experience procedure (EN-OE-100) including interface with the Site OE Coordinator. There is also a proceduralized requirement for use in the development for training material. Systematic Approach to Training Process procedure (EN-TQ-201) reinforces the need to consider using Operating Experience to re-enforce learning objectives. Modification ER 06-1498 (Modify Service Water Strainers for Manual Backwash) was selected based on review of the modification. Review of the modification found that training would be required. This modification provided a manual hand wheel to each strainer to provide for manual backwash capability in a loss of power event. The Operations Training Department reviewed this modification and initiated Training Evaluation Action Request (TEAR) VTY 2006-473. This initiated a NEEDs analysis which identifies if training is required and if so, what Training Department Information Guides (IG) and/or changes to the Simulator would be required. There were then actions assigned to address the changes. The following Instructor Guides were verified complete and reflect the changes made under modification ER 06-1498: Licensed Operator Re-qualification LOR-25-805 and Auxiliary Operator Training AOR-25-805 for modification training. There were no changes required for the simulator. Modification training was verified through Training department tracking documentation to have been completed during the Jan-Feb 2007 time frame. Operating Experience references were validated to have been referenced in Instructor Guide LOR-25-805 and included Licensee Event Report 97-024 and two Condition Reports. Procedures, Off Normal (ON) 3148 Loss of Service Water and Control Room Panel (CRP) 9-6 Alarm Response, were reviewed and the appropriate modification changes were verified to have been incorporated. Operations Training performance in the review and implementation of modification ER 06-1498 (Modify Service Water Strainers for Manual Backwash) meets expectations. As a result of review of the process and interviews with Training department management, along with review of the actions taken in response to this particular modification, it is concluded the Training department meets industry standards with respect to evaluating modifications and taking the appropriate actions to incorporate the required changes into training material.

ENVY has formal processes to ensure that when engineering changes are implemented for the Service Water System that reviews are conducted and changes made as required to operating procedures, training materials and the simulator to accurately reflect plant conditions. These processes are consistent with industry practices.

2.6.13 Criteria 13 - Corrective Action Program

An overall assessment of the corrective action program and its effectiveness was completed and is documented in section 1.2.5 that addresses the Criterion 13 (Corrective Action Program) questions for the overall Corrective Action Program.

A list of CRs back to the year 2000 was reviewed. Certain CRs from this list are discussed in detail in the sections above.

Based on these reviews it was determined that issues related to the SW system are being identified and entered into the Corrective Action Process. Corrective actions were assigned to appropriately address these issues.

Based on the process review discussed above it is concluded that the ENVY CR process for the service water system is being implemented consistent with industry practices.

2.6.14 Standard Review Plan Review

Standard Review Plan Review

The Vermont Legislation requested an assessment of 'deviations…from any regulatory requirements applicable to new nuclear reactors'. In subsequent discussions with the Vermont Department of Public Service, it was agreed that NSA would select two systems, HPCI and Service Water, from the existing list of systems already selected for review to evaluate them against applicable sections of the current Standard Review Plan. NSA would evaluate the differences between the ENVY Design Basis Document and the Standard Review Plan and assess the difference(s) for the effect on reliability.

Assessment Response

A special assessment was performed of the design basis of the Service Water System against the design requirements of current plants. As noted previously, the design basis of the Service Water System is contained in the Service Water System DBD. The design basis requirements for current plants which are also applicable to new plants are contained in NUREG-0800, U. S. Nuclear Regulatory Commission Standard Review Plan, Section 9.2.1 Station Service Water System, Revision 5, dated March 2007. This assessment was performed by comparing the General Design Criteria that were applied to ENVY during its licensing period against the equivalent General Design Criteria in the Standard Review Plan, Section 9.2.1, that applies today. The ENVY Service Water System was evaluated against these criteria and was found to be in compliance with their intent. This compliance is documented in the system DBD.

Service Water System Conclusions

Based on the document reviews and interviews noted above the following is a summary of SW System assessment conclusions. More detail is provided at the end of each individual section above.

The ENVY engineering design process is well documented, controlled and consistent with industry practice. The current design of the SW System is consistent with its original design basis and is adequately reflected in plant records and procedures.

ENVY has formal processes for identifying and correcting unanticipated operations outcomes. The corrective action system tracks the implementation of corrective and compensatory measures for the SW System when issues arise, including the revision of appropriate documents. These processes are consistent with industry practice.

ENVY has formal processes to ensure that when engineering changes are implemented for the SW System that reviews are conducted and changes made as required to operating procedures, training materials and the simulator to accurately reflect plant conditions. These processes are consistent with industry practices.

The data reviewed and interviews indicate that the SW System is tested, inspected, maintained and repaired to industry standards such as AP-913 Equipment Reliability Process. Component classification was performed; PM Basis documents created, and testing and inspections were scheduled and carried out. CRs were created to resolve issues found during testing and inspections.

START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION

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33. EN-DC-143, Rev. 8 (August 11, 2008), System Health and Component Health Reporting Process
34. ENN-SEP-IST-001, Vermont Yankee In-service Testing (IST) Program Section
35. OP4181, Service Water/Alternate Cooling System Surveillance
36. EN-DC-178, Rev. 2 (February 8, 2008) System Engineering Walk Downs
37. EN-DC-159, Rev. 2 (February 8, 2008), System Monitoring Program
38. EN-DC-153, Rev. (July 28, 2008), Preventive Maintenance Component Classification
39. EN-PL-170 Rev. 1 (Effective Date January 7, 2008), Entergy Nuclear Asset Management Planning
40. OP5265, Rev. 16 (June 26, 2008) Service Water Component Inspection and Acceptance Criteria -
41. VYC-2251, Rev. 1, Evaluation of Service Water UT Data.
42. PM Basis: ME057, Rev. 8 – Pump and Motor
43. PM Basis: ME022, Rev. 8 - Motor
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45. PM Basis: M270, Rev. 6 – Valve, Manual
46. Component Classification Status (Systems) Report (Wednesday, October 1, 2008)
47. Predictive Maintenance Watch Report (Date Accessed: September 8, 2008)
48. SW System MIC Index Status Report (Current September 20, 2008)
49. Component Health Report – Pumps (2nd Quarter 2008)
50. Performance Parameter Spreadsheet – Component Type: Motors – (Report for Quarter: 2008-Q2)
51. System Monitoring Plan – Service Water (Date Revised: September 13, 2007 – HRB; Date Reviewed: September 13, 2007 – KLF)
52. Infrared Inspection Program Schedule
53. Oil Sampling Program Schedule
54. Vibration Monitoring Program Schedule
55. Interviews with System Engineer.
56. Service Water Long Range Improvement Plan, Revision: Original

2.7 Electrical System: Back up and Stand-by [3(a)(1)]

Introduction

The plant electrical system provides power to the plant electrical components so that they can perform their required functions. It consists of an offsite AC system and an onsite standby power system. The offsite AC system, also known as the preferred power system, includes two physically independent circuits capable of providing power to the on-site electrical components independently of the on-site standby power system. It comprises the switchyard, the transmission system, and the switchyard equipment and controls needed for the operation and protection of the grid system. The on-site power system includes safety and non-safety-related networks that provide power to the various electrical components at the required voltage. START CONFIDENTIAL

END CONFIDENTIAL INFORMATION

The purpose of NSA electrical system review was to evaluate the Nuclear Regulatory Commission (NRC) findings associated with the electrical system identified primarily during three inspections, and confirm that the ENVY actions to address these findings were effective. These inspections included the 1992 electrical distribution system functional inspection (EDSFI), the 2004 component design basis inspection (CDBI), and the 2008 CDBI. NSA evaluated the effectiveness of corrective actions resulting from the Licensee Event Reports (LERs) associated with the electrical system since the 1992 EDSFI. The NRC findings and the effectiveness of ENVY's resolution of the findings are discussed, as applicable, in the subsections below. These subsections also include the LERs and the resulting corrective actions.

Methodology

The evaluation of the electrical system issues was performed through a review of inspection records, applicable design and operation documents, personnel interviews, equipment inspections when appropriate, and a comparison of ENVY practices to industry standards. The document reviewed included applicable sections of the UFSAR (Chapter 8) and Technical Specifications, NRC inspection reports addressing findings and resolution, condition reports and internal evaluations and resolution of the findings, and design calculations, when applicable. Interviews of cognizant engineers addressed the issues individually and in general to confirm that equipment involved could perform its intended functions.

Emergency Diesel Generators

Introduction

As indicated above, the emergency diesel generators (EDG) provide emergency AC power to the safety-related and important-to-safety plant loads. They also supply power to the DC system loads through the safety-related battery system. ENVY has two emergency diesel generators that are sized to accept the safety-related loads following a loss of offsite power with or without a loss of coolant accident.

In 1992, the NRC reviewed the ENVY electrical system to confirm its capability to perform its design safety functions. This review, known as the EDSFI, identified a number of issues related to the emergency diesel generators, as detailed below. In 2008, the Component Design Basis Inspection (CDBI) team evaluated several EDG design features that had been previously reviewed by the EDSFI team and follow-up inspections. The team found those features acceptable except for the EDG loading test criteria. Details of that issue are also provided below.

The purpose of the NSA team review was to verify that the ENVY's actions to address the NRC identified emergency diesel generators issues were adequate and that, based on its evaluation, reasonable assurance exists regarding the EDGs capability to provide quality and reliable power to the safety-related loads.

Observations

Item No. 92-81-01, Inadequate Valve In-service Test

START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION

The NSA team's evaluation of this issue concluded that the modification had removed the original cause for concern. Additionally, the CDBI team's 2008 review of the air starting system identified no findings of significance. The NSA team's evaluation uncovered no additional areas of concern with this issue.

Item No. 92-81-05, Diesel Generator Loading

The EDSFI team's concern with the diesel generator loading was twofold. START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION

The NSA team evaluation of the diesel generator loading issues concluded that: a) ENVY's resolution of the discrepancies between the load analysis and the ECCS integrated test were appropriate; and, b) the design bases for the load analysis were accurate. Since the evaluation uncovered no additional areas of concern with this item, the team also concluded that the NRC closure of this item was appropriate.

Item No. 92-81-09, Fuel Oil Transfer Pump Submersion

START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION The NSA review of this issue identified no further areas of concern.

Item No. 92-81-10, Fuel Oil Temperature Control

START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION

The NSA team's evaluation of the fuel oil temperature issue concluded that the analysis performed by ENVY was reasonable and that the follow-up actions were appropriate. Additionally, the team determined that the fuel oil system was evaluated by the 2008 CDBI team and found acceptable. The NSA team uncovered no additional areas of concern with this issue.

Item No. 92-81-11, Fuel Oil Transfer Piping

START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION

Based on the above review, the NSA team concluded that ENVY's resolution of the NRC concern was reasonable and that the resulting corrective actions were appropriate. No additional areas of concern were uncovered by the NSA team.

Item No. 92-81-12, Diesel Generator Room Temperature

START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION

The NSA team's evaluation of this issue found the licensee's review and conclusions reasonable and the corrective actions acceptable. Additionally, the 2008 CDBI team evaluated the diesel generator room ventilation exhaust and found it acceptable. The NSA team identified no additional areas of concern regarding diesel generator room temperature.

Item No. 08-08-02, Inadequate Design Control for Emergency Diesel Generator Testing Criteria

START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION

The NSA team evaluation of the issue and resolution by ENVY concluded that the corrective actions, when implemented by ENVY, are appropriate and in accordance with industry standards. The NSA team's review of this item identified no additional issues pertaining to generator testing.

Emergency Diesel Generator-Related Licensee Event Reports

ENVY's review of its database determined that the following Licensee Event Reports (LERs) related to the emergency diesel generators:

- LER-96-022-00, "Combination of Poor Man-Machine Interface, an Inadequate Procedure, a Mechanical Failure, and Inadequate Operating Experience Review Results in an Emergency Diesel Generator to Exceed Tech Spec Outage Time." Corrective Action 13 of CR 2008-03359 required the creation of a surveillance procedure to require a once3 per cycle test to demonstrate the capability of the diesel generators to supply the short time rating at 3025kw for two hours. This procedure is required before the next ENVY Refuel Outage.
- LER-96-029-00, "Process and Communication Inadequacies Result in the Failure to Analyze Emergency Diesel Generator Fuel Oil within Time Allotted by Technical Specification Surveillance Requirements,"
- LER-97-019-00, "An Inadequate Hydraulic Calculation Performed in Support of a 1982 Fire Protection Sprinkler System Modification Allowed Degradation of Redundant Emergency Diesel Generator Manual Sprinkler Sub-Systems."
- LER-98-04-00, "Seven-Day Diesel Generator LCO Exceeded Due to Inadequate Instructions in the Work Control Process Regarding Block Walls."
- LER-98-21-00, "Inadequate Licensing Basis Documentation Retrievability Results in the Failure to Meet IST Requirements for Diesel Fuel Oil Day Tank Level Control Valves,"

The NSA team's review of the above EDG-related LERs and of the ENVY evaluations and corrective actions described in the LERs concluded that such evaluations and corrective action were reasonable and that they correctly addressed each issue. This NSA review identified no further areas of concern.

Conclusions

Based on the above review, the NSA team concluded that the emergency diesel generator issues identified by the NRC and described above were correctly addressed by ENVY personnel. As indicated earlier, the 2008 CDBI team evaluated several EDG design features and found them acceptable. Similarly, within the scope of the above reviews, the NSA team concluded that the existing controls should provide reasonable assurance about the design capability and reliability of the emergency diesel generators.

Batteries

START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION

The station battery systems were reviewed by the NRC during the 1992 EDSFI to confirm the capability of the DC system to perform its design safety functions. All batteries systems were reviewed again during the 2008 CDBI. The findings identified by the NRC teams are described below.

The purpose of the NSA team review was to verify that the ENVY's actions to address the NRC identified battery issues were adequate and that, based on NSA's review, reasonable assurance exists regarding the batteries' capacity and capability to provide quality and reliable DC power to the safety-related loads.

Observations

Item No. 92-81-08, DC Bus Cross Connections

START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION The NSA team uncovered no additional areas of concern and concluded that closure of this item was appropriate.

Item No. 92-81-13, Battery Room Ventilation

START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION

The NSA team's evaluation of this item found the licensee's actions reasonable and the resolution acceptable. The NSA team uncovered no additional areas of concern with this issue.

Item No. 08-08-01, Inadequate Testing of Safety-Related Batteries

START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION

Conclusions

Based on the above review, the NSA team concluded that the battery issues identified by the NRC and described above were correctly addressed by Entergy. START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION Therefore, the NSA team also concluded that, based on its review, reasonable assurance exists about the design capability and reliability of the ENVY batteries.

Vernon Dam Tie

Introduction

Section 50.63 of Title 10 of the Code of Federal Regulations requires that nuclear power plants be able to withstand and recover from a station blackout (SBO) involving the loss of the off-site sources concurrent with a reactor trip and the loss of the on-site standby sources (emergency diesel generators). Subsequently, the NRC issued Regulatory Guide (RG) 1.155 which provided guidelines for complying with the SBO Rule. The RG was issued concurrently with similar guidelines prepared by the Nuclear Management and Resource Council (NUMARC) in its document NUMARC 8700. The guidelines by NRC and NUMARC provided licensees the procedure for determining the plant SBO coping capability and duration. The evaluation was intended to determine the 'likelihood and duration of the loss of the off-site power, the redundancy and reliability of on-site emergency AC power systems, and the potential for severe accident sequences after a loss of all AC power.'

In Section 3.2.5 of the RG, the NRC indicated that onsite or nearby alternate AC (AAC) power sources that were independent and diverse from the onsite emergency AC power sources could be used to cope with the SBO. The RG also stated that, if the AAC source met the criteria of the RG and could be demonstrated by test that it would be available within 10 minutes; a coping analysis was not required.

START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION

The purpose of the NSA team review was to verify that the ENVY's actions to address the NRC identified SBO issues were adequate and that, within the scope of review, there is reasonable assurance that the required electrical power sources are available to cope with the SBO event.

Observations

Item No. 04-08-01, Availability of Power from the Vernon Station

In 2004, the NRC conducted a component design basis inspection (CDBI). During the inspection the NRC evaluated the adequacy of the onsite and offsite electrical power sources. The areas evaluated by the CDBI team included the availability of the Vernon Hydro-Electric Station electrical power source that ENVY had credited for coping with a station blackout. START CONFIDENTIAL INFORMATION

END CONFIDENTIAL INFORMATION As also indicated in Section 1.1.2, above, the NSA team concluded that, within the scope of review, reasonable assurance exists about the design capability and reliability of the ENVY batteries.

Electrical Connections and Controls

Introduction

The ENVY power needs are normally supplied by the main generator through the unit auxiliary transformers. When the main generator is not operating, offsite power to the ENVY buses is supplied by the 115KV switchyard through the startup transformers. The 115KV switchyard is connected to the 345KV switchyard and transmission system through an autotransformer. A second source of power to the 115KV switchyard is derived from the 115KV Keene line.

An automatic transfer system exists between the unit auxiliary and startup transformers. Accordingly, a main generator trip and consequent loss of the normal supply results in the plant electrical loads being transferred automatically to the startup transformers that are powered by the autotransformer or the Keene line or both, depending on the load flow at the time of the transfer. The autotransformer has the capacity and capability to supply all the plant's load and the automatic transfer system was designed assuming its continued availability. The capability of the Keene line to supply the plant loads depends on the loading conditions of the line at the time of the transfer. Therefore, during normal plant operation, if the autotransformer fails or is otherwise unavailable, the plant enters a 7-day license condition for operation (LCO).

The purpose of the NSA team's review of this area was to confirm that adequate controls exist for ENVY to verify the capability of the offsite sources to provide reliable power to the plant loads during all modes of plant operation.

Observations

Item No. 04-08-02, Procedure for Assessing Offsite Power Availability

The CDBI team's review of operation procedure ON 3155, *Loss of Autotransform*er, observed that, in the event of a loss of the autotransformer, the operators were required to contact the Independent System Operator (ISO)-New England and determine the Keene line load limit, but the procedure provided no criteria for evaluating the operability of the line. The team also observed that a note on "Reference D" of the procedure required that ENVY motors be started sequentially, not simultaneously. The team's concern with this latter issue was that, in the event of a LOCA signal, if the offsite sources are available, the safety loads are designed to start simultaneously, not sequentially. Therefore, the simultaneous start of all safety-related loads could render the Keene line unavailable. Lastly, the team observed that the procedure allowed operation of the plant with a minimum bus voltage of 3600V, below the technical specification-stated minimum voltage of 3660 Volts. The concern with this issue was that the operation of the safety-related motors at the procedure-allowed voltage could result in improper equipment operation or damage.

In the analysis of the findings, the NRC evaluated the procedural deficiencies and concluded that such deficiencies did not cause an inadequate assessment of the off-site power operability by ENVY, nor did they cause the ENVY electrical system or components to be inoperable. Based on this conclusion, they issued a non-cited violation.

Regarding the first issue, as stated in condition report CR-VTY-2004-2803, the capability of the Keene line is primarily determined by the ISO. The programs at their disposal are capable not only to evaluate the immediate capability of the line, but also to predict future loads and, hence, its future capability. As stated previously, in the event of a loss of the autotransformer, the ENVY Technical Specifications requires that Operations declare a 7-day LCO. Therefore, the operators' actions are dictated by the requirements of the Technical Specification. In the intervening period, if the need arises, pursuant to discussion with the ISO and appropriate operability evaluation, ENVY can declare the Keene line qualified as an immediate power source and exit the LCO. However, if the conditions of the line change, the ISO is required to inform ENVY immediately and ENVY is required to re-enter the LCO. To ensure that the operating procedure adequately addressed this issue, ENVY revised the procedure and added a clarification.

Regarding the second issue (i.e., loading of the Keene line), ENVY recognized that, with the loss of the autotransformer, the operability of the ENVY electrical system was dependent on the capability of the Keene line to provide quality power to all loads under all plant operating conditions and accidents. The Keene line, like the autotransformer, is a standby source. The loss of the autotransformer results in the Keene line being the only standby source available on the 115KV switchyard. Therefore, ENVY must enter an LCO. This LCO assures that an adequate evaluation is performed of the capabilities of the Keen line to perform its intended function. Exiting the LCO also requires ENVY to perform an operability evaluation that must consider all operating conditions, including transients and accidents.

Regarding the minimum bus voltage requirements issue, ENVY concurred with the NRC observation and raised the minimum voltage allowed by the procedure from 3600 Volts to 3700 Volts.

Conclusions

Based on its review of the documentation provided by ENVY and discussions with plant personnel, the NSA team concluded that the Keene line concerns by the NRC were adequately addressed by the ENVY personnel and that, based upon this review, reasonable controls exist for the continued operability and reliability of the offsite sources. The procedure change pertaining to bus voltage was also considered reasonable. The minimum voltage issue is also addressed in Section 1.1.5 below.

Licensee Event Reports

Introduction

Licensee event reports (LER) are used to inform the NRC and the industry of unanticipated issues pertaining to the plant components and systems.

The purpose of the NSA team's review of electrical system-related issues was to evaluate the ENVY's actions regarding each issue and draw potential conclusions regarding the capability of the electrical system to provide quality and reliable power to the safety-related loads.

Observations

LER 93-013-00, Switchgear Anchorage

During a Refuel Outage structural inspection, ENVY engineering observed that the safety-related 4KV switchgears and the 480 VAC load centers had not been anchored according to the existing ENVY seismic design criteria. The engineers' review determined that the event was due to inadequate instructions by the manufacturer. To address this finding, ENVY corrected the design by adding additional bolts and welding of the components to the floor frame. Their extent of condition review identified no additional structural deficiencies.

The NSA team reviewed the LER and the corrective actions and concluded that ENVY had adequately addressed the finding.

LER 94-009-00, Vital AC Bus Transfer due to Lightning Strike

A lightning strike to the plant site resulted in the failure of several components and pieces of equipment. The failure of components internal to the vital AC transfer switch caused the transfer of the vital 120/240VAC source from the normal motor-generator power source to the alternate source. The resulting momentary voltage interruption caused a number of component actuations, including the partial isolation of the primary containment isolation system. The engineering evaluation of the event found that the path taken by the lightning strike affecting the vital AC system was through a vital AC control circuit that extended to the base of the stack to control an offgas isolation valve. The evaluation resulted in several plant corrective actions, including the installation of additional lightning/surge protectors and the maintenance of the existing ones.

The NSA team reviewed the LER and the corrective actions and concluded that ENVY had adequately addressed the event.

LER 96-019-00, Loose Reactor Protection System Breaker Termination

The trip of a reactor protection system breaker resulted in a half scram and a group III containment isolation. The subsequent review of the event by ENVY found the breaker trip resulted from heat developed in the breaker from a loose termination during plant construction. Corrective actions included review and correction of termination and the replacement of the breaker itself. Corrective actions also included thermography of other terminations and the review of current preventive and corrective maintenance.

The NSA team reviewed the LER and the ensuing corrective actions and concluded that ENVY had adequately addressed the event.

LER 96-028-00, Failure to Maintain Electrical Separation Requirement

ENVY's review of findings from a design plant installation discovered that the cable routing for the actuation circuits of the redundant low pressure coolant injection (LPCI) outboard isolation valve was not in accordance with the plant separation criteria. The review also found that the non-conforming wiring was the result of inadequate wire markings and drawing update in 1976 and inadequate component labeling in 1979. Ensuing corrective action resulted in the rewiring of the valves and the initiation of a root cause analysis to evaluate the need of additional corrective actions. The LER identified no additional actions resulting from the root cause analysis. The NSA team concluded that the actions initiated by ENVY were reasonable.

LER 97-002-01, Switchgear Vulnerability to Flooding

During a 1994 design installation, a question was raised regarding the potential for water intrusion into the switchgear rooms during a flooding. The subsequent ENVY evaluation determined that, during maximum postulated flooding conditions, an underground conduit could allow water to enter the vital switchgear rooms. The evaluation addressed the mechanics of the flooding and the actions required to limit any potential flooding impact. This review indicated that the flooding of concern is 17.1 feet above the previous high water condition achieved and, therefore, a low probability event. Corrective actions included the sealing of the conduit in question with high density silicone to reduce water intrusion to a manageable level, evaluation of methods for managing the potential of flooding, and revision of plant procedure OP 3127, *Natural Phenomena*, to minimize the impact of design basis flooding. .

The NSA team's evaluation of the event and the corrective actions identified in the LER concluded that ENVY's review of the event was acceptable and that the immediate actions taken were reasonable.

LER 97-006-04, Inadequate Electrical Cable Separation

During Appendix R program enhancements, ENVY discovered an electrical cable installation that did not meet the electrical separation design criteria. Specifically, ENVY found that a non-safety-related cable supplying power to a lighting panel was routed in the vicinity of cables associated with both Division I and Division II engineered safety feature (ESF) components, thereby allowing that a failure of the non-safety-related circuit could challenge redundant ESF trains. The ENVY evaluation concluded that this was the result of inadequate installation guidelines during the initial construction of the plant. Following the initial discovery, reviews by ENVY resulted in their identifying additional cables incorrectly routed. This discovery, in turn, resulted in ENVY's performing a 100% review of the cable routing drawings and the need for addressing the approximately 60 non-conforming cables. Many condition reports (called even reports at the time) were entered into the corrective action system. Resolution of the issues resulted, as appropriate, in cables being rerouted, cables and trays being relabeled, in justifying exemptions for some cables from separation criteria, or providing clarifications in the separation data. ENVY also conducted an independent evaluation of the condition and of the reasons for the failure to discover the non-conforming conditions earlier, during plant modifications. To ensure future compliance with the separation criteria ENVY also revised applicable plant procedures.

The NSA review of ENVY's evaluation of the separation issues and of the company's corrective actions concluded that the review was acceptable and the actions reasonable. Cable separation is also addressed separately in the report.

LER 97-021-00, Division SI and SII Powered Cables in the Same Manhole

During a drawing review, ENVY discovered that energized cables from divisional raceway system traversed both Division SI and Division SII equipment. This was contrary to the plant design separation criteria. The cables in question supply power to the cooling tower fan from either Division SI motor control center (MCC) 8C or Division SII MCC 9C. The review of this issue determined that the nonconforming condition was the result of inadequate design specifications and installation during initial plant construction. Corrective actions resulted in the opening of both supply breakers and the revision of the procedure to require that both breakers be kept open. ENVY also planned to conduct an evaluation to determine the best arrangement to supply power to the fan. The NSA team's evaluation of the event and the corrective actions identified in the LER concluded that ENVY's review of the event was acceptable and that the immediate actions taken were reasonable. The LER did not identify any design changes that resulted from the subsequent analysis.

LER 97-023-00, Component Failure in Main Generator Protection

During maintenance activities in the switchyard, errors made in the motor-operated disconnect switch sequence resulted in a perturbation in the 345 KV transmission system. This perturbation, combined with a failure of a current to flow comparator/relay in the generator protection circuit, resulted in a plant trip. The ensuing evaluation by ENVY determined that the comparator was not part of the original design of the generator and that it had been added to address a cooling water failure that could go undetected. ENVY also found that the relay incorrectly utilized 28 VDC contacts in a 125 VDC circuit. However, the actual cause of the comparator failure could not be found. Corrective actions by ENVY resulted in the removal of the comparator from the circuit and a continued investigation as to the best protection for the main generator. Additionally, ENVY addressed the issue from a human performance standpoint by providing refresher training and initiating procedure improvements.

The NSA team's evaluation of the plant event and ENVY's corrective actions concluded that the follow-up and evaluation by ENVY was acceptable and the resulting actions reasonable.

LER 98-006-01, Inadequate Maintenance Procedure

Following a routine maintenance of the core spray (CS) supply breaker, ENVY discovered that the breaker closing springs had failed to charge and that the spring charging motor was running continuously. The breaker review that followed found that the closing spring charging mechanism had failed. ENVY also found that an inadequate maintenance procedure had allowed three undesirable conditions to exist simultaneously on the CS supply breaker. Specifically, maintenance troubleshooting discovered that the breaker driving pawl had become mechanically bound as a result of dried lubricant, an improperly installed bushing, and an extra washer installed between the pawl and the cutter pin. The finding resulted in ENVY inspecting all similar breakers to assure that inadequate maintenance had not allowed similar conditions to exist on other breakers. Additionally, the lessons learned from the failure were discussed with responsible personnel and maintenance procedures were appropriately revised.

The NSA team's evaluation of the plant event and ENVY's corrective actions concluded that the follow-up and evaluation by ENVY was acceptable and the resulting actions reasonable.

LER 2004-003-00, Isophase Bus Duct Two-Phase Fault

An electrical fault in the isophase bus duct resulted in a main generator load reject and the scram of the plant. The ensuing evaluation by ENVY found that the electrical fault was the result of inadequate preventive maintenance. Specifically, ENVY's investigation found that loose material, due to a failed flexible connector in the "B" isophase bus duct caused grounds to occur. These grounds raised the voltage on the "A" isophase bus duct which, in turn, caused a surge arrester to fail. Entergy noted that, although the isophase bus duct underwent preventive maintenance each Refuel Outage, the cleaning and the inspection was limited to the standoff insulator. They concluded, therefore, that a more thorough inspection that included the bus duct and the flexible connectors would have observed the loose material and corrected the deficiency. To address this issue, besides replacing and inspecting the bus duct, ENVY revised an alarm procedure and the applicable preventive maintenance procedures.

The NSA team's evaluation of the plant event and ENVY corrective actions concluded that the follow-up and evaluation by ENVY was acceptable and the resulting actions reasonable.

LER 2005-001-00, Electrical Insulator Failure in the 345KV Switchyard

An electrical transient that initiated in the 345KV switchyard resulted in a generator load reject trip and the consequent reactor trip. ENVY's investigation found that the transient in the switchyard was caused by a failed porcelain electrical insulator in a motor-operated disconnect switch. A laboratory analysis of the insulator found that the failure was caused by a manufacturing defect. The laboratory specifically found a void area in the cement that attached the failed section of the insulator to the metal flanges and a geometric offset in the placement of the insulator in the flanges. These conditions, aggravated by wind loading in a certain direction caused a stress fracture to develop in the insulator. This stress fracture eventually caused the insulator to fail. Corrective actions included thermography examinations of other components in the 345KV and 115KV switchyards, scheduling of inspection of similar insulators, and revision to the preventive maintenance frequency for switchyard components.

The NSA team's evaluation of the plant event and ENVY's corrective actions concluded that the follow-up and evaluation by ENVY was acceptable and the resulting actions reasonable.

Conclusions

As outlined above, the licensee event reports address a variety of issues. Most of these issues could be traced to inadequate controls existing at the time of the plant construction. Three of the issues were directly or indirectly related to inadequate maintenance activities.

The NSA team's evaluation of the ENVY actions that followed each event concluded that such actions were generally thorough and that they adequately addressed the root cause of the event. Also, the operability evaluation of the equipment affected by the event was reasonable and consistent with the importance of the equipment.

Miscellaneous Electrical Issues

Introduction

The review of the electrical system by the EDSFI and CDBI teams identified additional issues that provided information regarding the capability and reliability of the electrical system and components, in general. The purpose of the NSA team was to review these electrical system-related issues, evaluate the ENVY's actions regarding each issue, and, based on its review, draw conclusions regarding the capability of the electrical system to provide quality and reliable power to the safety-related loads.

Observations

Item No. 92-81-02, Inadequate Breaker Test Control

The lack of a program to test periodically the functionality of the 480 V molded case circuit breakers was considered by the EDSFI team to be in Violation of NRC requirements.

To address this issue ENVY provided additional test information showing that testing was, in fact, being conducted in accordance with NEMA and EPRI recommended practices. Based on the information provided, the NRC withdrew this violation.

Based on NRC letter NVY 94-97, dated May 24, 1994, data provided by ENVY indicated that of 34 breakers tested, 10 breakers had failed to perform as expected; however, all the failed breakers were installed spares. Therefore, their failure was considered to be of no significance because they would not have been used without prior testing and calibration.

Since the spare breakers were seldom tested, in their evaluation of this item, the NSA team asked the ENVY cognizant personnel whether the spare breakers could be substituted for active breakers if needed. ENVY stated that any application requiring the utilization of installed spare breakers would be governed by a maintenance work order, and if required, a design change, therefore, casual substitution of spare breakers was not permissible by procedure. Based on the evaluation of this item, the NSA team concluded that ENVY's resolution of the issue was reasonable. No additional areas of concern were identified by the NSA team regarding molded case breaker testing.

Item No. 92-81-03, Inadequate Fuse Control

This concern related to the lack of a program for controlling fuse replacements. The issue was categorized by the NRC as a Violation of their requirements.

To address this issue, ENVY provided the NRC with additional information showing that an adequate fuse control program did exist at the plant. Based on this new information, the NRC withdrew the Violation.

The NSA team review of this item concluded that sufficient justification had been provided regarding the adequacy of the ENVY fuse control program. This review identified no additional areas of concern with fuse control.

Item No. 92-81-04, Separation from Preferred Power Source

The EDSFI team's review of the degraded grid relays setting concluded that, if the plant buses were operating at maximum load while being supplied from the switchyard (offsite) source and the switchyard voltage was at the minimum design limit of 113 KV, an accident signal could result in the degraded grid relays actuating and cause separation of the plant buses from the offsite (preferred) source and the loading of the buses on the standby diesel generators. The team observed that such separation could also occur if the reactor feed pump started during a design basis accident. The NRC identified this issue as an Unresolved Item.

ENVY's review of this finding resulted in its committing to install a temporary modification that would cause the cooling tower loads to trip in the event of a LOCA signal. This action would lower the maximum bus load and improve the bus voltage, such that separation of the switchyard source would no longer occur. Subsequently, ENVY installed a permanent modification. This modification replaced the existing nonadjustable degraded grid relays with higher accuracy adjustable relays. These relays had a reset voltage of 1% rather than the 3% in the original relays. The basis for the relay change was that the use of a lower reset percentage would allow the relays to reset earlier and, thus, prevent separation of the buses from the switchyard source. Additionally, ENVY installed a design modification that would prevent the diesel generator breaker from closing onto a live bus.

The NSA team evaluation of this issue concluded that the design changes installed by ENVY were reasonable and that they adequately resolved the potential bus separation concerns. No additional areas of concern were identified during this review.

Item No. 92-81-06, 120 V AC Protective Devices Coordination

The EDSFI team's review of the circuit breaker coordination study found that, for the 120 V AC safety-related loads, the study was limited in scope and did not adequately show that the loads were properly protected. The NRC identified this issue as an Unresolved Item.

As a follow-up to the inspection, ENVY developed a new calculation (VYC-1247). This new calculation demonstrated that adequate coordination existed for 120 V AC protective devices, that the protective devices were selectively coordinated, and that the safety-related circuits were protected from overloads and short circuits. The NRC subsequent review of this calculation resulted in their closing the item.

The NSA team's review of this item identified no additional areas of concern and concluded that the issue had been adequately addressed and resolved.

Item No. 92-81-07, 120 V DC Protective Devices Coordination

The EDSFI team found that the circuit breaker coordination study for the 120 V DC safety-related circuits was informal, limited in scope, and did not include many of the circuit breakers. The sample review of the circuit breakers included in the study showed that some areas existed where coordination was limited. The NRC identified this issue as an Unresolved Item.

Using a new calculation (VYC-1188), ENVY showed that coordination of the 120 V DC protective devices was generally acceptable. ENVY also showed that, with the addition of some recommended enhancements, the protective devices were selectively coordinated and the safety-related circuits were adequately protected from overloads and short circuits.

The NSA team evaluation of this item concluded that the issue had been adequately addressed by ENVY and appropriately resolved. The team identified no additional areas of concern.

Item No. 04-08-03, Degraded Relay Setpoint Calculations

The CDBI team evaluated the degraded grid relay setpoints to confirm that adequate voltage was available to the safety-related components such that they would be able to perform their safety functions under worst grid voltage condition. This review found that the Technical Specifications allowed a minimum voltage of 3660 VAC at safety-related buses 3 and 4, whereas the analysis performed by ENVY was based on the minimum anticipated voltage at the grid, which translated to 3951 Volts and 3809 Volts at Buses 3 and 4, respectively. Because the voltage study calculation, VYC-1088 was not conservative, the team concluded that two additional calculations that established the voltage available at the safety-related motor control centers and at the motor operated valves were similarly not conservative. Based on the justifications provided by VY, the NRC concluded that the issue had very minor safety significance and considered the finding a non-cited violation.

As a result of the NRC finding, ENVY initiated two condition reports. In the resulting operability evaluation, ENVY indicated that it had not prepared a calculation that verified acceptability of available voltage at the various components if the voltage at the 4 KV buses dropped to the TS minimum level. An analysis of component voltages, assuming minimum TS bus voltage identified two motors that would have been operating at less than 90% of their rated voltage. The evaluation concluded that the voltage was acceptable. Voltages at motor control centers were considered to be equally acceptable. The calculations were corrected to address the NRC finding.

As indicated in Section 1.1.4, above, the NRC observed that procedure ON 3155 allowed operation of the plant with a minimum bus voltage of 3600V, and, therefore, below the minimum voltage allowed by the technical specification, 3660V. Subsequently, ENVY changed the procedure value to 3700V. This change was based on the TS value (3660V) adjusted for instrument error (40V)

The NSA team's evaluation of the degraded grid relay setpoint issue, determined that reasonable steps were taken by ENVY to address the NRC findings. In this case, to assure adequate voltage to the components, they initiated separation of the buses from the offsite source at the minimum anticipate grid voltage. ENVY's evaluation of the available voltages at the various components if the bus voltage dropped to the TS stated minimum value confirmed that the TS value was acceptable. The NSA team concluded that the bus voltage available from the offsite source was acceptable and that the safety-related loads would receive sufficient voltage to perform their safety functions.

Item No. 04-08-04, Ungrounded 480 VAC Electrical System

During their review of the electrical system the CDBI team observed that the two 480 VAC safety-related load centers were operating ungrounded, i.e., their source of power was derived through a transformer connected in a delta-delta arrangement. The team also observed that these buses powered some non-safety-related loads. Therefore they expressed a concern that, during a design basis event, this unprotected equipment could develop intermittent/arcing ground faults that could cause excessive voltages to be impressed on the 480V electrical system. In this case, the protective devices potentially would not be able to isolate the faults and, therefore, would allow damage to the bus powered equipment to occur.

This issue was referred to the NRC office of Nuclear Reactor Regulation (NRR) for review. The NRR assessment concluded that ENVY was in compliance with its licensing basis and that their design criteria were in conformance with industry design criteria. NRR considered the issue not risk significant because "(a) the likelihood of an arcing/intermittent ground fault occurring during a high energy line break or seismic event was low, (b) the occurrence of the arcing ground fault at an intermittent frequency sufficient to produce excessive voltage was more unlikely, and (c) the occurrence of a second arcing ground fault at the same time on the redundant and independent 480 VAC system that also occurred with an intermittent frequency sufficient to produce excessive voltage was not credible." Based on the NRR review, this item was closed without action.

The NSA team's assessment of this issue concluded that the NRR conclusions were reasonable and that the item was appropriately closed.

Conclusions

As in the case of the LERs, the above NRC findings pertain to a variety of engineering issues. The most notable of these issues were the ones related to the degraded grid relay settings because they assure that the integrity of the system is maintained and that adequate voltage is provided to the safety-related components.

The NSA team's review of the above issues found that ENVY's actions that followed the identification of those issues were acceptable and that, based on this review, the reliability of the affected components was maintained.

General Conclusions

The issues identified by the NRC during the electrical distribution system functional inspection (EDSFI) in 1992 and in the component design basis inspections (CDBIs) in 2004 and 2008 affected on-site and off-site sources of the plant electrical power system, as well as the components and programs related to such sources. Additional data points regarding these sources were provided by the LERs related to the electrical system.

The NSA evaluation of these issues found that the licensee's evaluations were thorough and in conformance with the significance of the issue. The review of the condition reports associated with these issues found that, when appropriate, they included root or apparent cause evaluations, that the ensuing corrective actions were generally appropriate and effective, and that, when applicable, extent of condition was also evaluated. Based on the scope of the above review, the NSA team also concluded that reasonable assurance exists regarding the capability of the AC and DC electrical systems at ENVY to provide quality and reliable power to the ENVY safety-related loads.

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Procedures and Specifications

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2. OP-2126 Rev. 55, Diesel Generators
3. OP-3127 Rev 25, Natural Phenomena
4. OP-4126 Rev. 82, Diesel Generators Surveillance
5. OP-4209 Rev. 10, UPS Battery Performance Test
6. OP-4215 Rev. 27, Main Station Battery Performance/Service Test
7. OP-4217, Rev. 11, Alternate Shutdown Battery AS-2 Service/Performance Test
8. OT-3122 Rev 21, Loss of Normal Power
9. Lesson Plan LOT-00-602 Rev. 25, Operational Transient (OT) Procedures
10. Lesson Plan LOT-00-604 Rev. 18, Operational Transient (OT) Procedures - Simulator

Drawings

1. G-191372, Sheet 1, Rev. 67, 125 V DC One Line Wiring Diagram
2. G-191372 Sheet 2, Rev. 24, 125 V DC One Line Diagram
3. G-191372 Sheet 1, Rev. 18, 125 V DC One Line Diagram
4. B191301, Sheet 1362, Rev. 17, CWD Turbine Bldg. Steam Heater Fans
5. B191301, Sheet 1323, Rev. 7, CWD Heating & Ventilation Annunciator Connections

Documents

1. VYNPS UFSAR Rev. 22
2. Document 24DC Rev. 16, Design Basis Document for 24 Vdc System – ECCS and Neutron Monitoring
3. Document 125DC Rev. 23, Design Basis Document for 125 VDC Systems
4. NRC Letter, NVY 91-98, dated 6/5/1991, Vermont Yankee Station Blackout Analysis
5. NRC Letter, NVY 92-184, dated 9/30/92, Electrical Distribution System Functional Inspection of Vermont Yankee, Report No 50-271/92-81
6. NRC Letter, NVY 93-015, dated 1/27/93,
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10. NRC Letter, NVY 94-097, dated 05/24/94, “NRC Inspection Report 50-271/94-05”
11. NRC Letter, NVY 95-020, dated 01/30/95, “NRC Inspection Report 50-271/94-30”
12. NRC Letter, NVY 95-148, dated 10/17/95, “NRC Inspection Report 50-271/95-15”
13. NRC Letter, dated 10/19/2005, “NRC Integrated Inspection Report 05000271/2005004”
14. NRC Letter, NVY 04-130, dated 12/02/2004, “Vermont Yankee Nuclear Power Station NRC Inspection Report 05000271/2004008”
15. NRC Letter, NVY 06-010, dated 2/1/2006, “NRC Integrated Inspection Report 05000271/2005005”
16. NRC Letter, dated 10/20/2005, “Vermont Yankee Nuclear Power Station Problem Identification and Resolution NRC Inspection Report 05000271/2005006”
17. NRC Letter, dated 9/27/2006, “Vermont Yankee Nuclear Power Station NRC Component Design Basis Inspection Report 05000271/2006007”
18. NRC Letter, NVY 08-096, dated 09/26/2008, “NRC Component Design Basis Inspection Report 05000271/2008008”
19. VYNPC letter, BVY 92-124, dated 10/30/92, “Response to NRC Electrical Distribution System Functional Inspection of Vermont Yankee, Report No. 50-271/92-81, Reply of a Notice of Violation”
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21. VYNPC letter, BVY 08-043, dated 8/14/2008, “Update of Vermont Yankee Program Commitments Relative to Periodic Testing of Alternate AC Source”
22. Engineering Report VY-RPT-05-00004, Rev. 0, “VYNPS EPU SBO Coping Analysis Report”
23. Calculation VYC-836, Rev. 13, “Diesel Generator Loading”
24. Calculation VYC-1188, Rev. 1, “125 VDC Relay and Circuit Breaker Coordination”
25. Calculation VYC-1053, Rev. 8, “MOV Voltage Analysis”
26. Calculation VYC-1088, Rev. 3, “Vermont Yankee 4160/480 Volt Short Circuit/ Voltage Study”
27. Calculation VYC-1247, Rev. 5, “120VAC Circuit Breaker Coordination”
28. Calculation VYC-1314, Rev. 3, “Determination of Minimum Pickup Voltage for SCE MCC Control Circuit Devices during Normal Operation Transient Conditions”
29. Calculation VYC-1502, Rev. 0, “Control Room Heatup due to Loss of HVAC”

30. Calculation VYC-2133, Rev. 0, "Fuel Oil Storage Tank (FOST) Post Accident Minimum Temperature Evaluation"
31. Calculation VYC-2153, Rev. 0, "125 VDC Battery A-1Electrical System Calculation"
32. Calculation VYC-2154, Rev. 0, "125 VDC Battery B-1Electrical System Calculation"
33. Calculation VYC-2175, Rev. 0, "Fuel Oil Storage Tank External Piping Analysis Post Loss of Power (LNP)"
34. LER-93-013-00, "Safety Related Switchgear Found to Have Less Anchorage than Required by the Original Plant Design," dated 10/18/1993
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42. LER-97-006-04, "Use of Inadequate Design Specification/Implementation Document during Initial Plant Construction Results in the Failure to Maintain Proper Electrical Separation of Electrical Cables," dated 3/19/1998.
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48. LER-98-21-00, “Inadequate Licensing Basis Documentation Retrievability Results in the Failure to Meet IST Requirements for Diesel Fuel Oil Day Tank Level Control Valves,” dated September 1, 1998.
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2.8 Primary Containment System [3(a)(4)]

Introduction

As part of the reliability assessment of the ENVY, the structural integrity and maintenance of Primary Containment was identified as area that was included in the technical assessment of the station.

Technical Area Description

The ENVY Primary Containment is a General Electric Mark-I Pressure Suppression Containment System consisting of the drywell, the pressure suppression chamber (torus), the connecting vent system between the drywell and the torus, and isolation valves. The code of construction for the primary containment structure is the ASME Section III – 1965 Edition with the Winter Addenda. Primary containment consists of a number of systems and components directly related to the safe operation of the plant. The technical areas considered in the scope of this assessment are:

- Drywell Shell
- Torus Shell
- Torus Issues Identified at Fitzpatrick Generating Station
- Isolation Valves, Type A, B and C Testing

An important part of the ongoing operation and maintenance of the primary containment structure is an in-service inspection program that systematically examines and evaluates critical features of the structure. The code of record for the ENVY in-service inspection program is the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for In-service Inspection of Nuclear Power Plant Components" (ASME Section XI – 1998 Edition with 2000 Addenda). The code of record was modified by the adoption of subarticle IWA-4530 of the 2003 Addenda of the ASME Section XI – 2001 Edition. This adoption was approved by an NRC Safety Evaluation (NVY 05-094) dated July 25, 2005.

Another important feature of the primary containment is the ability to isolate the internal containment atmosphere in order to mitigate the resultant loads from any accident scenarios and to limit the release of any fission products to the environment. This is done through a series of isolation valves and penetrations that are designed and constructed to withstand the potential post-accident pressures developed inside containment.

Methodology

This assessment was performed through a variety of methods including: document reviews, site visits, plant walk downs, interviews with plant personnel, and comparison of features at ENVY with generally acceptable programs and practices in the nuclear power industry. Plant walk downs included visual observation of the exterior areas of the drywell and the torus where normally accessible. The assessment methodology was targeted at reviewing the technical areas of the scope that provided insight into long term plant reliability.

Interviews were conducted with the engineers most responsible for the primary containment issues associated with this technical area. The interviews were intended to question the plant personnel on the past, present and future status and maintenance of some of the critical parameters of primary containment and to discuss long term industry issues associated with primary containment.

The observations that follow are structured in alignment with the technical areas considered in the scope of this review and are based on the information gathered at the site during the assessment period.

Drywell Shell Observations

The drywell is that portion of primary containment that contains the reactor vessel, residual heat removal (RHR) pumps, inboard main steam isolation valves (MSIVs) and piping and cooling systems to support reactor operation. The drywell shell is constructed of carbon steel and was originally designed and fabricated to General Electric Specification 21A5837. Piping and electrical penetrations penetrate the drywell shell in order to provide the necessary process piping and electrical power and instrumentation to the systems inside the drywell. Additional penetrations in the form of equipment and personnel access hatches also form part of the containment boundary.

The drywell shell is constructed such that it rests on a large concrete pedestal with a sand cushion interface between the lower portion of the drywell steel shell and the concrete pedestal. As the lower portion of the spherical drywell shell rises up to the approximate elevation of the vent system piping, the sand cushion terminates. From that point to the upper portions of the drywell shell, there is an air gap between the drywell steel shell and the concrete shield enclosure surrounding the drywell. This gap is approximately 1” to 2” in width.

The interior side of the drywell shell is coated to protect the steel shell and to provide a clean surface for the interior of the drywell. Visual inspections of accessible portions of the interior of primary containment are performed during Refuel Outages on a scheduled basis in accordance with the Vermont Yankee In-service Inspection Program Section SEP-CI-001. These visual inspections classify and evaluate the condition of the surface coatings. The condition of the coating provides an indication of the condition of the base metal substrate. For these inspections, ENVY has developed screening criteria to efficiently disposition a range of potential findings that may be discovered during an inspection.

The surface defects can be generally classified as pits, gouges, arc strikes, or cracking. The inspection criteria are applied to all visually accessible areas of the drywell, torus shell plates and the downcomer and vent region. The technical bases of the acceptance criteria for the inspection results are documented in calculation VYC-2043, Rev. 0, *Evaluation of Primary Containment Localized Thinning and Screening Criteria for ASME XI IWE Inspections* and calculation VYC-1032, Rev. 1, *Torus Shell and Vent System Thickness Requirements.*

The ASME IWE General Visual Containment Inspection is contained in Entergy engineering standard ENN-EP-S-001. These inspections are performed by a certified Level III Examiner qualified to the requirements of ASME Section XI 1992 Edition, 1992 Addenda, or 1998 Edition, no addenda, Subsection IWE, Subparagraph IWE-3510.1. These inspections are performed in accordance with the frequency schedules contained in SEP-CI-001. During the inspection, the examiner visually evaluates the interior surfaces of the drywell shell, all accessible interior torus surfaces above the water line and numerous containment penetrations. The acceptance criteria of the visual examinations of each area inspected are based on pre-determined values in the standard. The surface areas are checked for pits, gouges, metal cracking, metal corrosion or discoloration, coating blistering, coating flaking and rusting or staining. Minor anomalies are noted throughout the inspection checklist forms and identified conditions that do not meet the acceptance criteria are documented and have either further engineering analysis performed or are repaired.

The inspection reports reviewed showed that thorough inspections have been performed on all of the required areas and components. The interior of the containment liner plate, all accessible penetrations and all 16 torus bays were visually inspected. In one instance, the torus visual inspection required that further ultrasonic measurements be taken. The ultrasonic measurements were subsequently taken and found to be acceptable.

The results of the visual inspections from RFO 24 (April 2004) were reviewed and found to meet the in-service inspection criteria stipulated in the criteria and procedures. The results from RFO 27 (October 2008) were not as yet available to the review team, however, the ENVY personnel interviewed stated that there were no unacceptable conditions or adverse trends identified during this most recent inspection.

Another aspect of the containment steel shell is attachments that are welded to the interior or exterior shell wall. Most of the primary containment welded attachments were installed at the time of original construction. Welded attachments include pipe supports, penetrations through the shell, conduit and electrical raceway supports and miscellaneous attachments for cable, lighting, ventilation, etc. During the changes required for the Mark I Containment Modifications performed in the late 1970's, additional structural attachments were made in the torus to accommodate the steam relief valve quenchers and to provide additional support for the downcomer vent header.

In discussions with plant personnel, there have been no welded attachments performed in containment over the last several years. At original installation, welding on any portions of the drywell or torus shell required a qualified welder with appropriate post-installation inspection criteria. Although there is no formal ongoing inspection program dedicated to welds, once installed and accepted, the inspection of welded attachments becomes part of the IWE General Visual Containment Inspection.

In December of 1986, the NRC issued Information Notice 86-99 to provide licensees with information on a potentially significant safety problem regarding the degradation of a steel containment resulting from corrosion. Another nuclear generating station had discovered water in the gap between the containment drywell steel shell and its outer concrete shield. The cause of the water intrusion was determined to be a leak in the bellows at the drywell to reactor cavity seal. The bellows was subsequently repaired and the water intrusion ceased. To determine if the water in the gap had caused damage to the steel containment, the plant measured the existing steel containment wall thickness using ultrasonic testing techniques. The testing revealed an apparent loss of some thickness on the gap side of the steel shell immediately above the concrete pedestal upon which the drywell rests. In this area, the gap is packed with sand and contains drain piping to alleviate any potential water accumulation. This plant initiated a remediation program to identify and repair any portions of the steel containment in the sand cushion area that were below acceptable design requirements.

This NRC Information Notice served to alert all nuclear power plants with similar design features of the potential sand cushion corrosion issue. ENVY has a similar design feature as identified in the NRC Information Notice. ENVY has 8 drywell sand cushion drain lines extending radially out from the bottom of the drywell (under the vent header assemblies) and terminating at the drywell concrete pedestal wall at about elevation 231'. The drain lines provide a run-off pathway if water is introduced into the air gap surrounding the drywell shell, thereby minimizing the effect of leakage and corrosion on the external drywell shell. These drains are examined in accordance with the Containment In-service Inspection Program and are credited as part of the Aging Management Program – Containment In-service and Containment Leak Rate Program for License Renewal but are not ASTM or Regulatory required.

These drain lines are internally examined for freedom of obstructions and integrity up to the fine mesh screen at the point of intersection of the drywell shell and the sand cushion. These examinations are performed per the Containment In-service Inspection Program Section SEP-CI-001. The results of the drain line inspections performed in February of 2007 were reviewed. The examinations were performed from the torus room and use a boroscope to visually inspect the drain lines. Although the examination of the 8 drain lines noted some debris at the end of drain line and some kinking or bending of the PVC pipe, the results were evaluated and deemed acceptable since none of the existing conditions on the drain lines would adversely impact their ability to function to dewater any accumulated water at the sand cushion drywell gap interface. It is important that the examination of the drain lines continue for the operating life of ENVY and that any adverse trends are addressed and corrected.

Another area of industry concern has been on the interior side of containment on the lower portions of the drywell shell. Since the steel surface of the bottom of the drywell shell is spherical, the bottom several feet on the interior of the drywell are filled with reinforced concrete to provide a level floor area for the base of drywell. It has been identified in the industry that moisture and/or water running down the interior side of the drywell shell has infiltrated the area between the concrete floor and the steel containment shell and has created the potential for corrosion of the steel shell in this area.

ENVY performed Technical Evaluation No. 2000-011 to address the potential corrosion issues on the drywell subsurface area. The evaluation was reviewed and was found to adequately address the condition of this inaccessible area per the requirements of 10 CFR 50.55a (x) (A). Additionally, ultrasonic testing (UT) measurements were taken around the band where the drywell steel meets the moisture barrier. The UT measurements indicated that there was no unacceptable loss of metal in this area. The moisture barrier on the interior of the drywell was replaced at ENVY in 2001 via MM-2000-010. The area of the moisture barrier to drywell shell interface is also visually inspected as part of the IWE general visual containment inspection performed by procedure ENN-EP-S-001.

Based on the areas reviewed and the data received for review, the current condition and the programmatic controls and inspections performed for the primary containment shell are generally consistent with industry practices and the results meet industry standards.

Torus Shell Observations

The torus is a circular (toroidal) structure that is nominally half-filled with water to cool and suppress any excess energy generated during the steam production process. The torus at ENVY is approximately 28' in cross-sectional diameter with a horizontal interior diameter of approximately 70' and a horizontal exterior diameter of 126'.

The torus shell at ENVY is nominally 0.584" thick at the bottom of the cylindrical shape. There is very little excess design margin with regards to the thickness of the shell. The required thickness of the torus shell to maintain the resultant loads from various accident scenarios has been evaluated to assure that the required shell thickness exists. A thorough evaluation of the torus shell, vent system and other internal components of the torus have been evaluated in calculation VYC-1032, Rev. 1, *Torus Shell and Vent System Thickness Requirements*. This calculation also serves as the basis for ongoing torus shell and torus component inspections as required by the in-service inspection program.

At the time of original plant construction, the lower half of the interior of the torus shell was coated in order to provide a protective layer for the base carbon steel shell. Industry experience has shown that the coatings in constant contact with water on torus shells deteriorate over long periods of time such that the coatings may no longer be able to provide the full protective measures for which they were designed. The torus has been a constant focus of attention in nuclear plants that employ such a structure. Only the GE Mark 1 BWR Containments use the torus structure as the suppression chamber for various reactor transients.

In October of 1988, the NRC issued Information Notice No. 88-82 to alert licensees of degraded coatings in BWR containments. The Information Notice stated that the coating degradation had no immediate impact on plant operation but cautioned that coating degradation could potentially lead to torus shell degradation which could jeopardize containment integrity. In 1998, ENVY undertook a torus recoating project. This is a significant project since it involves de-watering the torus, cleaning it, sand blasting the shell down to base metal, surface preparation, recoating and curing. This effort was performed during the 1998 Refuel Outage.

While the sand blasting portion of this project removes the remnants of the initial coating, it must be performed carefully such that it does not remove the base metal. Therefore, in order to assure that the required base metal design thickness was retained, the torus shell thickness was verified at this time after the sand blasting had been performed. Torus shell thickness measurements were taken at numerous locations prior to the recoating of the torus by using ultrasonic testing techniques. These measurements showed that the required base metal thickness of the torus shell was acceptable and that any areas at or below minimum wall thickness requirements were evaluated by design engineering.

To further monitor shell thickness and corrosion, the lower half of the torus shell was mapped at various locations so that repeat inspections and UT measurements could be taken over time and trending could be evaluated. The UT measurements are performed in a systematic process in order to gauge material thickness and any corrosion. This process can provide information regarding corrosion rates and assist in the early identification of potential thickness reduction problems. The ongoing inspection and measurement of the torus shell is performed in accordance with the Containment In-service Inspection Program Section SEP-CI-001.

A Memorandum of Understanding (MOU) was entered into by the Vermont Department of Public Service (DPS) and Entergy VY. A portion of the MOU stipulates that:

> *"During and following the completion of the Refuel Outages currently scheduled for 2007 and 2008, Entergy VY will perform detailed visual inspections of the torus to confirm that there are no potential leakage paths. The inspections will look specifically at work that was performed during the outage that may have had contact with the torus. Entergy VY will consult with the DPS in developing any new inspection procedures, and any new revisions thereto, for conducting such visual inspections. Entergy VY will perform daily operator rounds in accessible areas of the torus to identify any potential leakage paths".*

In a letter dated June 29, 2007, ENVY informed the DPS of the results of the torus inspection performed during the 2007 outage and stated that the results showed the torus to be in very good condition with no identified potential leakage paths. Interviews with plant personnel indicated that the results of the visual inspection of the torus for the 2008 outage were also acceptable. The documented results were not yet available at the time of this writing but will be forwarded to DPS per the requirements of the MOU.

With the torus recoating project of 1998, the data collected at the time of that project and the ongoing long term inspection and maintenance of the torus, it is expected that the torus shell should be able to perform its intended design functions into the future. Due to the critical nature of this structure and its nominal shell thickness design margin, continuous monitoring and inspections are essential to the long term health of this structure. Also, ENVY needs to continue to participate in the BWR Mark I Owners Group in which participating licensees share information regarding technical issues associated with significant industry containment issues. It should be noted that not all of the Mark I BWRs of the ENVY vintage have embarked on a torus recoating effort.

Based on the areas reviewed and the data received for review, the current condition and the programmatic controls and inspections performed for the torus shell are generally consistent with industry practices and the results meet industry standards.

Torus Issues Identified at Fitzpatrick Generating Station

In June of 2005, personnel at the James A. Fitzpatrick Nuclear Power Plant discovered a torus leak near a torus support. The leak was located about 5 feet below the waterline and just below the high-pressure coolant injection (HPCI) turbine exhaust pipe. The leak was characterized as a slight seepage with streaking and a small puddle below the leak. Subsequent nondestructive examination determined that the leakage was from a small through-wall torus crack, which was "X"-shaped with an approximate 4.6 inch maximum length. Upon identification of this condition, the reactor was shut down.

To correct this condition, the licensee installed an approximately 13 inch outer diameter torus repair plate with a full-penetration weld joint. Pressure testing and inspection of the torus and drywell were completed after the repairs were completed. An NRC special inspection team reviewed the licensee's repair methods, root cause and extent-of-condition determinations, and corrective actions before the reactor was restarted.

The licensee performed a root cause investigation of the event, and after eliminating a number of possible causes (thermal fatigue, clearing load phenomena, metallurgical discontinuity, weld defects, corrosion, flow-induced phenomena, flow-accelerated corrosion, cavitation, and direct jet impingement), the licensee concluded that the most likely cause for the initiation and propagation of the crack was the hydrodynamic loads of the turbine exhaust pipe during HPCI operation coupled with the highly restrained condition of the torus shell at the torus column support. The cracking occurred in the heat-affected zone of the lower gusset plate of the ring girder at the torus column support

The licensee concluded that most probable cause of the crack was fatigue fracture resulting from cyclic stressing due to condensation oscillation during HPCI operation. Condensation oscillation induced fatigue occurs when cyclic pressure induces alternating stresses. This would induce cyclic stresses in the area of the torus shell localized at the HPCI exhaust line termination point. This area is highly restrained by support welds and ring girder welds. These condensation oscillations induced on the torus shell may have been excessive due to a lack of an HPCI turbine exhaust pipe sparger that many licensees have installed. A sparger at the end of a high pressure discharge line tends to dissipate the energy released by the high pressure exhaust steam and distribute it over a large area so as not to locally stress a specific area.

In response to this industry identified issue, ENVY conducted its own investigation to determine if a similar condition could exist in its torus shell or at the HPCI turbine exhaust discharge point. ENVY reviewed the HPCI and RCIC steam exhaust lines within the torus. Unlike the Fitzpatrick Station, the ENVY's HPCI and RCIC discharge piping extend through a 60^{o} elbow and exit approximately 9 ft. below the water surface for HPCI and approximately 8 ft. below the water surface for RCIC. The HPCI discharge piping has a sparger at the exit point with none provided for RCIC. Additionally, during RFO 24, visual examination was performed on the torus external surface and on the internal surface above the water line and as far below the water line as possible. Ultrasonic measurement examination was performed in all areas where coating issues were identified. No concerns were identified in the HPCI and RCIC areas.

Based on the review of ENVY's response to this issue and the actions taken to assure that the problem discovered at Fitzpatrick has been adequately assessed, ENVY's approach to the resolution of this issue is consistent with industry practices and the results meet industry standards.

Isolation Valves Observations

One of the dominant design features of primary containment is its ability to contain the pressures and loads resulting from the various accident scenarios that may occur inside containment. In order to assure that the integrity of primary containment is maintained, systematic testing is performed on the containment structure as a whole as well as on its specific pressure boundary components. These components consist of mechanical, electrical and piping penetrations in the drywell and torus steel shells in order that electrical power, instrumentation, water and steam can enter/exit the reactor vessel and its supporting systems. Additionally, there are personnel and equipment access hatches that form part of the containment boundary. For the integrity of primary containment, these penetrations must withstand their respective design loads and pressures such that any postulated accident scenario can be isolated and contained within the primary containment structure.

To demonstrate the satisfactory performance of the primary containment structure as well as its respective isolation valves and penetrations, testing is periodically performed to assure that primary containment integrity is maintained. This testing is typically referred to as Type A, Type B and Type C leakage rate testing. The overall implementing programmatic procedure for the ENVY Primary Containment Leakage Rate Testing (Appendix J) is SEP-APJ-009. The governing Entergy fleet procedure for leakage rate testing is EN-DC-334.

Type A testing is the Integrated Leakage Rate Test (ILRT) and is described in the Vermont Yankee UFSAR in Section 5.2. ENVY's Technical Specifications (Section 6.7.C) specifies that the primary containment leakage rate testing program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, entitled *Performance Based Containment Leak-Test Program*, dated September 1995. Regulatory Guide 1.163 specifies that NEI 94-01, Rev. 0, *Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J* provides methods acceptable to the NRC for complying with the provisions of Option B (performance based option). The Type A ILRT provides assurance that the primary containment structure and its various components will perform its design function following a design basis accident.

The Type A ILRT was last performed at ENVY in 1995. Typically, the next scheduled Type A ILRT would be required to be performed at a ten year interval from 1995, which would mean the next test would take place in 2005. However, ENVY requested an extension of the 10 year period to a 15 year period. The NRC granted this extension in a SER contained in NVY 05-108 dated August 31, 2005 and Technical Specification Amendment 227 provides for a one time basis extension to the Type A test interval from 10 years to 15 years. Therefore, the next scheduled Type A ILRT is now scheduled for RFO 28 which is planned for the spring of 2010.

The results of the 1995 Type A ILRT and verification test were reviewed and the testing was considered successful. The summary of the 1995 Type A ILRT was documented in a report prepared by General Physics Corporation dated July, 1995. The report stated no specific corrective actions planned since the as-found and as-left test data indicated that successful testing was completed during this test.

The MOU previously mentioned in this section provided for some information regarding integrated leakage rate testing. A portion of the MOU stipulates that:

> *"Entergy VY shall perform a Type A Containment Leak Rate Test (the "Type A Test") during the station's Refuel Outage in 2010 (the "Outage") and make the results available to DPS within 60 days of the Outage's completion".*

It is expected that ENVY will perform this test and supply the results to DPS as stipulated above. Any anomalies or discrepancies noted during this test should be reviewed by the DPS.

Type B and C testing are referred to as Local Leakage Rate Testing (LLRT) since they examine specific components that penetrate the containment shell. Type B LLRTs are performed on containment penetration seals and Type C LLRTs are performed for all containment isolation valves. Type B and C local leakage rate testing is performed prior to working on or accessing a component (as-found condition) and also performed after the component has been worked on or secured for continued use (as-left condition). The as-found condition provides indication of how well the component has performed since its last LLRT and the as-left condition provides the basis that the component will perform its intended design function until the next LLRT is performed. Over time, the LLRT program develops a performance based type of history for each penetration and valve in the LLRT program. This enables the program manager to effectively monitor and test the penetrations and/or valves in a systematic manner. Unlike the Type A ILRT, the Type B and C LLRTs are performed on a more frequent basis.

Regarding local leakage rate testing, the MOU previously mentioned stipulates that:

> *"Following the completion of the refueling outrages currently scheduled for 2007 and 2008, Entergy VY will provide DPS with a summary of the results of all primary containment leakage tests performed during those outages"*

In a letter June 29, 2007, ENVY informed the DPS of the results of the leakage rate testing and inspections during the 2007 Refuel Outage (RFO 26). The results of the testing showed that all leakage rate testing was completed with acceptable results and no areas of concern were identified. An advance copy of the leakage rate testing results performed during the 2008 Refuel Outage (RFO 27) were also reviewed and compared with the 2007 test results.

The LLRTs performed during the 2007 Refuel Outage included approximately 87 penetrations and valves. The LLRTs performed during the 2008 Refuel Outage included approximately 99 penetrations and valves. The number of tested components varies since, based on past performance history and the non-intrusion into certain components, certain penetrations and valves can have their LLRTs extended for a number of operating cycles. However, components such as the entrance airlock, MSIVs, drywell head flange and drywell access flange, are tested every outage.

This review of the 2007 and 2008 LLRT test data showed acceptable results that were consistent between the 2007 and 2008 tests. Due to the operating environment and cycling of the Main Steam Isolation valves (MSIVs), these components typically require maintenance work for the successful passage of their LLRTs. At ENVY, experience has shown that the MSIVs have a problem with stem scoring. A root cause analysis was conducted in 2004 after valve inspections revealed this condition. Most of the MSIVs have been modified to replace a steel spacer (bushing) with a carbon spacer. Six of the eight MSIVs were modified in 2004 and one was modified this past outage. One MSIV remains to be modified.

The MSIV that received the modification during this outage passed the leak rate test. The stem scoring occurs during a fast valve stroke of MSIVs with the steel spacer. The 86B MSIV that was modified this outage showed stem scoring, passed the leak rate test, but was replaced as a pre-emptive measure.

In this past outage, two MSIVs (80A, 86A) failed the leak rate test due to corrosion products on the seat. Maintenance action included valve disassembly, cleaning, seat cutting, and re-assembly, and the passing of the leak rate test. An investigation of events leading to these leak rate failures was requested.

Based upon the areas reviewed and the data received, the programmatic controls, inspections and tests for the implementation of the containment leakage rate testing are generally consistent with industry practices and the results meet industry standards.

General Conclusions

Based on the review of the information received during this assessment, ENVY is performing its containment inspections and leakage rate testing consistent with industry programs and standards. The team did not identify any issues that would impact long term reliability of the drywell shell, torus shell or the isolation valves.

Detailed visual and non-destructive examinations per the requirements of the ENVY In-service Inspection Program need to be continued at a high performance level for the life of the station. Trending of potentially degraded areas and early identification of adverse conditions can mitigate issues affecting station reliability.

Due to the critical nature of the torus shell structure and its shell thickness design margin, continuous monitoring and inspections are essential to the long term health of this structure. Visual inspections combined with ultrasonic testing will provide an assessment of the current state of the condition of the shell and provide trending indications for potential corrosion rates at various portions of the shell. For the long term reliability of the torus, continual monitoring and trending must be a primary focus point for the health of the structure. Continued participation in industry groups such as the BWR Mark I Owners Group is important so that ENVY can maintain a proactive approach to industry issues that may arise in the future.

The results of the Type A ILRT to be performed in the spring of 2010 need to be submitted to the Vermont Department of Public Service per the requirements of the Memorandum of Understanding. The remaining items requested by the DPS in the MOU also need to be submitted to the DPS as the results become available.

The programmatic controls, procedural requirements, inspections and testing performed for primary containment are consistent with industry practices and the overall results of the assessment performed meets expectations. However, as noted above, this critical structure and its components must be continually monitored and assessed during plant operation and during Refuel Outages.

References

Procedures and Specifications:

1. OP-4115, Rev. 60, "Primary Containment Surveillance"
2. OP-4029, Rev. 10, "Type A – Primary Containment Integrated Leak Rate Testing"
3. OP-4030, Rev. 72, "Type B and C Primary Containment Leakage Rate Testing"
4. OP-4031, Rev. 12, "Type B and C Primary Containment Leakage rate Calculations and Evaluations"
5. EN-DC-334, "Primary Containment Leakage Rate Testing (Appendix J)"
6. ENN-DC-120, Rev. 0, "ASME Section XI Code Programs"
7. NN-EP-S-001, Rev. 1, "IWE General Visual Containment Inspection"
8. ENN-NDE-9.05, Rev. 1, "Ultrasonic Thickness Examination Report", Inspection Report 07-023 for Torus Bays 1 - 16
9. SEP-CI-001, Rev. 1, "Vermont Yankee Containment In-service Inspection (CI) Program Section"
10. SEP-CI-001, Rev. 1, "Vermont Yankee Containment In-service Inspection (CI) Program Section" with the inclusion of the Sand Drain Tube Inspections of February, 2007
11. SEP-APJ-009, Rev. 1, "Vermont Yankee Primary Containment Leakage Rate Testing (Appendix J) Program Section"
12. General Electric Specification 21A5837, Rev. 3, "Drywell and Suppression Chamber Containment Vessels"

Drawings:

1. 5920-9100, Rev. 2, "General Plan – Torus Modifications"
2. 5920-9111, Rev. 1, "Existing Suppression Chamber SRV Lines with Removal Details"

Documents

1. Memorandum of Understanding (MOU) dated May 2, 2006 between Entergy Nuclear Vermont Yankee, LLC, and Entergy Nuclear Operations, Inc. (together "Entergy VY"), and the Vermont Department of Public Service (the "DPS")

2. Advanced Copy of the Results of Local Leakage Rate Testing for RFO 27 (2008 Outage)
3. VYM 98/132, Memorandum dated 7-21-1998 entitled "Summary of ASME Section XI Subsection IWE Torus shell & Downcomer Inspections Performed During the 1998 Refuel Outage"
4. VYM 99/148, Memorandum dated 11-29-1999 entitled "ASME Section XI IWE Report on Evaluation of 1998 Torus Inspections"
5. VYM 99/112, Memorandum dated 12-7-1999 entitled "Pre-Evaluated Acceptance Criteria for Primary Containment ASME Section XI IWE Inspections"
6. Calculation VYC-1032, Rev. 1, "Torus Shell and Vent System Thickness Requirements"
7. Calculation VYC-2043, Rev. 0, "Evaluation of Primary Containment Localized Thinning and Screening Criteria for ASME XI IWE Inspections"
8. Thickness Calibration Data Sheet No. 8053-01-063 dated 5/8/2001
9. Technical Evaluation No. 2000-011, Evaluation of Drywell Subsurface Condition, Elevation 238' "
10. BVY 92-91, dated 7/21/1992, "1992 Vermont Yankee Reactor Containment Building Integrated Leakage Rate Test Report"
11. General Physics Corporation Integrated Leakage Rate Test dated July, 1995
12. NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J"
13. BVY 02-62, dated 10/4/2002, "Integrated Leak Rate Test Interval Extension"
14. BVY 04-077, dated 10/5/2004, "One-Time Integrated Leak Rate Test (ILRT) Interval Extension"
15. NVY 05-108, dated 8/31/05, "One-Time Extension of Integrated Leak Rate Test Interval"
16. NRC Regulatory Guide 1.163, entitled "Performance Based Containment Leak-Test Program", dated September 1995
17. NRC IN 86-99, "Degradation of Steel Containments"
18. NRC IN 86-99, Supplement 1, "Degradation of Steel Containments"
19. NRC IN 88-82, "Torus Shells with Corrosion and Degraded Coatings in BWR Containments"
20. NRC IN 88-82, Supplement 1, "Torus Shells with Corrosion and Degraded Coatings in BWR Containments"
21. NRC Information Notice 2006-01, "Torus Cracking in a BWR Containment"
22. VY Main Steam System Design Basis Document, Document MS, Revision 20

Condition Reports:

1. Condition Report CR-JAF-2005-02593, dated 6-27-2005, "Root Cause Analysis Report – Torus Leak Discovered Near the Support Between Bays "A" and "P" – Revision 1. Report date of 7-26-2005

2. Condition Report LO-NOE-2005-00909, "CR-JAF-2005-02593-RC, Torus Leak Discovered Between The Support Between Bays "A" and "P" and associated Root Cause Analysis Report, Rev. 1 dated July 26, 2005.
3. Condition Report LO-NOE-2006-00556, "IN-2006-001 Torus Cracking in a BWR Mark I Containment".
4. ER 20022211, V2-80B Local Leak Rate Test Results Exceed Acceptance Criteria
5. ER 20022212, V2-86B Local Leak Rate Test Results Exceed Acceptance Criteria
6. CR-VTY-2004-00836, Type B and C Primary Containment Leakage Rate Testing
7. CR-VTY-2004-00839, Type B and C Primary Containment Leakage Rate Testing
8. CR-VTY-2004-00841, Type B and C Primary Containment Leakage Rate Testing
9. CR-VTY-2008-04326, SB-16-19-6A/B & 7A/B failed Appendix J Local Leakage Rate Testing

2.9 Underground Piping Program Evaluation [3(a)(7)]

Introduction

Act 189 included an in-depth inspection of 'an underground piping system that carries radionuclides'. However, there are no underground piping systems carrying radionuclides at ENVY. As an alternative and in agreement with the Department of Public Service and the Public Oversight Panel, the buried piping in the Service Water System was selected for a detailed examination of the ENVY underground piping inspection program.

Methodology

The examination of the buried piping inspection program was conducted utilizing document reviews, personnel interviews, site visits and examinations of prior inspection records. The document reviews included UFSAR description of the Service Water System, the Service Water DBD, P&IDs, the ENVY License Renewal application, the NRC SER of License Renewal of VYNPS, NRC License Renewal Inspection Report, buried piping inspection procedures, and other applicable documents.

Observations

The Buried Underground Piping Inspection Program at ENVY was conducted on an opportunistic basis, i.e., when an excavation occurred that resulted in uncovering a buried piping system, that uncovered portion of the pipe was examined to determine the state of the protective coating and the condition of the external surface of the pipe. During plant construction, all buried steel surfaces were covered with coal-tar epoxy coating and an outside wrap in accordance with the EBASCO coating guide for its application to steel surfaces.

To date, three such opportunistic inspections have been conducted for which inspection records were maintained, a 24-inch service water system pipe in 2003, a 12-inch fire protection system pipe in 2007, and a 78-inch augmented off-gas (AOG) system vent pipe in 2006. These inspections were conducted in accordance with the requirements of PP 7030 Structures Monitoring Program Procedure LPC No. 1 dated July 9, 2002. In each case the examined portion of the pipe was in excellent condition. These findings, after more than 30 years of operation, tend to support the conclusion that the combination of coating and soil conditions provide for a low potential of external pipe surface deterioration in the buried underground piping systems at this site.

In late 2007, Entergy Nuclear issued a fleet procedure, EN-DC-343 *Buried Piping and Tanks Inspection and Monitoring Program*. This procedure requires each site in the Entergy Nuclear fleet to develop its own site specific program in accordance with the requirements of this procedure. The procedure specifies the program content, scope, ranking methodology, priorities and inspection frequency. It provides guidance for assessing corrosion risk at a particular site based on soil conditions, component material and external protective treatment of the component. Based on the level of corrosion risk it provides required inspection intervals for initial inspections and inspection intervals.

The buried underground piping program covers the piping and tanks identified in the license renewal application, piping that could provide a path of plant-generated radioactive material contamination to groundwater, and other piping that could present an environmental concern. The program includes inspection and monitoring of selected operational buried piping and tanks for external corrosion, including crevice, general, microbiologically induced, and pitting corrosion.

ENVY has developed an action plan for implementing the site specific program and to date has identified the buried sections of piping and tanks, preliminarily performed an impact assessment, and preliminarily evaluated corrosion risk. ENVY performed this assessment using conservative assumptions regarding coating condition and soil resistivity. The results of this effort are documented in a table entitled *Buried Piping and Tanks Inspection and Monitoring Program*. In order to finalize this tabulation ENVY needs to perform soil resistivity measurements and to determine coating quality. For service water piping ENVY has identified four pipe segments, three 24-inch and one 20-inch, all of which are preliminarily judged to be high impact because of the conservative assumptions regarding coating condition and soil resistivity. This piping is carbon steel, coated with coal-tar epoxy and wrapped with asbestos wrap. Once the coating condition and soil resistivity are determined a more realistic assessment of its impact will be concluded. ENVY expects to implement the remaining parts of the action plan, including site specific program development and a long term inspection plan, by mid-2009. ENVY's current program status is consistent with the industry.

EPRI is developing a guidance document providing methods and analyses for performing inspections and Entergy is actively involved in this program. Additionally, Entergy has an action plan to develop the guidance and define methodologies for inspecting, monitoring, evaluating inspection data and mitigating corrosion failures in buried piping/tanks systems for the Entergy fleet. This action plan is scheduled to be completed in the first quarter of 2009.

In its License Renewal Application, ENVY has committed to the US NRC to provide by March 21, 2012, enhanced guidance for performing examinations of buried piping by specifying that coating degradation and corrosion are attributes to be evaluated.

General Conclusions

Given the buried piping inspection findings to date, the application of the ENVY program described above should provide reasonable assurance of the reliability of the service water system buried piping as far as external corrosion is concerned. Monitoring and mitigation of internal corrosion of service water piping is covered in a separate program, which also includes the monitoring and mitigation of corrosion of the internal surfaces of buried service water piping.

Document References

1. EN-DC-343 Rev. 1, Buried Piping and Tanks Inspection and Monitoring Program
2. EN-DC-343, Buried Piping and Tanks Inspection and Monitoring Program Action Plan
3. Entergy Nuclear Operations Buried Piping Action Plan
4. EBASCO Coating Guide CP-25, Coating Steel with Coal-Tar Epoxy Coating, Rev. 10/7/1962

5. PP 7030, Structures Monitoring Program Procedure, LPC No. 1
6. VYNPS License Renewal Application
7. NRC SER, License Renewal of VYNPS
8. Vermont Yankee Nuclear Power Station, License Renewal Commitment List, Revision 0
9. NRC License Renewal Inspection Report 05000271/2007006
10. G-191159, SH. 1 & 2, Flow Diagram, Service Water System
11. Vermont Yankee Buried Underground Piping Presentation and Interview, M. LeFrancois (ENVY) et al.
12. Vermont Yankee Service Water System Presentation and Interview, H. Breite et al.
13. Follow-up Interview re: BPTip, G. Bailey and M. LeFrancois (ENVY)

2.10 Cable Separation Practices [3(c)]

Introduction

As part of the reliability assessment of ENVY, cable separation was identified as an issue to be included in the technical assessment of the station. The scope of this review activity included the separation of safety systems including physical and electrical separation. Other nuclear power plants have experienced issues with cable separation and it is the intent of this assessment to evaluate whether ENVY has adequate cable separation, has properly evaluated past industry experience with regards to cable separation and has the capability to move forward and operate the station reliably with a comprehensive cable separation program.

System Description

Electrical cable separation is an important design feature of a nuclear power plant. Based on the defense-in-depth approach to the design of a nuclear plant, redundant equipment and components must be adequately separated to assure safe operation and safe shutdown of the plant in the event of equipment malfunction or failure due to any number of conditions. The electrical cables that provide power, instrumentation and control to this equipment and components must also be adequately separated such that damage to these cables does not adversely impact the operation and/or control of the respective functions of this equipment.

Cable separation is controlled through a series of specifications and procedures to assure that proper cable separation is defined at the outset of plant design, implemented through the construction process and controlled as plant changes and improvements are made during the operation life-cycle of the plant. One such improvement was 10 CFR 50, Appendix R, Section III.G that was back fit to ENVY to ensure that adequate cable separation was provided to survive the effects of a fire. 10 CFR 50, Appendix R cable separation requirements apply to all US nuclear power plants.

Methodology

This assessment was performed through a variety of methods including document reviews, site visits, and interviews with plant personnel, in plant walk-downs and comparison of certain features at ENVY with standard acceptable practices in the nuclear power industry.

Some of the key documents reviewed included the UFSAR sections related to cable separation, the original General Electric project standard for cable separation, the original EBASCO specification for cable separation, the current ENVY specification for cable separation, the ENVY fire hazards analysis, the ENVY safe shutdown analysis, NRC inspection reports, internal audits, self assessments, condition reports and other associated documentation on the cable separation issue.

Interviews were conducted with the engineer most responsible for cable separation at the station as well as with the licensing individual most cognizant of cable separation issues. The interviews were intended to question the plant personnel on the past, present and future status and maintenance of the cable separation program and to gain an understanding of their knowledge of the issues involving cable separation.

Plant walk-downs were performed with the engineer most responsible for cable separation. The purpose of the walk-downs was to visually inspect some of the plant areas critical to the cable separation issue.

Based on the information gathered during the assessment, a general comparison of the condition of the ENVY cable separation program and plant parameters was judged against plants of similar vintage, design and licensing requirements.

Evolution of US Nuclear Power Plant Fire Protection Programs

The following background frames the changes made to Nuclear Power Plant Fire Protection Programs and provides insights into the reason for the evolution of nuclear power plant cable separation requirements. During the initial implementation of the U.S. nuclear reactor program, regulatory acceptance of fire protection programs at nuclear power plants was based on the broad performance objectives of General Design Criterion 3 (GDC 3) in Appendix A to 10 CFR 50. Appendix A establishes the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety. GDC 3 addresses fire protection requirements and specifies, in part, that (1) structures, systems, and components important to safety must be designed and located to minimize the probability and effects of fires and explosions, (2) noncombustible and heat-resistant materials be used wherever practical, and (3) fire detection and suppression systems be provided to minimize the adverse effects of fires on structures, systems, and components important to safety. However, during this early stage of nuclear power regulation, given the lack of detailed implementation guidance for this general design criterion, the level of fire protection was generally found to be acceptable if the facility complied with local fire codes and received an acceptable rating from its fire insurance underwriter. Thus, the fire protection features installed in early U.S. nuclear power plants were very similar to those installed in conventional fossil-fuel power generation stations.

A fire at the Browns Ferry Nuclear Power Plant, Unit 1, on March 22, 1975, was a pivotal event that brought fundamental change to fire protection and its regulation in the U.S. nuclear power industry. The fire started when plant workers in the cable spreading room used an open flame to test for air leakage through a non-fire-rated (polyurethane foam) penetration seal that led to the reactor building. The fire ignited both the seal material and the electrical cables that passed through it, and burned for almost 7 hours before it was extinguished using a water hose stream. The greatest amount of fire damage actually occurred on the reactor building side of the penetration, in an area roughly 40 feet by 20 feet. More than 1600 cables, routed in 117 conduits and 26 cable trays, were affected and, of those cables affected, 628 were important to safety. The fire damage to electrical power, control systems, and instrumentation cables impeded the functioning of both normal and standby reactor cooling systems and degraded plant monitoring capability for the operators. Given the loss of multiple safety systems, operators had to initiate emergency repairs in order to restore the systems needed to place the reactor in a safe shutdown condition.

The investigations that followed the Browns Ferry fire identified significant deficiencies, both in the design of fire protection features and in licensee procedures for responding to a fire event. The investigators concluded that the occupant safety and property protection concerns of fire insurance underwriters did not sufficiently encompass nuclear safety issues, especially in terms of the potential for fire damage to cause the failure of redundant success paths of systems and components important for safe reactor shutdown. In its report (NUREG-0050, February 1976, *Recommendations Related to Browns Ferry Fire*), the NRC Browns Ferry special review team recommended that the NRC (1) develop detailed guidance for implementing the general design criterion for fire protection and (2) conduct a detailed review of the fire protection program at each operating nuclear power plant, comparing it to the guidance developed.

In May 1976, the NRC issued Branch Technical Position (BTP) APCSB 9.5-1, which incorporated the recommendations from the Browns Ferry fire special review team and provided technical guidelines to assist licensees in preparing their fire protection programs. As part of this action, the staff requested each licensee to provide an analysis that divided the plant into distinct fire areas and demonstrated that redundant success paths of equipment required to achieve and maintain safe shutdown conditions for the reactor were adequately protected from fire damage. However, the guidelines of APCSB 9.5-1 applied only to those licensees that filed for a construction permit after July 1, 1976.

In September 1976, in an effort to establish defense-in-depth fire protection programs, without significantly affecting the design, construction, or operation of existing plants that were either already operating or well past the design stage and into construction, the NRC modified the guidelines in APCSB 9.5-1 and issued Appendix A to APCSB 9.5-1. This guidance provided acceptable alternatives in areas where strict compliance with APCSB 9.5-1 would require significant modifications. Additionally, the NRC informed each plant that the guidance in Appendix A would be used to analyze the consequences of a postulated fire within each area of the plant and requested licensees to provide results of the fire hazards analysis performed for each unit and the technical specifications for the present fire protection systems.

Early in 1977 each licensee responded with a fire protection program evaluation that included a fire hazard analysis. These analyses were reviewed by the staff using the guidelines of Appendix A to APCSB 9.5-1. The staff also conducted inspections of operating reactors to examine the relationship of structures, systems, and components important to safety with the fire hazards, potential consequences of fires, and the fire protection features. After reviewing licensee responses to the BTP, the staff determined that additional guidance on the management and administration of fire protection programs was necessary, and in mid-1977, issued Generic Letter 77-002, which provided criteria used by the staff in review of specific elements of a licensee’s fire protection program, including organization, training, combustible and ignition source controls, firefighting procedures and quality assurance. Many fire protection issues were resolved during the BTP review process, and agreements were included in the NRC-issued Safety Evaluation Reports (SERs).

By the late 1970s to early 1980, the majority of operating plants had completed their analyses and implemented most of the fire protection program guidance and recommendations specified in Appendix A to the BTP. In most cases, the NRC had found the licensees' proposed modifications resulting from these analyses to be acceptable. In certain instances, however, technical disagreements between licensees and the NRC staff led to some licensees' opposition to adopt some of the specified fire protection recommendations, such as the requirements for fire brigade size and training; water supplies for fire suppression systems; alternative, dedicated, or backup shutdown capability; emergency lighting; qualifications of penetration seals used to enclose places where cables penetrated fire barriers; and the prevention of reactor coolant pump oil system fires. Following deliberation, the Commission determined that, given the generic nature of some of the disputed issues, a rulemaking was necessary to ensure proper implementation of NRC fire protection requirements.

In November 1980, the NRC published the 'Fire Protection' rule, 10 CFR 50.48, which specified broad performance requirements, as well as Appendix R, *Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979*, to 10 CFR Part 50, which specified detailed regulatory requirements for resolving the disputed issues.

As originally proposed (Federal Register, Vol. 45, No. 1 and 5, May 22, 1980), Appendix R would have applied to all plants licensed prior to January 1, 1979, including those for which the staff had previously accepted the fire protection features as meeting the provisions of Appendix A to APCSB 9.5-1. After analyzing comments on the proposed rule, the Commission determined that only three of the fifteen items in Appendix R were of such safety significance that they should apply to all plants (licensed prior to January 1, 1979), including those for which alternative fire protection actions had been approved previously by the staff. These items are fire protection of safe shutdown capability, Section III.G (including alternative, dedicated, or backup shutdown systems), emergency lighting, Section III.J, and the reactor coolant pump oil system, Section III.O. Accordingly, the final rule required all reactors licensed to operate before January 1, 1979, to comply with these three items even if the NRC had previously approved alternative fire protection features in these areas (Federal Register, Vol. 45, Nov. 19, 1980). In addition, the rule established an exemption process for licensees, provided that the presence of required a fire protection feature for which an exemption was sought would not enhance fire protection safety, or that such modifications would not be detrimental to overall safety (10 CFR 50.48(c)(6)). Appendix R to 10 CFR Part 50 and 10 CFR 50.48 became effective on February 17, 1981. 10CFR50, Appendix R, Section III.G is applicable to ENVY and contains the specific requirements for cable and component separation.

Observations

The original design specification for cable separation was contained in General Electric (GE), 22A1421, *Electrical Equipment, Separation for Safeguards System*, dated October, 1968 which defined the requirements for the separation and identification of the Reactor Protection System (RPS), Primary Containment Isolation System (PCIS) and Engineered Safeguards Systems (ESS) to ensure independence of redundant equipment such that the single failure criteria is satisfied. This GE design specification provided a comprehensive set of design criteria for the systems listed above but did not address non-safety-related or non-vital plant systems. GE was the nuclear steam system supply contractor for ENVY.

EBASCO provided a document entitled *Ground Rules for Separation and Identification of Reactor Protection and Safeguard Systems – Related Electrical Equipment and Wiring*, Rev. 3, dated June 7, 1971. This document was considered the installation specification to implement the GE criteria and as such included criteria specific to ENVY. EBASCO was the general contractor for construction of ENVY.

ENVY Design Engineering developed VYS-027, *Specification for Separation for Reactor Protection, Engineered Safety Feature and Auxiliary Support Systems – Related Electrical Equipment and Wiring*, Rev. 14, dated June 27, 2006 to update and clarify the original ground rules established in the EBASCO Report. VYS-027 provides criteria for new systems and plant modifications, and it provides a clearer definition of separation of analog instrumentation. It also establishes a comprehensive set of criteria for preparation of design changes initiated after January 1, 1991. VYS-027 establishes a coherent and definitive set of criteria to provide physical separation and electrical isolation of circuits and components so that the safety functions required during and following any design basis event can be accomplished.

Appendix I of the original FSAR includes questions and answers with the NRC regarding the licensing basis of ENVY. ENVY was asked to submit cable installation design criteria to preserve the independence of the redundant RPS and ESS circuits. In response, ENVY stated that the non-vital cables are not to be run in the wireways of more than one ESS and Emergency Core Cooling System (ECCS) division. This response is dated October of 1970 with the same language that exists in today's version of the UFSAR.

Cable separation criteria are primarily contained in two sections of the UFSAR:

- Section 7.2.3.10, Wiring
- Section 8.4.6, Cable Installation and Separation Criteria

During plant construction, ENVY installation of engineered safeguards cabling was based on the EBASCO 'ground rules' document that specified the degree of separation, based on the potential hazards in a particular zone of the power plant. These zones were classified in four separate categories as follows: Mechanical Damage (Missile) Zone, Fire Hazard Zone, Cable Vault Room and Main Control Panel Room. The system separation requirements were defined for the Reactor Protection System (RPS), Primary Containment Isolation System (PCIS), Engineered Safeguard Systems (ESS), and On Site Standby Power Sources.

As ENVY operated over time, some limitations in the original cable separation criteria were recognized. For example, the original criteria did not address 'low-level' circuits or post accident monitoring requirements. In 1987, ENVY installed two divisions of low level instrumentation trays to provide separate redundant low level instrumentation paths. Prior to this, any non-RPS safety-related low level instrumentation circuits were installed in conduit to achieve required separation. Industry events such as the Brown's Ferry fire brought further focus to the cable separation issue with subsequent back fit requirements promulgated in 10 CFR Part 50, Appendix R. ENVY implemented design changes to address Appendix R in the mid 80's to address the new separation requirements for redundant fire safe shutdown systems.

In 1995, ENVY initiated a reverification of the Appendix R Circuit Analysis. This was a major undertaking that analyzed the circuit routing of approximately 1,500 Appendix R safe shutdown cables. This effort verified that the analyzed cables were routed in accordance with the separation criteria outlined in Appendix R. The effort also identified the need for some drawing corrections to address minor discrepancies.

Recent NRC inspections do not contain any significant findings related to cable separation issues or compliance to Appendix R with regards to cable separation.

The senior staff engineer most responsible for the cable separation program was interviewed along with a senior lead licensing engineer. The interview included questions and answers concerning industry issues and problems associated with cable separation and how ENVY has dispositioned these issues over its operating history. One of the focused aspects of the discussions was the comparison of the ENVY cable separation program to other nuclear power plant's cable separation programs including a comparison of the issues identified at the Maine Yankee nuclear power station. Maine Yankee was shut down approximately 10 years ago and has since been decommissioned. One of the areas at Maine Yankee that led to the shutdown was problems associated with cable separation.

Cable separation is an important parameter in the design of a nuclear plant's electrical system and as such has received a high degree of attention from the NRC and other auditing bodies. Cable separation was a critical issue at the Maine Yankee station. Based on the root cause investigation of the cable separation issues at Maine Yankee, many factors contributed to their cable separation problems. Among these were ambiguous and conflicting criteria, inconsistent and changing goals which reduced engineering sensitivity to important issues such as cable separation, a high threshold for problem identification and a culture that limited the scope of problems and minimized corrective actions.

Plant walk-downs were performed as part of this assessment to visually inspect some of the plant areas critical to the cable separation issue. The plant areas walked-down were as follows:

- 4KV East Switchgear Room – Elevation 248'-6"
- 4KV West Switchgear Room – Elevation 248-6"
- Cable Vault – Elevation 260'-6"
- Reactor Building – Elevation 252'-6"

Each of the above areas was accessed and a visual inspection of the areas was performed. The switchgear rooms were originally constructed as one large area and then, as part of the Appendix R analysis, a one hour fire rated wall was constructed to split the room into two rooms (i.e. east and west) with each room containing separate divisions of electrical raceway.

To provide adequate cable separation for the different divisions of safe shutdown circuits, the feeder cable for Motor Control Center (MCC) 9B is wrapped in a one-hour rated fire barrier in the west Switchgear Room. The barrier is constructed of Interam E-54A and it was observed to be installed in a manner consistent with other similar qualified fire barriers of this type. Both Switchgear Rooms exhibited good housekeeping practices at the time of the inspection.

The cable vault (cable spreading room) was also walked-down during the inspection. The cable vault is located directly beneath the Main Control Room such that numerous cables of both divisions penetrate the Control Room Floor and enter the Cable Vault. Through a series of conduits, cable trays and wireways, the cables are routed to various areas of the plant. Cables in the Cable Vault are separated by a horizontal distance of 3 ft. and all vertical cable risers are enclosed with metal covers. This configuration meets the plant's design criteria as outlined in the original GE and EBASCO specifications as well as ENVY's current electrical separation criteria specification (VYS-027). The walk-down observed that the required vertical tray covers were installed as required.

During the approximate period of 2001 to 2003, there were a number of discrete cable separation issues identified in the Cable Vault. These issues were typically documented on Condition Reports and dispositioned on an individual basis. These issues included cable routing problems, cable tray riser identification and cable tray riser physical configuration (missing covers or covers being ajar). As the result of this apparent adverse trend in cable separation issues, the station initiated a Condition Report to evaluate the adverse trend and to attempt to correct the situation. Interim compensatory measures included a check for vertical riser tray covers on all Cable Vault trays and a heightened awareness for the replacement of these covers. For long-term corrective action to ensure that cable tray covers remain installed as required and comply with VYS-027 requirements, the station requires a barrier breach to remove a cover for maintenance. The barrier breach program ensures that the cover gets reinstalled as required.

ENVY is committed to Sections III.G, III.J, III.L and III.O of 10CFR50, Appendix R. Appendix R provides for proper cable and component separation to assure the safe and reliable shutdown of the station in the event of a fire. The safe shutdown portion of Appendix R requires the selection of cables and components of different safe shutdown trains and their proper separation to assure that one train of safe shutdown cables and components remain "free of fire damage" in the event of a fire. Thus, the Appendix R analysis serves as a technical method to assure that safe shutdown cables are adequately separated in the station.

Using the requirements of Sections III.G and III.L of Appendix R, the capability to achieve hot shutdown must exist given a fire in any area of the plant. Section III.G of Appendix R provides four methods for ensuring that the hot shutdown capability is protected from fires. The first three options as defined in Section III.G.2 provide methods for protection from fires to equipment needed for hot shutdown:

1. Redundant systems including cables, equipment, and associated circuits may be separated by a three-hour fire rated barrier; or,
2. Redundant systems including cables, equipment and associated circuits may be separated by a horizontal distance of more than 20 feet with no intervening combustibles. In addition, fire detection and an automatic suppression system are required; or,
3. Redundant systems including cables, equipment, and associated circuits may be enclosed by a one-hour fire rated barrier. In addition, fire detectors and an automatic fire suppression system are required.

The last option as defined by Section III.G.3 provides an alternative shutdown capability to the redundant trains damaged by the fire.

4. Alternative shutdown equipment must be independent of the cables, equipment and associated circuits of the redundant systems damaged by the fire.

In the 1995 – 1997 timeframe, ENVY performed a comprehensive re-analysis of its Appendix R safe shutdown analysis to assure compliance to the regulation and to verify adequate cable separation for safe shutdown cables. The routing of required Appendix R safe shutdown cables was documented on datasheets and verified by plant walk-downs. Using these datasheets, system availability matrices were prepared which document a fire area by fire area assessment of available safe shutdown systems. The system availability matrices are found in Attachment C of VYC-1507.

Included with the Appendix R analysis are approximately 11 exemption requests where ENVY has sought regulatory relief from certain aspects of Appendix R due to specific plant configurations. One of the exemptions (See Section 6.5 of the safe shutdown capability analysis) requested relief from the 20 ft. separation between redundant trains since only 18 ft. was available based on cable tray configurations. The NRC granted this exemption request.

However, during a recent self-assessment, it was identified that the actual distance between the two redundant cable trays at the nearest point was only 17'- 7½". As a result, the station revised and re-submitted the exemption request to the NRC to request relief based on the 17'-7½" separation distance. This is still an open issue at the time of the writing of this report. Although the station believes this exemption request will be granted by the NRC, there is a risk that the NRC may reject the exemption and require other action on the part of the station. The station has several options available to resolve this issue should the NRC deny the exemption request.

An important aspect of the cable separation program is the management of the cable information with respect to routing and cable location. Currently, the database at ENVY is basically a manual. As changes are made or questions are asked, the system must be manually tracked on a page by page basis.

During the Appendix R re-examination, a computer based program was used as a tool to effectively analyze all of the cable and component data required to manage the cable separation requirements for Appendix R. However, to date, ENVY does not have a recognized controlled computer database to manage its cable separation issues. This is done currently by manual review and interface with the responsible engineers. Most nuclear power plants have a recognized computer based cable management program that enables the accurate mapping of the cables in the station and can examine issues when potential cable separation discrepancies are discovered or when plant modifications are in the engineering design stage. It was discussed with plant personnel that ENVY intends to move its cable data into a computer based management system, however, it must conform to the Entergy corporate model for this type of program, such that the same program can be used at all of the Entergy sites. Currently, no approved transition plan is in place to migrate the safe shutdown cable routing information to a computer based data management system.

NRC Generic Letter 2006-03 requested all licensees to confirm their compliance with existing applicable regulatory requirements in light of information regarding certain fire barrier systems known as HEMYC and MT. Specifically, although HEMYC and MT fire barrier systems may be relied upon to protect electrical and instrumentation cables and equipment that provide safe shutdown capability during a fire, NRC testing has revealed that both of these materials failed to provide the level of protective function intended for compliance with existing regulations. The NRC requested licensees to describe the use of these fire barrier systems as well as other fire barrier systems at their respective stations.

ENVY provided its response to the NRC in letter BVY 06-045, dated June 6, 2006. In its response, ENVY stated that the plant currently has no HEMYC or MT fire barrier material installed in applications that are relied upon for separation or safe shutdown purposes, having removed the last of such material in July of 2005. The company also stated that it has never had the MT fire barrier material installed in the plant.

The HEMYC material formerly installed in credited applications has been replaced with 3M Interam E-54A fire barrier material. This replacement was performed under Engineering Request 05-0400 which was accomplished in 2005. During the plant walk down, it was identified that there is still one HEMYC installation in the plant, however, this installation is not relied upon or credited for cable separation for fire safe shutdown. Based on the review of the documentation provided for the ENVY response to GL 2006-03, it appears as though this issue was adequately addressed and dispositioned.

NRC inspection reports were also reviewed as part of this assessment. Over the operation history of the plant, NRC inspections questioned some specific cable separation issues at the station. In NRC Inspection Report 50-271/97-03, dated May 8, 1997, the NRC reviewed the results of the Appendix R cable re-analysis performed at the site. In the inspection report, the NRC noted some areas where cable routing was in direct conflict with the cable separation criteria identified in UFSAR Section 8.4.6. Most of these cable separation issues were self-identified by ENVY personnel as they performed the Appendix R re-analysis. ENVY staff had initiated a 100 percent review of all cable and conduit lists (CCLs) and cable tray isometrics to determine the full extent of the cable separation issue. This review identified several electrical cable separation design specification non-conforming conditions. In this inspection report, the NRC concluded that the ENVY staff was continuing to aggressively examine the plant design basis and address design and installation inconsistencies in a timely manner. However, the NRC questioned ENVY's immediate operability assessment that was based upon reliance on safety class electrical breakers protecting the potentially impacted safety related cabling and developed Unresolved Issue (URI) 97-03-02 to further inspect and track this item.

NRC Inspection Report 50-271/97-05, dated August 19, 1997, continued to track URI 97-03-02. In this report, the NRC described ENVY's implementation of the process for disposition of the non-conforming cable separation issues as appropriate and felt that operability determinations were adequately supported.

NRC Inspection Report 50-271/97-12, dated March 3, 1998, further examined URI 97-03-02. In this report, the NRC stated that the ENVY's staff identification and corrective actions to address electrical cable separation design issues were appropriate and were executed or planned to be accomplished in a time period commensurate with their safety significance. As a result, the NRC closed URI 97-03-02.

The most recent NRC Triennial Fire Protection Inspections were also reviewed. NRC Inspection Report 05000271/2007008, dated April 12, 2007, included an inspection of safe shutdown and alternate shutdown capability including electrical circuit reviews. For this inspection, the NRC identified no findings of significance. NRC Inspection Report 05000271/2004010, dated January 26, 2005, was also reviewed. In this inspection, the NRC identified a finding of very low safety significance relating to the electrical isolation for a RCIC valve that may have caused spurious operation of the valve in the event of a fire in the control room or cable vault. ENVY analyzed and dispositioned the finding to the satisfaction of the NRC.

Ongoing plant design and modification issues are currently controlled using the modification process. Procedural guidance leads one performing a plant modification to answer a number of screening and evaluation questions that are aimed at effectively controlling cable routing and cable separation issues.

Cable separation design issues currently reside with the Electrical Engineering Group within the ENVY Engineering Department. In accordance with the Entergy corporate model, the function of electrical separation and the Appendix R Safe Shutdown Program is being transferred to the Program and Components Group. ENVY staff interviewed indicates the individual responsible for fire safe shutdown is currently in-place. The history and knowledge associated with cable separation issues is an important and needs to be diligently transferred from one group/individual to another. This is a potential watch area since the work force in general is aging and the appropriate transfer of history, knowledge and documentation is critical to the ongoing success of proper cable separation at ENVY.

General Conclusions

Based on the review of the information received from the station, personnel interviews conducted and the plant walk downs performed, the cable separation design issue appears to have been designed, implemented and controlled in a manner consistent with good industry practices.

There were no cable separation issues as they relate to NRC Electrical Distribution System Functional Inspection of ENVY, Report No. 50-271/92-81.

There were no cable separation issues related to EPU modifications.

It is recommended that the current cable database be transferred into an industry accepted computer data management system to assure long term compliance and effective design control.

Although it is anticipated that the resubmitted exemption request to the NRC regarding proper cable separation on elevation 252'-6' of the Reactor Building will receive NRC approval, this is an open issue and needs to be modified and monitored to its final resolution.

The transfer of the cable separation program and Appendix R Safe Shutdown Program from the Electrical Group to the Programs and Components Group needs to be performed in a controlled manner to insure the proper transfer of knowledge considering the prolonged evolution of cable separation requirements over time.

References:

Procedures and Specifications

1. Design Specification - General Electric, 22A1421, dated October 1968, Electrical Equipment, Separation for Safeguards System
2. EBASCO Services, Ground Rules for Separation and Identification of Reactor Protection and Safeguard Systems – Related Electrical Equipment and Wiring, Rev. 3, dated June 7, 1971
3. Specification VYS-027, Rev. 14, Separation Criteria for Reactor Protection, Engineered safety Feature and Auxiliary Support Systems – Related Electrical Equipment and Wiring

Drawings

1. G-191372 Sheet. 1, Rev. 67, 125DC One Line Wiring Diagram
2. G-191372 Sheet. 3, Rev. 18, 125DC One Line Wiring Diagram

Documents:

1. US Atomic Energy Commission Report of Inspection CO Report No. 271/70-4, signed on August 25, 1970
2. UFSAR Section 7.2.3.10, Rev. 3, Wiring
3. UFSAR Section 8.4.6, Rev. 17, Cable Installation and Separation Criteria
4. VYNPS Fire Hazards Analysis, Rev. 10
5. VY SSCA, Rev. 9, Safe Shutdown Capability Analysis

6. VYC-1057, Rev. 1, Appendix R Safe Shutdown Analysis, Attachments A thru I.
7. EN-DC-115, Rev. 3, Engineering Change Development.
8. EN-DC-128, Rev. 3, Fire Protection Impact Reviews.
9. EN-LI-100, Rev. 7, Process Applicability Determination.
10. NVY 92-184, NRC Electrical Distribution System Functional Inspection of Vermont Yankee, Report 50-271/92-81, dated September 30, 1992.
11. BVY 92-124, dated October 30, 1992, "Response to NRC Electrical Distribution System Function Inspection of Vermont Yankee, Report No. 50-271/92-81, Reply to a Notice of Violation".
12. BVY 92-135, dated December 7, 1992, "Response to NRC Electrical Distribution System Function Inspection of Vermont Yankee, Report No. 50-271/92-81, Unresolved Items and Other Issues".
13. NVY 97-069, NRC Inspection Report 50-271/97-03, dated May 8, 1997.
14. NVY 97-135, NRC Inspection Report 50-271/97-05, dated August 19, 1997.
15. NVY 98-029, NRC Inspection Report 50-271/97-12, dated March 3, 1998.
16. NVY 05-009, NRC Triennial Fire Protection Inspection Report 05000271/2004010, dated January 26, 2005.
17. NVY 07-045, NRC Triennial Fire Protection Inspection Report 05000271/2007008, dated April 12, 2007.
18. Fire Protection Audit QA-9-2004-VY-1 dated March 1 – 12, 2004
19. Fire Protection Audit QA-09-2006-VY-1 dated January 9 – 24 2006.
20. Fire Protection Audit QA-9-2008-VTY-1 dated January 21 – 31, 2008.
21. Snapshot Assessment / Benchmark for the Appendix R Safe Shutdown Methodology dated November, 2007.

Condition Reports:

1. CR-VTY-1996-0184: Two issues related to cable separation with regards to Appendix R, Section III.G.2 in the Reactor Building.
2. CR-VTY-1997-0303: Non-Safety Related 480VAC feeder cable is routed in both SI and SII divisional tray contrary to VYS-027 and FSAR 8.4.6.
3. CR-VTY-1997-0317: Operability Assessment for Separation of Feeder Cable from MCC 8A to LP-NE-1A.
4. CR-VTY-1998-0181: Contrary to UFSAR 8.4.6, 23 low-level instrument cables are routed along with control cables in a cable tray.
5. CR-VTY-1998-0234: Cable 11322U does not conform to separation criteria (UFSAR 8.4.6/VYS-027).
6. CR-VTY-1998-1225: Cables routed contrary to UFSAR.

7. CR-VTY-2000-1413: Two cables identified that violate electrical separation criteria.
8. CR-VTY-2000-1896: Error in Close-Out memo for BM0 2000-11 – Cable Tray Risers.
9. CR-VTY-2002-2826: Potential cable Separation Violation – RPS Trays.
10. CR-VTY-2003-01399: Adverse trend for cable vault cable separation issues identified by external source.
11. CR-VTY-2003-02268: Cable tray riser walk down identifies additional missing riser covers.
12. CR-VTY-2004-01655: Pre startup walk down of Cable Spreading Room riser covers identified additional housekeeping issues with riser covers.
13. CR-VTY-2008-00299: Discrepancy in NRC approved Appendix R Separation Distance identified during FP/App R QA.

2.11 Large Electric Motor Program Horizontal Review [3(b)]

Introduction

As part of the scope of the ENVY Reliability Assessment a review of the large electric motor program was performed. This evaluation was based on industry standard large electric motor program good practices. ENVY is a member of the Electric Power Research Institute (EPRI). EPRI is a non-profit organization that provides research and guidance to the utility industry and from which many of the base documents and knowledge for the equipment reliability programs are derived. EPRI has established industry good practices for managing large electric motors for power plants. This review also included, as applicable, the motors within the vertical scope of the selected systems.

Methodology

This specific assessment was performed through a variety of methods including document reviews, site visits, system walk-downs, plant personnel interviews, and comparison with generally acceptable programs and practices in the nuclear power industry.

Interviews were conducted with the Predictive Maintenance Engineer and Motor Component Engineer, and Supervisor for the issues contained in this technical area. The interviews were intended to question all aspects of plant activities that affect the reliability of the ENVY critical plant motors. The observations are closely aligned with the technical areas within the scope of this review and are based on the information gathered at the site and through the documents received during the assessment period.

Observations and Findings

ENVY began commercial operation in 1972. The large electric motors are thus of the same vintage. The initial motors were installed with C class winding insulation and designed for a forty-year life expectancy. Several of these motors were refurbished since initial operation, and under the Motor Replacement Program all critical motors have been or will be refurbished.

The ENVY motor program is comprised of all motors greater than 200hp and those smaller that are considered critical to plant operation. These motors are then maintained (tested, repaired, refurbished, and/or replaced) in accordance with the industry standard maintenance basis and practices.

Program Highlights

- Number of 4KVmotors in the ENVY motor program is 34.
- Seven critical to generation 4KV motors have been refurbished under the Motor Replacement Program, and eight critical 4KV motors are of the original vintage.
- Types of motors in program are mostly induction type motors that are greater than 200hp with the exception of the Circulating Water Pump motors, which are synchronous in design.
- System Engineers determine the criticality of the motors within their respective systems.

- Motor Component Engineer determines the condition (health) and required maintenance action of the plant motors.

Overview of the ENVY Motor Testing Program

The ENVY Motor Program is part of the larger PdM Program for the plant rotating equipment. The motors are scanned to determine an adequate heat profile with thermography scans, analyzed for rotational defects with vibration surveys, and undergo lubrication analysis to determine bearing condition and wear. However, the Motor Program also utilizes test equipment that is specifically designed to test the motors' winding insulation system. These tests are conducted specifically to detect motor related faults, i.e., broken rotor bars, voltage imbalance, winding insulation degradation, etc.

For winding insulation testing, the Baker Instruments AWA and Explorer II were purchased earlier this year. This series of test equipment offers a safer and more repeatable means of performing the standard IEEE motor winding tests. It should be noted that the IEEE tests conducted prior to the purchase of the AWA involved several disconnects and reconnects to the motor windings for a series of high voltage tests. These tests can routinely approach voltages of 9KV and require the connection, discharge, and reconnection of several test devices. The Baker AWA greatly reduces the risk of personnel shock since the connection, discharge, and disconnection to the motor occurs only once.

The Baker Instruments Explorer II is an online motor tester and has the capability of detecting several faults during operation that may not have otherwise been detected during the off-line motor testing with the AWA. Two of these faults are current imbalance and voltage imbalance. The Explorer II is a recent improvement in motor testing that offers the advantage of testing a motor without its removal from service. Additionally, the motor can be tested more often. Prior to its introduction to the utility industry all motor winding testing was conducted once the motor was removed from operation, i.e. each refuel cycle.

Introduction of the Baker Instruments AWA and Explorer II into the ENVY Motor Program involved the training of plant Electricians and the PdM Engineer.

The PdMA MCEmax, similar to the AWA and Explorer, offers a series of off-line and off-line motor winding tests within one unit. The PdMA MCEmax is being phased out since the purchase of the Baker units.

Overall the ENVY Motor Program conducts 40 motor winding tests during an outage, and will conduct a similar number of on-line motor tests once the Baker Explorer II is fully implemented into the ENVY Motor Program.

The age of the ENVY motors make them subject to end-of-life types of failures. These motor failure modes are being addressed by the ENVY Motor Program with monitoring and trending of motor testing results, the implementation of improved testing technologies, and the Motor Replacement Program to maintain or regain the forty-year winding life expectancy.

The Motor Replacement Program is an outgrowth of Entergy's Motor Management Program. Within the Fleet-wide Motor Management Program, all Entergy nuclear motors (4KV and greater) were evaluated on a weighted scale (age, environment, service factor, winding temperature, testing results, etc.) to determine and prioritize the motors most in need of funding.

The incorporation of Partial Discharge into the ENVY Motor Program is being investigated. Partial Discharge (PD) is a technology that is designed to detect winding insulation degradation and will primarily address the older motor windings still in use at ENVY.

The motor testing practices at ENVY meet industry standards and practices.

Summary of ENVY Predictive Maintenance Program

In understanding the status of the ENVY Motor Program it is also important to know the basics of the Predictive Maintenance Program (PdM) since the PdM Program is a significant part of the Large Electric Motor Program.

The primary purpose of the Predictive Maintenance Program (PdM) at ENVY is to monitor on a periodic basis the physical parameters of the plant rotating equipment to detect faults and provide maintenance direction prior to the equipment's loss of operational function and availability.

The ENVY PDM Program functions as a matrix organization whereas the PdM Engineer acts as a coordinator in providing the expertise, direction, and analysis capabilities, with the data collection and sampling is done by the plant craft maintenance personnel. The Maintenance Personnel perform the vibration data collection and lubrication sampling. The Maintenance Electricians perform the electrical testing and thermography surveys.

The major capability of the ENVY PDM Program includes the following:

- Vibration Analysis
  - Machinery balance
  - Machinery alignment
  - Bearing analysis
  - Soft-foot and other base or mounting issues
  - Broken rotor bars
- Thermography
  - Machinery overheating and cooling issues
  - Motor Control Centers (MCC) connections
  - Transformer heat profiles and cooling issues
  - Motor connection boxes – overheating connections
  - Steam Traps and system diagnostics
- Lubrication Analysis
  - Anti-friction bearing faults in races, balls, and/or cage

- Sleeve bearing faults due to rubbing, gouging, and excessive wear
- Gears - gearboxes

- Ultrasonic Testing
  - Arcing and sparking of connections at motor connection box, MCC, and extensive use in the switchyard and transformers
  - Anti-friction bearing fault detection on motor, pumps, and fans
- Corona Testing
  - Switchyard
- Dissolved Gas Analysis
  - Transformers
- SF6 Gas Leak Detection
- Electric Motor Testing
  - Standard IEEE based motor testing
  - PdMA MCEmax (being phased out)
  - Baker Instruments AWA
  - Baker Instruments Explorer II

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Table 10: Scheduled PdM Surveys and Sampling

PdM Technology	**Scheduled Tests**
Vibration Collection/Analysis	
Thermography	
Lubrication Samples/Analysis	
Ultrasonic Testing	
Corona Testing	
Dissolved Gas Analysis	
SF6 Gas Leak Detection	
Electric Motor Testing	
Total	

The ENVY Predictive Maintenance program meets industry standards and practices.

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Operations

The ENVY Operations Department has procedures to operate the motors for system/plant use and conducts Operator Rounds for equipment that is available for operational use. Twice per twelve-hour shift the ENVY Auxiliary Operators make their equipment rounds, termed 'operator rounds.' These rounds include the large electric motors discussed in this assessment. The Auxiliary Operator is responsible for recording information such as system pressures – suction and discharge, fluid levels, and general motor condition, i.e., leaks and changes to normal motor operation. Additionally, the Control Room Operator also completes a check-off list that includes motor parameters (motor temperature, motor amperage, speed, etc.). Prior to the start of a motor, the motor is visually inspected by the Auxiliary Operator to ensure a proper level of lubrication in each bearing, adequate cooling water flow (if applicable), and its general physical condition and readiness for operation.

Once per quarter the safety-related motors are operated to ensure their satisfactory operation. During these surveillance tests, motor and system parameters are measured and the determination is made as to system operability.

Within each operational procedure is a precautions and limitations section which typically includes the number of allowable motor starts within a given timeframe. ENVY does include this motor precaution in its procedures; however they are inconsistent in content. In the corrective action searches and interviews this procedural wording inconsistency was not found to have caused any motor damage nor an excessive number of motor starts at ENVY. Investigating this inconsistency will be highlighted as an area of improvement in the conclusion section.

The operation of motors at ENVY is consistent with industry standards and practices with the minor exception noted above.

System Engineering

The System Engineers are responsible for the operability of their system(s). This responsibility includes the overall system functionality including the components within that respective system. In the case of large electric motors, the Motor Component Engineer has the responsibility for the motors. This responsibility overlaps with the System Engineers' for the system. However, it is the Motor Component Engineer that ensures the motor is tested; its parameters are trended, and makes repair determinations to ensure its availability for operation. The Motor Component Engineer keeps the System Engineers informed as to the 'health' of the motors within their systems.

The PdM Engineer distributes a 'watch list' twice per month to the plant management. All plant personnel can access the 'watch list' through the company website. The Motor Component Engineer distributes a Motor Component Health Report which is also available on the ENVY website.

The System Engineers perform a walk down of their respective systems with a checklist. Within that checklist is typically a section for the motors within that system. However, it was observed that not all System Engineers used a checklist during their walk downs.

The interaction of the System Engineers with the Motor Component Engineer at ENVY is consistent with industry standards and practices where a Motor Component Engineer position is utilized.

Corrective Actions

Once an anomaly is found a Condition Report (CR) is written. The anomaly is typically found by the Operations or Maintenance Departments while performing their tasks.

The Public Oversight Panel indicated some concern in the resolution of CRs and its related maintenance action at ENVY. In the sampling of motor related CRs from 2005 to 2008 it was observed that all the motor related CRs were satisfactorily resolved in a timely manner. Each CR is listed with a due-by date and completed date. If a problem persists or is re-occurring, a Root Cause Analysis can be initiated. A Root Cause Analysis is a thorough review of a situation or event to determine the base or root of the problem and determine an appropriate corrective action.

Inspections

There are 3 basic types of inspections of ENVY Large Electric Motors.

- System Engineer walk-downs and Operator Rounds
- FME and Housekeeping Inspections
- PM Inspections

The System Engineer walk downs and Operator Rounds were previously covered. The rounds are a good method of managing the plant assets on a recurring basis as in the case of Operator Rounds, which occurs four times per day.

The FME and Housekeeping rounds offer another good method of inspecting the area and the equipment in that area. Each plant area has a delegated responsible person.

Every six years a major maintenance activity in the form of a Preventative Maintenance (PM) task is performed on the ENVY large electric motors. The following are the tasks performed for each motor undergoing the PM Inspection:

- Disassembly
- Cleaning
- Bearing Inspection
  - Sleeve Measurements
    - Babbitting (if applicable)
  - Anti-Friction Bearing Replacement
- Seal Inspection
- Winding Inspection

  - Cleanliness
    - Leaks (and identification)
    - Residues
  - Looseness/wedge tightness
  - Discharge Tracking
  - Insulation Cracking
  - Leaks and Residue
- Oil Cooler Inspection (if applicable)
- Rotor Inspection
  - Rotor Bars
  - Electrical Testing if necessary
- Air Cooling Inspection
  - Cooling Fan
  - Cooling Vent
- Reassembly
- Functional Checks

The inspections conducted at ENVY are consistent with industry standards and practices.

Long Term Asset Management

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Succession Plan

ENVY Management has exhibited a weak succession plan for the Motor Component Engineer and Predictive Maintenance Engineer positions. With an aging workforce, as with ENVY, it is important to have a quality succession plan to quickly and efficiently bring the following Engineer into these positions. It is noted that several positions at ENVY are unfilled for various reasons, of which, the most repeated reason was the lack of certainty related to license renewal.

The Component Engineer – Motors and Predictive Maintenance Engineer is very knowledgeable and experienced and an effort should be taken to ensure that much of this knowledge and experience is captured or passed on to the next generation. The main source of knowledge management at ENVY that would aid in a position turnover has been the System Engineers Notebook. Many of the notebooks seen at ENVY from various System Engineers were not updated and will be listed as an area for improvement.

The ENVY Succession Plan for the Motor Component Engineer position is below industry standards and practices as discussed above.

General Conclusions

Based on the interviews and the information received during this assessment, it is determined that the ENVY motor program meets industry standards and practices. No major issues were identified that would negatively impact the long-term reliability of the ENVY large electric motors. Likewise, several improvements were noted that would have a positive impact on motor reliability.

However, several minor areas are noted for improvements.

- Ensure the precautions statements in the Operations procedures are consistent in content for number of allowable motor starts per timeframe
- Investigate the use of EMI and Partial Discharge testing for inclusion into the motor testing routine
- Enhance the Succession Plan for the Motor Component Engineer position
- Improve System Engineer Notebooks and ensure that they are updated on a regular basis
- Ensure the use of checklists during the engineering walk-downs

References

Procedures

1. ENVY Procedure – OP 5235, AC and DC Motor Maintenance
2. ENVY Procedure – AP 0211 Rev. 6, Predictive Maintenance Process
3. ENVY Procedure – EN-DC-344 rev. 0, Motor Program Governance Document
4. Entergy Corporate Procedure - EN-DC-344, Rev. O, Motor Program Governance Document
5. ENVY Procedure – OP 2110 rev. 74, page 13 of 54, precaution 21 Normal Starting (Reactor Recirculation, Motor Starts).
6. ENVY Procedure – OP 2124 rev. 113, page 13 of 124, precautions 1, 2, (RHR, Motor Starts).
7. ENVY Procedure – OP 2123 rev. 40, page 6 of 26, precaution 1, (Core Spray, Motor Starts).
8. ENVY Procedure – OP 2181 rev. 109, page 11 of 60, precaution 2, (Service Water, Motor Starts).
9. ENVY Procedure – OP 2172 rev. 45, page 11 of 36, precaution 14, (Feedwater, Motor Starts).
10. ENVY Procedure – EN-QV-121 rev. 2, “Supplier Qualification/Maintenance of Qualifications.”
11. ENVY Procedure – EN-QV-123 rev. 2, “Supplier Audits/Surveys.”

Documents

1. Response to ER Request 05-0440, “Develop a Motor Rewind Program for Large Electric Motors, Dated 5/10/05.
2. Predictive Maintenance Watch Report, November 8, 2008.
3. Document request 281, “Component Health Report – Large Motors,” 2nd Quarter 2008.

4. PM Basis Document – ME-022 rev. 8.
5. PM Basis Document – ME-030 rev. 5.
6. EPRI Technical Report – "Aging Assessment Field Guide."
7. MEC-05-001 rev. 4 – "AC/DC Motor Maintenance Lesson Plan," June 2007.
8. OJT/TPE Guide 705008 rev. 10, "Inspect and Maintain Electric Motors," October 2006.
9. OJT/TPE Guide 705004 rev. 0, "Inspect and Maintain Large Electric Motors," December 2003.
10. OJT/TPE Guide 717703 rev. 16, "Use of Maintenance Test Equipment," September 2008.
11. Document request 186, "List of Spare Motors Onsite and Offsite," September 2008.
12. "ENVY – 4KV Motor Rewind Action Plan," June 2006.

2.12. Flow Accelerated Corrosion (FAC) Program

Introduction and Methodology

This technical focus area was evaluated based on a review of documentation associated with core processes, procedures, and interviews with the FAC Program Engineer, Engineering Programs Supervisor and the Components and Programs Manager. The evaluation included a review of eddy current testing and results from the October 2008 RFO 27 outage inspections. The programmatic application of the CHECKWORKS software program in support of the FAC Program was also evaluated.

FAC Program Description

The ENVY Flow Accelerated Corrosion (FAC) Program is implemented in accordance with procedure EN-DC-315. This procedure provides overall programmatic guidance for the FAC program at ENVY and across the Entergy Fleet. The program is focused on ensuring that all plant systems and piping which are susceptible to the effects from Flow Accelerated Corrosion are effectively monitored. It should be noted that all feedwater shells have been replaced with FAC resistant materials.

Observations

Over the past two to three years there have been three individuals who have functioned as the FAC Program Engineer, with the current individual being in position for the past eight months. Station management has recognized the need to provide accelerated training for him and has utilized the services of an external company (ENERCON) in support of training and mentoring support. This has consisted of on-site as well as off-site support leading up to and including RFO 27. In addition, the original FAC Program Engineer who has since left the company and is credited with developing and implementing the ENVY FAC Program, is often contacted for support which he provides. A review of the current engineer's Engineering Qualification Card per ENN-TK-ESPG-042 indicates he is in compliance with the procedural requirements per EN-TQ-104 *ESP Initial Training*.

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Flow Accelerated Corrosion Program Effectiveness

The topic of whether the FAC program utilized the most current version of the CHECKWORKS Software program over the past 2-3 years was discussed. The program engineer stated via CR VTY-2006-02699 (August 30, 2006) that the CHECKWORKS predictive models were not current at the time this issue was identified per the CR, as an administrative issue with respect to the FAC Program and had no effect upon plant equipment or operations. One month after CR VTY-206-02699 was issued, the CHECKWORKS predictive models were updated (Ver. 1.0f) and the CR closed out.

On December 14, 2007 the VY CHECKWORKS SFA Conversion was approved per the *Vermont Yankee CHECKWORKS SFA Conversion Report* Doc # 0719-03. This report resulted in the implementation of CHECKWORKS SFA 2.2 which is currently in use today and considered the latest approved version.

In addition, the issue currently being addressed by the Atomic Safety and Licensing Board (ASLB) per ASLBP No. 06-849-03-LR, the ASLB issued an order on November 24, 2008 concluding that Entergy has demonstrated that the effects of aging for FAC will be managed for the period of license extension. The contention material and findings by the ASLB were reviewed and it was concluded by the NSA team that this will adequately addressed as part of the ASLB review.

RFO 27 Results

In RFO -26 a 'shiny area' was located on Feedwater Heaters 5A/B near the tube to tube-sheet region (CR 2007-01993). The corrective action was to inspect in RFO 27 to determine/assess wear rate based on thinning/pitting and eddy current testing. It was concluded as a result of inspections during RFO 27 that no wear was evident based on the inspection results. In addition, the fact that the FW Heaters are made of chrome-molly significantly mitigates the potential for FAC.

During RFO 27 the following Feedwater Heaters had shell side inspections performed:

- 1 A/B, 2 A/B, 4B.

No damage or material degradation was found. No tube 'brightness' indicative of erosion or pitting was found similar to that found in RFO 26 inside 5A/ 5B FWH (CR 2007-01993). The shell side inspections were performed on the 1 and 2 heaters as part of the normal, periodic inspection process. The 4B heater was inspected as the heater was suspected to have a tube to shell leak; none was found.

The following Feedwater Heaters were eddy current tested:

- 4B, 5A/B

The 4B heater was inspected due to a suspected leak. A complete eddy current inspection was performed and no indications of a leak were found. The 5A/B feedwater heaters were inspected as part of the follow-up actions to CR 2007-01993. Tubes potentially susceptible to erosion/pitting discussed above were tested for indications of accelerated thinning and pitting neither heater had any indication of tube thinning or pitting (per iTi, integrated Technologies, Inc., PR No. 34-37 and 34-39; preliminary report). A single leaking tube was found on the 5A heater and was subsequently stabilized and plugged. All feedwater heaters (1-5) were deemed satisfactory for return to service, with normal periodic inspections per the ENVY PM program planned for RFO 28.Based on the RFO 27 observation of 5A/B an 'extent inspection' of FWR 1A/1B, 2A/2B and 4B were conducted with no similar conditions noted.

Additional Areas of Interest

During the time period in which this reliability assessment was conducted there was an ongoing investigation referred to as NEC Contention 4: Flow- Accelerated Corrosion". Testimony report ASLBP No. 06-849-03-LR was reviewed but due to the confidentiality of the issues and personnel involved, ENVY notified the audit team that it would not be providing information associated with this issue until the investigation/testimony had been completed.

General Conclusions

To date there are no indications that accelerated flow rates due to the Extended Power Up-rate are resulting in FAC that could impact equipment reliability. Future continuation of current testing methods should be monitored to ensure that no FAC issues develop over time.

In conclusion, based on the information received/reviewed during this assessment, all indications reflect that the current FAC program meets industry standards and is in compliance with applicable station procedures.

Document References

1. EN-DC-315 VY: Flow Accelerated Corrosion Inspection Program Procedure
2. VY-RPT-05-00012 Vermont Yankee Piping and Flow Accelerated Corrosion Inspection Program
3. ENN-TK-ESPG-042: Engineering Qualification Card

4. EN-TQ-104 “ESP Initial Training”
5. EN-DC-329 “Engineering Programs Control and Oversight”
6. iTi integrated Technologies, Inc. PR No. 34-37 & 34-39; preliminary report
7. NEC Contention 4: Flow- Accelerated Corrosion”. Testimony report ASLBP No. 06-849-03-LR

2.13 Reactor Building Crane/Hoists Maintenance and Testing - Technical Review

Introduction and Assessment Methodology

The NSA team reviewed the Reactor Building crane failure that occurred on May 12, 2008 when lowering a loaded transfer cask onto the refueling floor. The NSA team evaluated this event, and performed a limited review to understand the extent of issues associated with Cranes/Hoists at the Vermont Yankee Nuclear Facility that could impact future plant reliability. This included a review of related CRs pertaining to hoists and cranes, various report documents, evaluation of the use of Operating Experience (OE) by ENVY personnel pertaining to cranes/hoists and ENVY personnel interviews. The list of specific questions (email letter "Crane Items" dated 10/30/08) asked by the Vermont State Public Oversight Panel is referenced in the Document References within this report section and responses are provided herein.

Technical Focus Area Description

The NSA team technical focus included a detailed review of the Reactor Building crane failure, and evaluation of industry OE associated with the lifting and loading of fuel into casks for interim storage, and horizontal review of all CRs related to ENVY Plant cranes and hoists.

Observations

During discussions with ENVY personnel, it was determined that in 1981, ENVY TECH SPECS (technical specifications) related to the Reactor Building crane were changed. This change appropriately removed from TS 4.12.G.1 any reference to a specific year of standard ANSI B30.2. Added to TS bases was reference to ANSI B30.2-1976 Overhead and Gantry Cranes**.**

Also discussed with ENVY personnel was the decision in February 2007 not to load test the Reactor Building crane prior to ENVY's first spent fuel storage campaign (i.e. loading spent fuel from the spent fuel pool into dry casks for placement on an Independent Spent Fuel Storage Installation (ISFSI)). Design Engineering personnel and Project Management personnel decided not to test the crane because the weight of a loaded cask, 97.9 tons, was less than the crane rating and a load test was not required per ANSI B 30.2. They stated that there is no specific document such as an Engineering Evaluation that documents this decision. This particular question, as well as others, were part of preparation for the first ISFSI campaign, and would have been part of ISFSI project meetings and/or action item list(s). To obtain approval for an ISFSI ENVY was required to submit a 10 CFR 72.48.212 report to the NRC. This report was subsequently approved by the NRC. Included in that report are several evaluations/discussions including capacity and single point failure vulnerability. One of which discusses the Reactor Building crane, which primarily focuses on providing the background regarding if the crane is single failure proof, crane capacities, etc.

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General Conclusions

The maintenance program for the Reactor Building crane meets industry standards. However implementation has been less than acceptable as discussed below.

The significance of less than adequate Reactor Building crane performance is that future reliability of the plant could be impacted if the crane is not available or fails during the performance of a spent fuel storage move or other outage activity. The quantity and trend of plant crane and hoist related Condition Reports, (e.g. Reactor Building Crane CRs), indicates that issues exist with effectively implementing the intended Crane/Hoist Program.

It is recommended that the current focus on resolving crane related issues at the ENVY station continue until actual performance improves over the long term. This should include, during review of the Reactor Building Crane MR (a) (1) recovery plan, consideration to expedite corrective actions e.g. Crane control system upgrades. Also continuing to monitor and trend minor crane/hoist related issues is recommended.

Document References

- Letter “Crane Items” dated 10/30/08 from the Vermont State Oversight Panel
- ANSI B30.2-1967 Overhead & Gantry Cranes
- ANSI B30.2-1976 Overhead & Gantry Cranes
- TriVis Assessment of the ENVY Reactor Building Crane & Heavy Loads Evaluation Report.
- CR-VTY-2008-2043 Reactor Building Crane Malfunction
- CR-VTY-2008-04802 Reactor Building Crane Main Disconnect Switch Failure
- VYSE-MRL-2007-024 REV 2 Maintenance Rule Performance Evaluation/Action Plan for Hoist System
- Buildings System Health Reports; 3rd QTR 2007, 4th QTR 2007, 1st QTR 2008

2.14 General Conclusions

The following are the overall conclusions for the six selected systems in the areas of: Design Bases (Criteria 1, 2, 3, 9, 10, 11), Equipment Reliability (Criteria 5, 6, 7, 8), Operations and Training (Criteria 4 & 12), and the Corrective Action Program (Criteria 13). The overall conclusions for the Technical Focus Areas are also provided below.

Design Bases

The assessment confirmed that for plant modifications, installation, design and procurement activities the processes and procedures are in place at ENVY to assure the conformance of the facility to its design and licensing bases, including its operation at the uprated power level. The assessment also confirmed that for the reviewed systems, the processes and procedures are being appropriately implemented. It also determined that the EPU License Amendment request appropriately considered the licensing and design bases of the plant.

The assessment reviewed the current General Design Criteria applicable to new plants and determined that the ENVY original and current design met the intent of those criteria. The assessment also reviewed ENVY's process for controlling margin changes, including margin changes associated with the EPU Program. For the systems reviewed, the assessment determined that changes to regulatory requirements were properly controlled and implemented, that adequate margins have been maintained for the uprated power level for the proposed period of license extension.

Equipment Reliability

The scoping and equipment criticality analysis performed for each of the selected systems was consistent with industry good practices. The process is documented and well controlled.

The bases for each testing and Preventive Maintenance (PM) task to be performed for important components within the selected systems was consistent with industry good practice, well documented and easily accessible by the System Engineers.

Testing and inspections that are required as per the maintenance bases documents for each system were typically implemented and performed within the designated time periods consistent with industry good practices

The current overall performance monitoring process at ENVY for the six selected systems meets industry standards with the following exceptions:

- Inconsistencies exist for system performance monitoring, system walk downs, and system notebooks.
- The communications between component engineers and System Engineers is sometimes informal and relies on experience and relationships and is not driven by program/procedure adherence.
- The individual component health reporting process is not automated and therefore case histories are not easily shared across the ENVY site and/or the Entergy fleet.

- The ENVY site manages system health in a matrix approach where the System Engineer, component engineer, PM lead, and program engineers have responsibility for special processes but no distinct single point accountability exists for ensuring system health. This matrix approach is currently effective for the six systems because system, component, and program engineers are experienced. As experience levels decrease due to attrition and retirements, it will be more difficult to ensure system health with this matrix approach. Industry top performers have implemented the system manager concept, which emphasizes single point accountability for overall system health.

The Long Term Asset Management processes at ENVY include refurbishment and replacement strategies, aging & obsolescence management and capital budgeting. Many fleet standard processes and procedures relating to Long Term Asset Management have been recently developed. The implementation of some of these processes/procedures as they apply to the six selected systems is not yet complete at ENVY. Some application inconsistencies were observed across the six systems.

Operations

Based on interviews, review of processes, procedures and other documents, and review of responses to unanticipated plant conditions and events for the 6 systems reviewed, the actions taken were assessed to meet industry standards. These actions included implementing compensatory and corrective actions, completing appropriate documentation and making procedures changes.

Training

Based on interviews, review of processes, procedures and other documents, and review of responses to plant modifications for the 6 systems reviewed, the actions taken were assessed to meet industry standards. These actions included revising operations procedures and training materials, modifying the simulator and conducting training.

Corrective Actions

An overall assessment of the corrective action program and its effectiveness was completed and is documented in section 1.2.5.1 that addresses the criterion 13 (Corrective Action Program) questions for the overall Corrective Action Program. The six system Condition Report reviews support the conclusion of Section 1.2.5.1 Corrective Action Program.

Technical Focus Areas

As part of this assessment, seven technical focus areas were reviewed. These areas were:

1. Electrical System: Back up & Standby
2. Primary Containment System
3. Underground Piping Program Evaluation
4. Cable Separation Practices
5. Large Electric Motor Program Horizontal Review

6. Flow Accelerated Corrosion (FAC)
7. Reactor Building Crane

Five of the Technical Focus Areas reviewed were determined to meet industry standards. The Electrical System: Back up & Standby, the Primary Containment System, the Underground Piping Program, and the Flow Accelerated Corrosion Program (FAC) all meet industry standards and provide reasonable assurance that long term plant reliability will be maintained. Details can be found in Sections 2.7, 2.8, 2.9 and 2.12.

Two of the Technical Focus Areas reviewed meet industry standards with minor exceptions. The Cable Separation Program, while meeting industry standards, could be improved if ENVY transferred their current cable separation database into an industry accepted computer data management system. Their program would then be more in line with industry best practices. Details can be found in Section 2.10. The Reactor Building Crane is maintained in accordance with industry standards except the implementation of maintenance practices and the learning process associated with equipment issues can be improved. This improvement will enhance the long-term reliability of the reactor building crane. Details can be found in Section 2.13.

Appendix A: Specific Responses to 5 Assessment Goals and Objectives

In addition to the Principal Conclusions included in the Executive Summary, NSA was also tasked with addressing five specific goals and objectives identified in Act 189. These areas were relative to conformance of the facility; effectiveness of plant processes and programs; and any operational shortcomings. NSA was then to draw conclusions on overall performance. The following is NSA's response to these five areas:

Goals and Objectives

1. Assess the conformance of the facility to its design and licensing bases, for operating up to 120 percent of its original intended power production level, including appropriate reviews at the plant site and its corporate offices.

   The assessment confirmed that the processes and procedures are in place at ENVY to assure the conformance of the facility to its design and licensing bases, including its operation at the uprated power level. The assessment also confirmed that for those systems reviewed, those processes and procedures are being appropriately implemented. This confirmation is based on document reviews and interviews conducted at the plant site and corporate offices which determined that the design and licensing bases is well documented, controlled, and properly reflected in plant records. It also determined that the EPU License Amendment request appropriately considered the licensing and design bases of the plant.

2. Identify all relevant deviations, exemptions, or waivers, or any combination of these from any regulatory requirement applicable to ENVY and from any regulatory requirement applicable to new nuclear reactors; and, verify whether adequate operator margins are retained despite the cumulative effect of any deviations, exemptions, or waivers for the present licensed power level for the proposed period of license extension.

   The assessment reviewed the processes and procedures that control deviations, exemptions and waivers from regulatory requirements at ENVY. The assessment also confirmed that for the reviewed systems, those processes and procedures are being appropriately implemented. For two systems, HPCI and Service Water, NAS reviewed the current General Design Criteria applicable to new plants and determined that the ENVY original and current design met the intent of those criteria. The assessment also reviewed ENVY's process for controlling margin changes, including margin changes associated with the EPU Program. For the systems reviewed, the assessment determined that changes to regulatory requirements were properly controlled and implemented, and that adequate margins have been maintained for the uprated power level for the proposed period of license extension.

3. Assess the facility's operational performance, and the facility's reliability for continued power production, giving risk perspectives where appropriate.

   The ENVY site has been a reliable performer throughout its operating life. The plant has been operated consistent with industry standards. The NSA team believes current level of reliability could be maintained through an extended operating period, based on the management policies, processes and guidance currently in place. However, some specific challenges to plant reliability are identified in this report. These warrant additional management attention to ensure future reliable operations.

4. Evaluate the effectiveness of licensee self-assessments, corrective actions, and improvement plans.

   Department Managers establish assessment schedules and ensure that assessments and associated corrective actions are performed in a quality and timely manner. The Corrective Action and Assessment Manager develops and maintains a consolidated site schedule of committed assessments and assures that applicable information is entered into the Plant Condition Report System (PCRS) to support tracking and trending. The self-assessment program utilizes focused and snapshot assessments to address problem areas and adverse trends. Corrective actions and improvement plans from self-assessments are entered into PCRS.

   All plant system issues are included in the corrective action program. Issues are entered as Condition Reports (CRs) and tracked to closure in PCRS. The type of causal analysis performed and corrective action taken is dependent on the significance of the issue. Trend analyses are preformed and corrective action taken with respect to any identified adverse trends. All issues, corrective actions, and improvement plans are entered in and tracked via PCRS.

   The NSA team has concluded that the ENVY's self-assessment process, Corrective Action Program, and associated improvement plans meet industry standards.

5. Determine the cause or causes of any significant operational shortcomings identified and draw conclusions on overall performance.

   This objective is addressed in the content of the Executive Summary.

Appendix B: Benchmarking Report

Vermont Yankee Benchmark Report

NSA was asked by The Vermont Department of Public Service to perform a benchmark study of key performance measures comparing Vermont Yankee Nuclear Station (ENVY) to other nuclear plants.

As part of a response, NSA identified three (3) groups of performance data:

- Unit Performance Data
- Organization Staffing Data
- Equipment Reliability Data

In addition to industry comparisons, The Vermont Department of Public Service has request a comparison to 'sister plants' which they identified. Eleven plants were identified representing (15) units. Eight (8) responded with data representing eleven (11) units. The sister plant list is shown in Section 1.4, Attachment 1.

The project had three objectives:

- Present industry median, and highest and lowest quartiles for each of the indicators.
- Present sister plant median, and highest and lowest quartiles for each of the indicators.
- Compare these industry and sister plant levels to those at Vermont Yankee.

NSA used the following approach in creating this report:

1. Assemble industry nuclear data for year-end 2006
2. Collect sister plant data
    a. Requested through Vermont Yankee, and presented to NSA anonymously through a ENVY contractor.
    b. Data was presented for 2005, 2006, 2007 and October 2008.
    c. Where dual unit data was presented, the Organization Staffing data was divided by a factor 1.5 to get an equivalent single unit comparative data point.
3. Engage Navigant Consulting to provide industry nuclear staffing data
    a. Proprietary industry data based on ¾ of the US nuclear plants reporting.
    b. Most plant data is for year-end 2007.
4. Establish seven subsets of industry groups (Section 1.4, Attachment 2) from which to develop comparisons
    a. Compute median, and highest and lowest quartile levels for the key performance measurement area
5. Obtain Vermont Yankee data
    a. Self reported by Vermont Yankee
    b. Data was presented for 2005, 2006, 2007 and October 2008.
6. Prepare this report
    a. Compares the Vermont Yankee data to that of select industry groups in each of the key performance measurement areas

Unit Performance

Benchmark charts are provided below. Most charts have two comparative indicators: an industry benchmark and a sister plant benchmark. Both show a median (middle unit), lowest quartile (1/4 of the units have values below this level), and highest quartile (1/4 of the units have values greater than this level). The charts are defined in Section 1.4, Attachment 3.

The industry benchmarks are comprised of data from relevant industry groups (e.g., All Units, or Single Site Units, or All BWR Units; see Section 1.4, Attachment 2, for definitions).

The sister plant benchmark is comprised of data that is averaged from 2005 to October 2008. The sister plant data was collected by Vermont Yankee through its contractor.

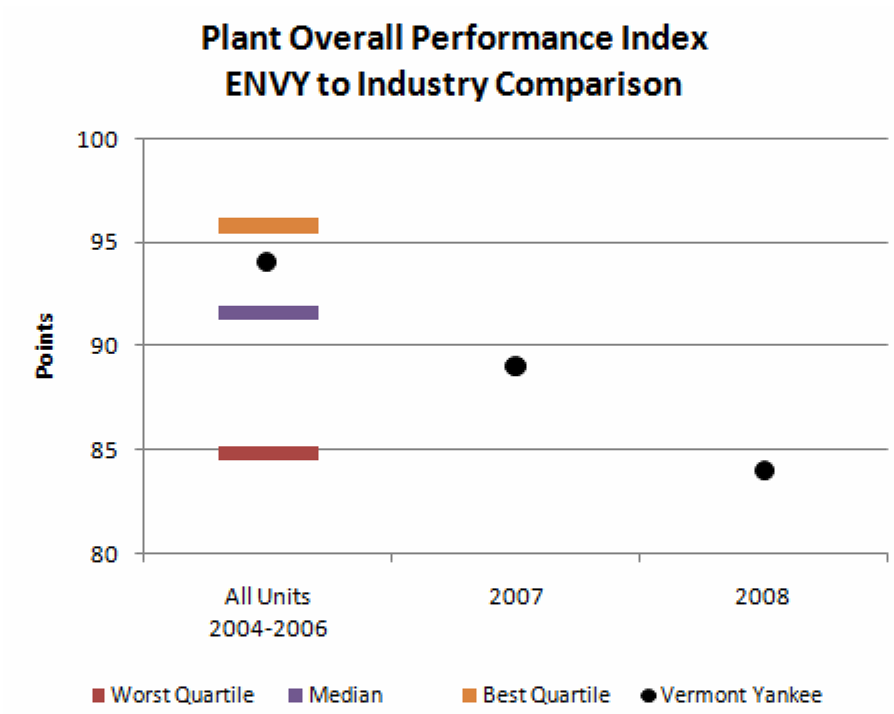

Figure 1 – Plant Overall Performance Index (2 year rolling average)

ENVY performs between median and top quartile as compared to the industry through 2006. However, in 2007 and 2008, ENVY plant overall performance index has trended down to approximately 84%. Factors that are influencing the downward trend include capability factor, forced loss rate, radiation exposure, chemistry index, and industrial safety.

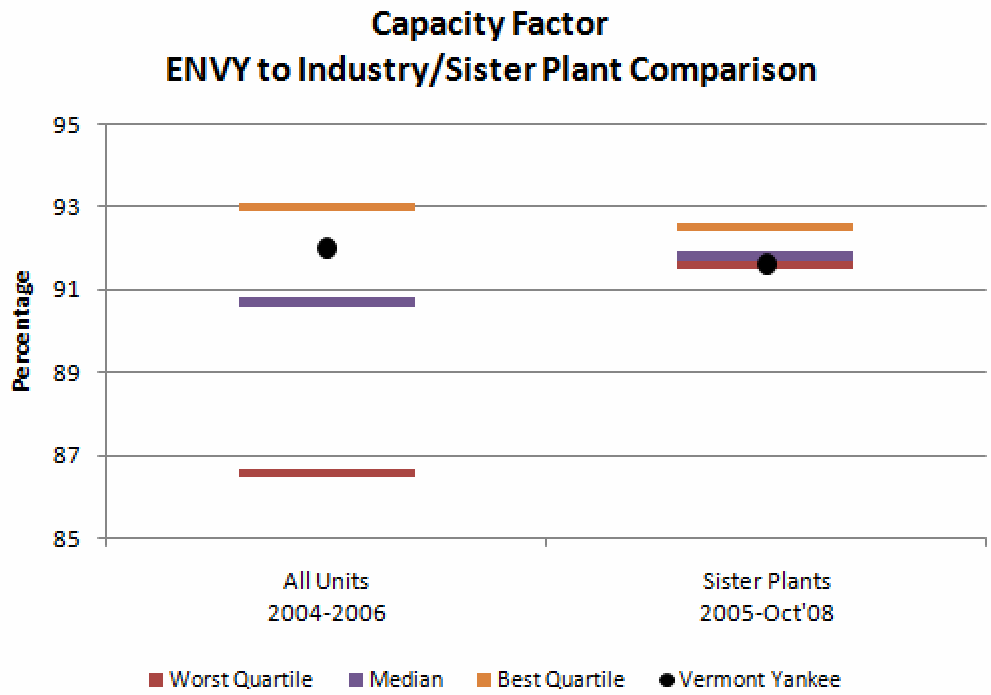

Figure 2 – Capacity Factor

Plant capacity factor at ENVY falls between median and top quartile performers as compared to industry and is consistent with the median sister plant performer.

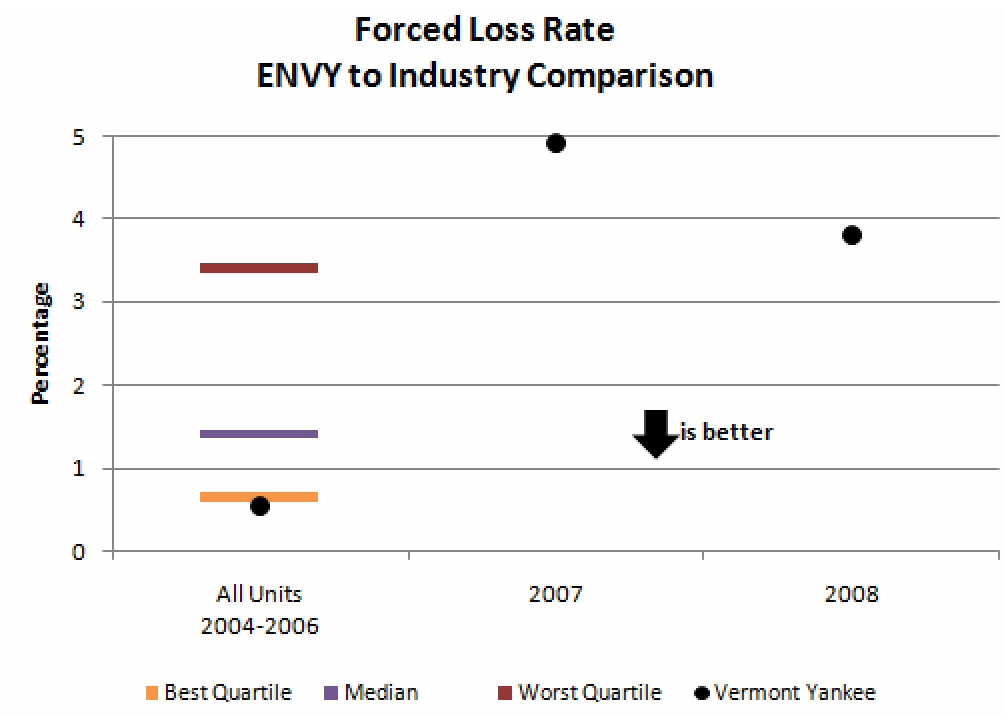

Figure 3 – Forced Loss Rate

Through 2006, ENVY was a top quartile performer in forced loss rate as compared to the industry. A few events in 2007 and 2008 have resulted in an increased forced loss rate.

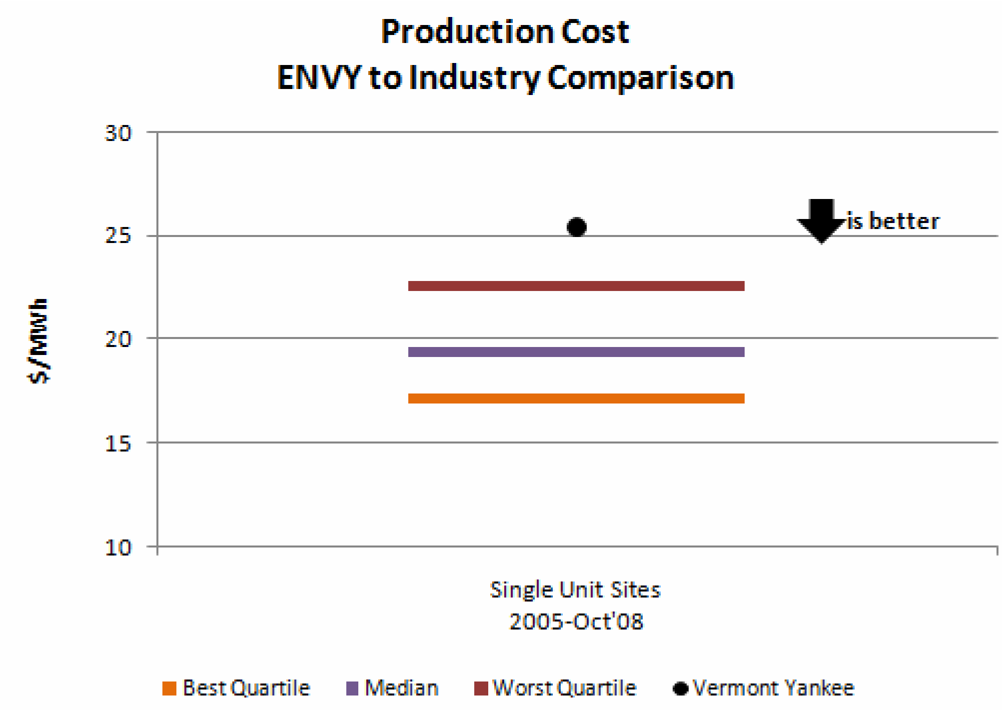

Figure 4 – Production Cost

ENVY production costs in dollars per MWhr have been historically high compared to the industry single unit sites.

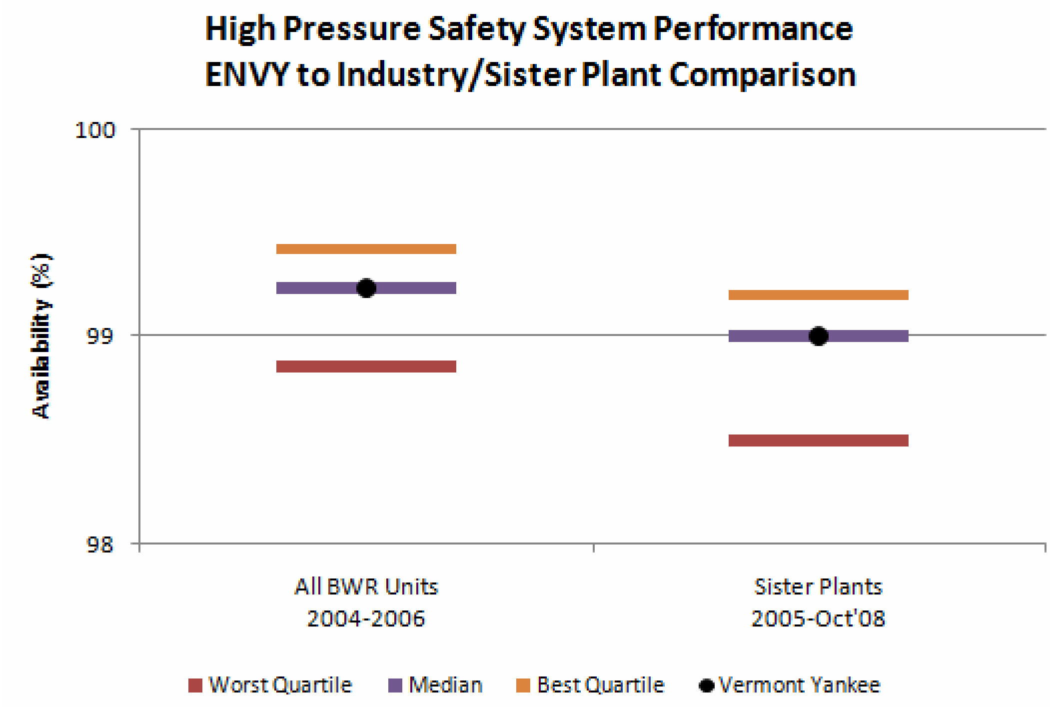

Figure 5 – High Pressure Safety System Performance

High pressure safety system performance at ENVY is consistent with the median performer as compared to both overall industry and sister plants.

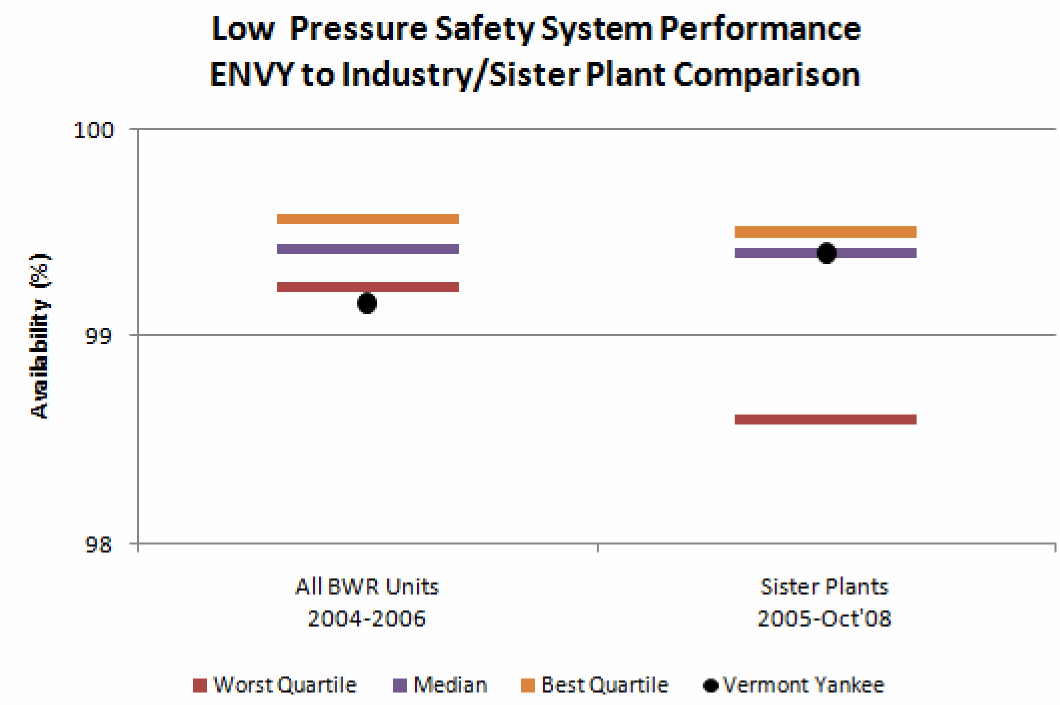

Figure 6 – Low Pressure Safety System Performance

Low pressure safety system performance at ENVY is consistent with median sister plant performers.

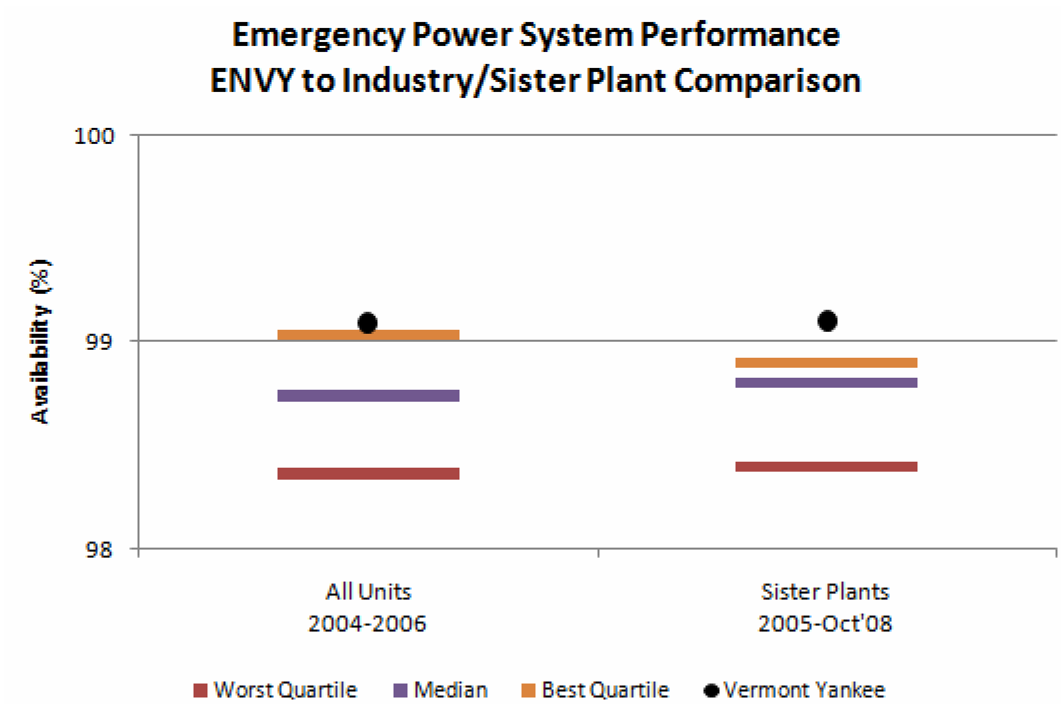

Figure 7 – Emergency Power System Performance

Emergency power system availability at ENVY is consistent with industry good performers.

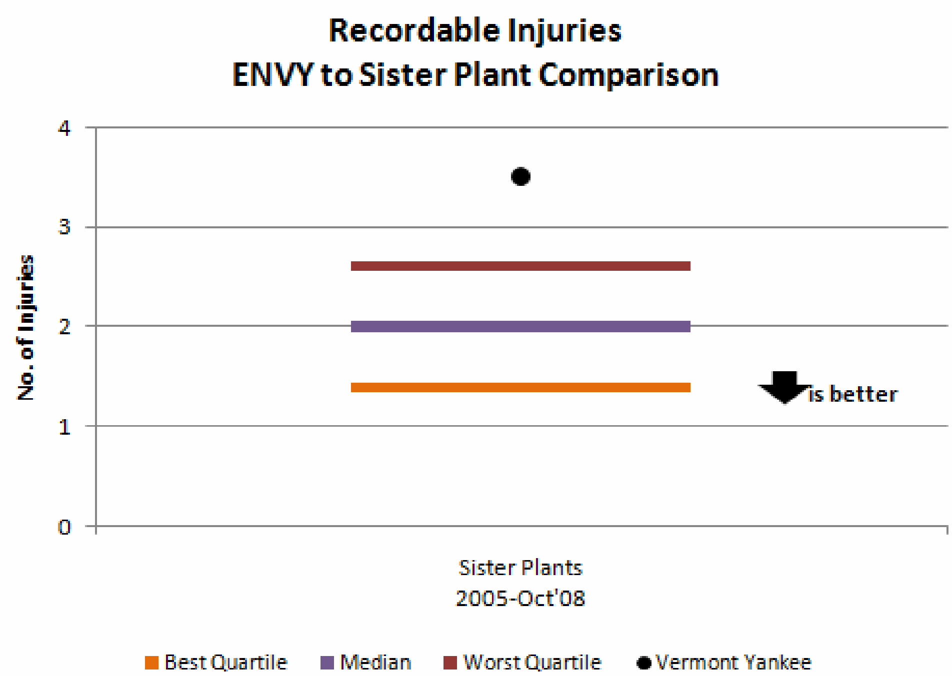

Figure 8 – Recordable Injuries

Industrial safety at ENVY is a watch area because they are bottom quartile performers as compared to the sister plants.

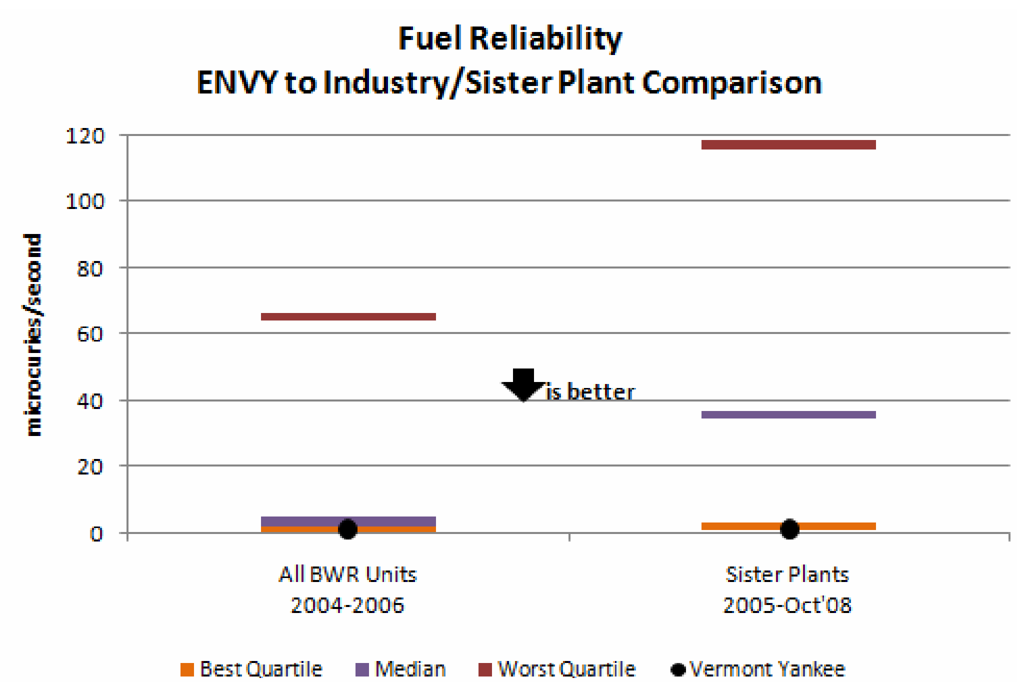

Figure 9 – Fuel Reliability

Fuel reliability at ENVY is consistent with industry best performers.

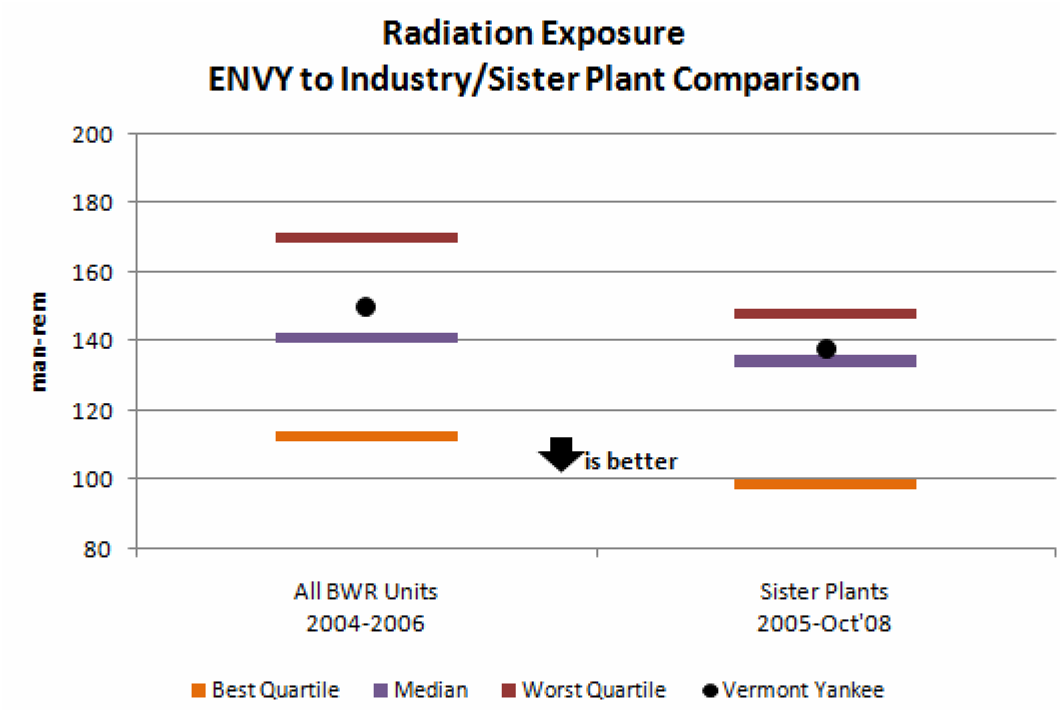

Figure 10 – Radiation Exposure

Radiation exposure at ENVY is just below median performers as compared to industry and sister plants.

Organization Staffing

Benchmark charts are provided below. Most charts have three comparative indicators: two industry benchmarks and a sister plant benchmark. The indicators show a median (middle unit), lowest quartile (1/4 of the units have values below this level), and highest quartile (1/4 of the units have values greater than this level). The charts are defined in Section 1.4, Attachment 4.

The industry benchmarks are comprised of data from single unit BWR and PWR plants (see Section 1.4, Attachment 2, for definitions). The data was provided by Navigant Consulting, and represents staffing levels as of 2007 for most units.

The sister plant benchmark is comprised of data collected by Vermont Yankee through its contractor, and represents staffing levels as of October 2008.

The Vermont Yankee data was self reported and reproduced on each benchmark indicator, and represents staffing levels as of October 2008.

Figure 11 – Plant Staffing

Staffing numbers indicate slightly lower than industry comparable units at ENVY. However, ENVY data may not accurately reflect use of long term contractors at the site (as per the sister plant reported data).

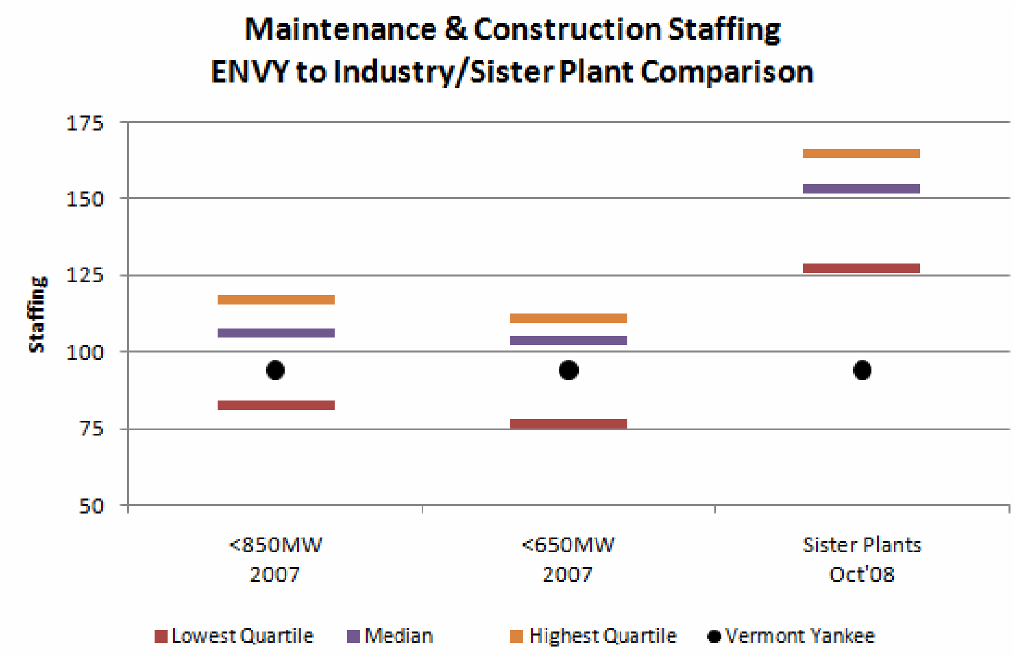

Figure 12 – Maintenance & Construction Staffing

Maintenance and Construction staffing at ENVY is 30 to 50 percent lower than sister plants. However, ENVY data may not accurately reflect the use of contracted personnel.

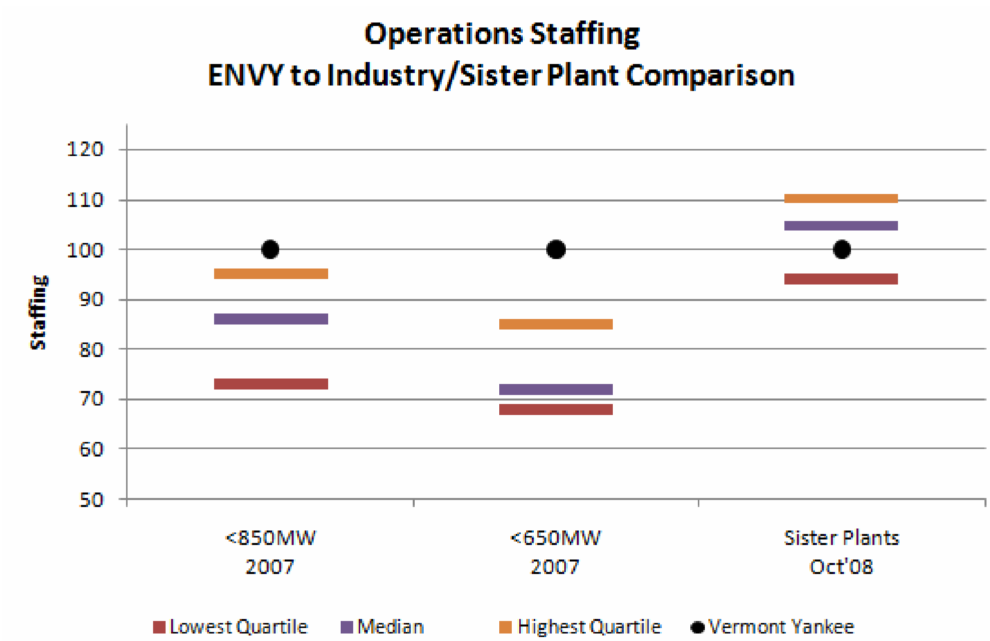

Figure 13 – Operations Staffing

Operations staffing levels at ENVY are above the majority of the industry peers and in the middle range as compared to the sister plants.

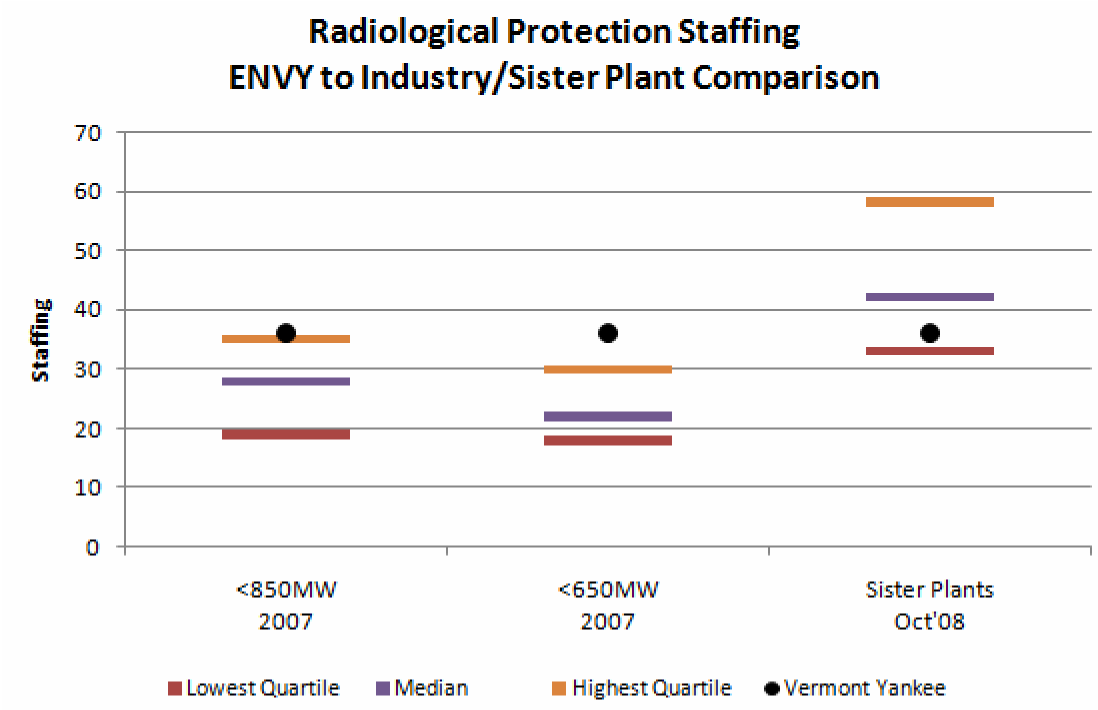

Figure 14 – Radiological Protection Staffing

ENVY has adequate radiological protection staffing as compared to industry peers. However, ENVY data may not adequately reflect use of long term contractors.

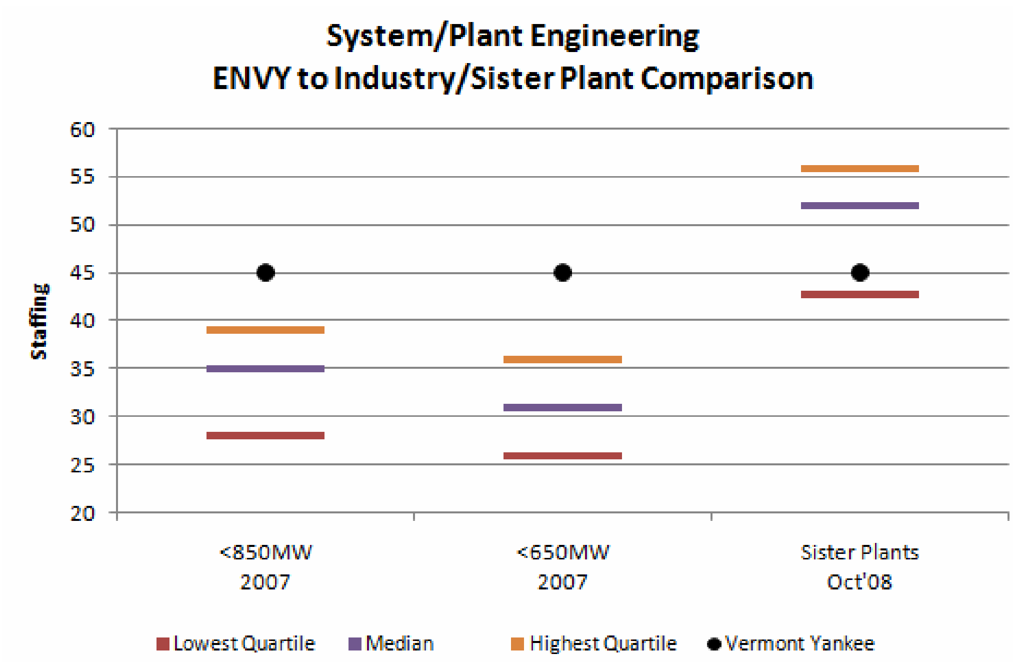

Figure 15 – System/Plant Engineering

System/Plant engineering staffing levels at ENVY are high compared to industry peers and at the lower end of the range as compared to the sister plants.

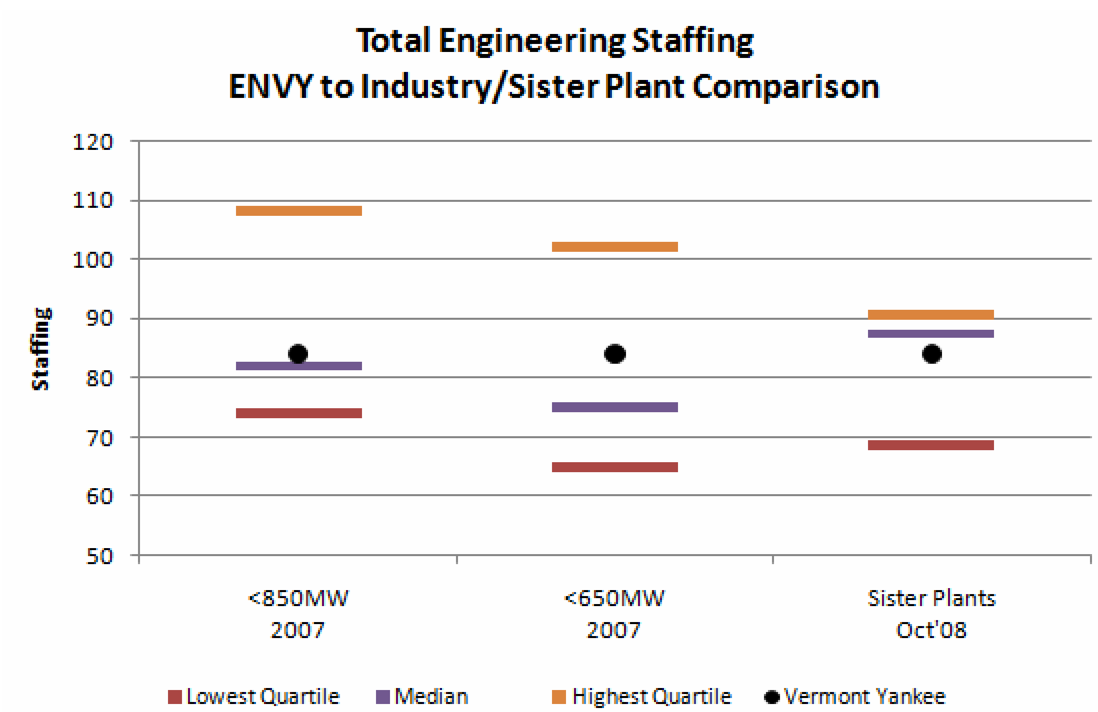

Figure 16 – Total Engineering Staffing

Total engineering staff levels at ENVY are consistent with industry peers.

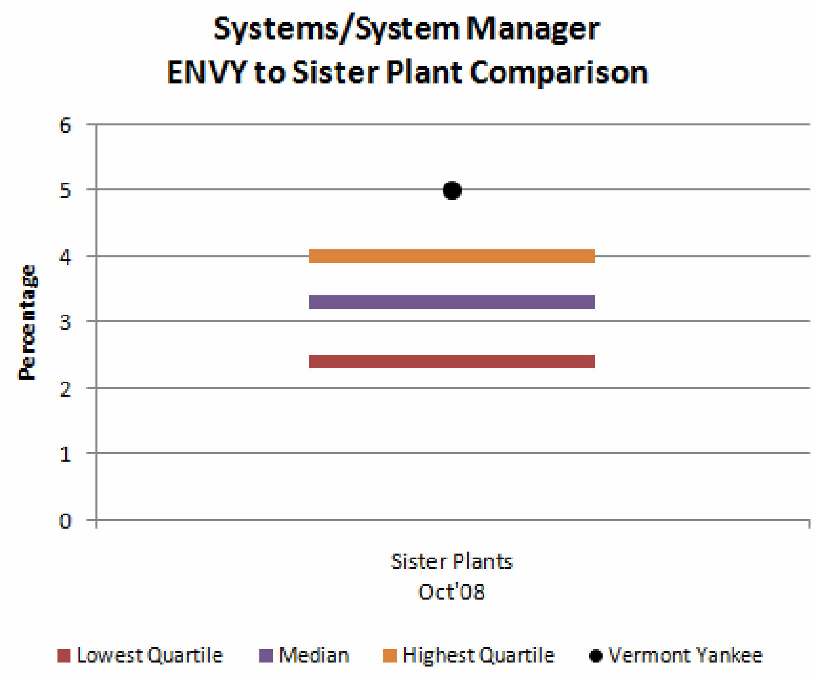

Figure 17 – Systems/System Manager

ENVY has 19 slots billeted for System Engineers and the average number of systems per System Engineer is 5. This is among the highest level of systems per System Engineer across all sister plants

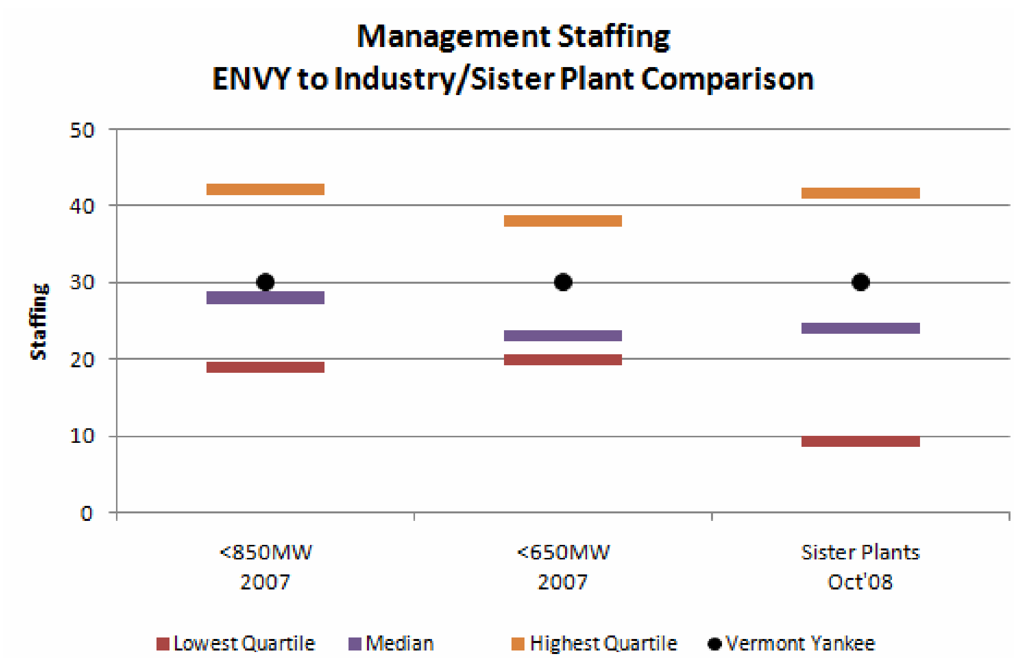

Figure 18 – Management Staffing

Management staffing at ENVY is slightly above the median levels of industry peers.

Equipment Reliability

Equipment Reliability Index (ERI) Performance Metrics

The Nuclear Power Industry has worked together to define and standardize the Equipment Reliability (ER) processes and definitions. This standardization provides for reliable, transparent ER comparisons to industry best practices. As part of this standardization effort, the Nuclear Industry's "Equipment Reliability Working Group", comprised of most of the US Nuclear Generators, has created standard ER Performance Metrics referred to as the "ER Index" (ERI). This Index includes 19 Leading (forward looking) and Lagging (backward looking) performance indicators that are collected and reported in a standard method by a majority of US Nuclear Power Generating Companies. The chart shown in Section 1.4, Attachment 5, is an example of this Index referencing specific ERI indicators.

The performance indicators are separated into Leading and Lagging indicators for the purpose of providing insight to both historical and future plant equipment reliability.

ER performance data is collected periodically by each utility utilizing a standard performance criteria spreadsheet. Performance values associated with each indicator contribute to an overall ERI point value ranging from 1 to 100 points. The ERI point value is then assigned a color based on predefined criteria; Red (< 50 points), Yellow (> or = 50 and < 65 points), White (> or = 65 and < 85), or Green (> or = 85)

It is also possible to determine the quartile in which a utility ranks. The results from each utility are then compiled and reported back to the participating members in a transparent for comparison purposes.

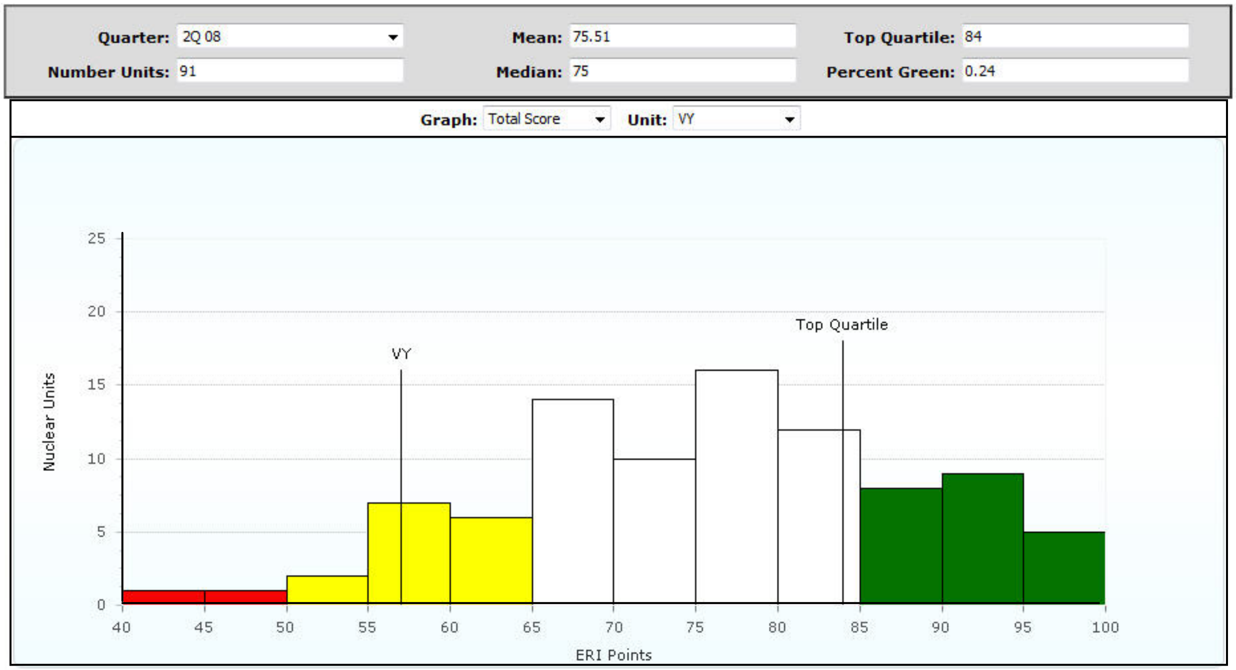

Figure 19 – ENVY vs. Industry Equipment Reliability Performance

The graph in Figure 19 depicts the reported 2nd Quarter 2008 ERI metrics as compiled from 91 U.S. Nuclear Units. The graph also indicates the score for the Vermont Yankee (ENVY) facility, as well as its Quartile reference. Vermont Yankee has an overall ER Index of 57 which is in the yellow region, or bottom quartile of industry performance.

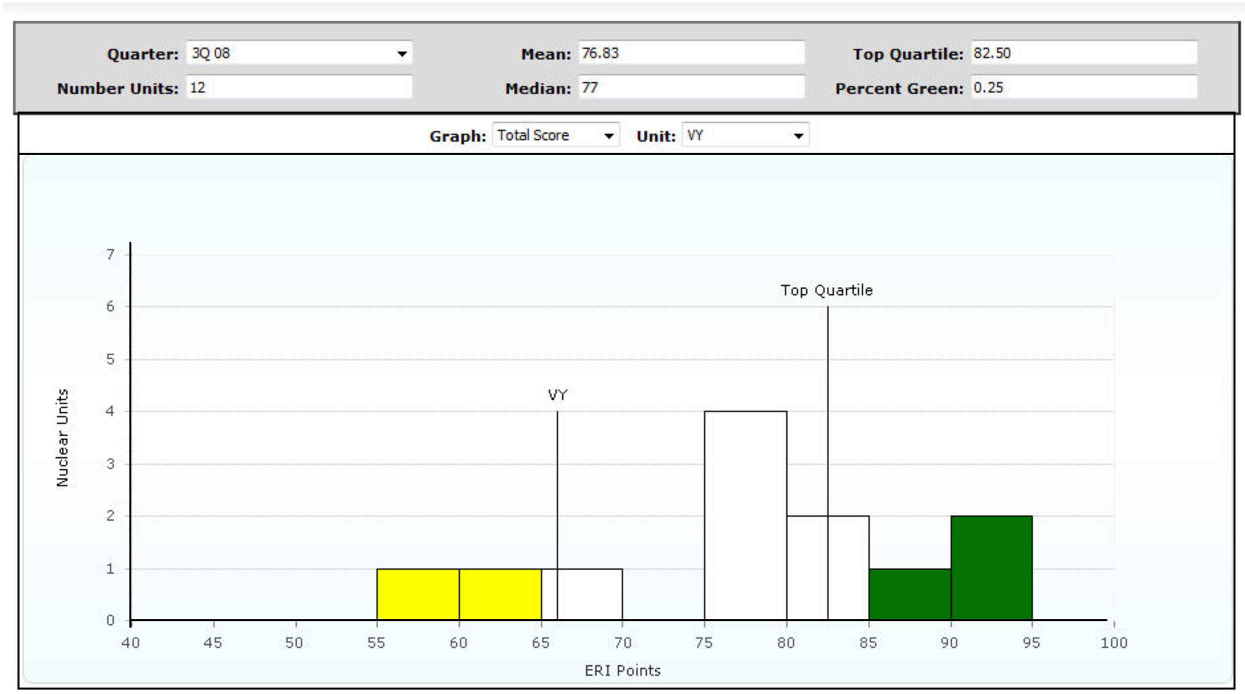

Figure 20 - ENVY vs. Sister Plants Equipment Reliability Performance

It should be noted that the following ER performance comparisons of ENVY to its 'Sister Plants' is for the 3rd Quarter 2008. The graph shown in Figure 20 compares the overall ER Index for ENVY as compared to its "Sister Plants", previously described in this report. As indicated, the ENVY overall ERI score of 66 ranks 10th of the 12 units reported, including Vermont Yankee.

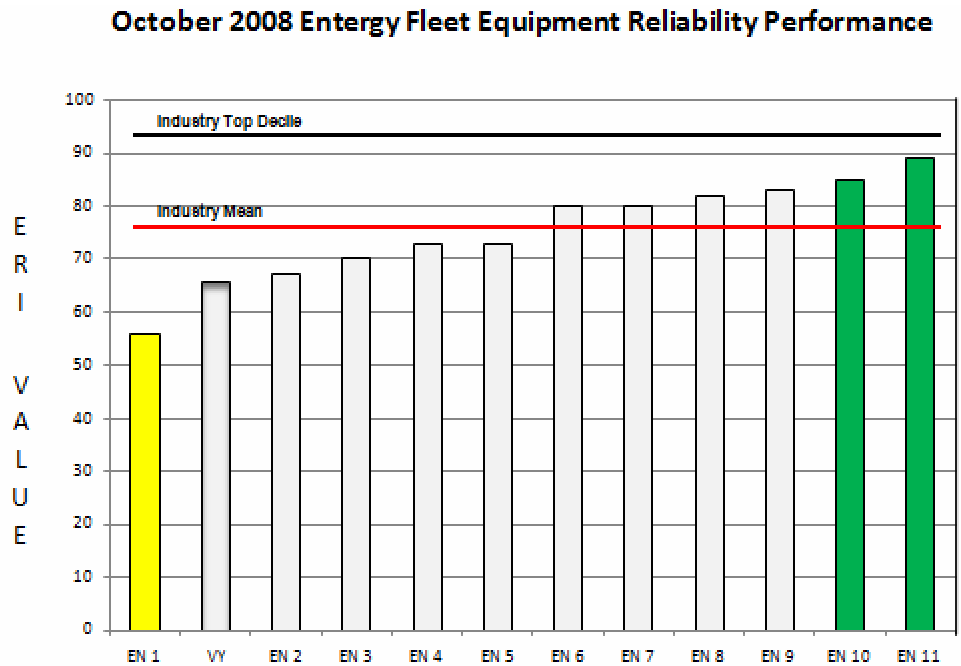

Figure 21 – ENVY vs. Entergy Fleet Equipment Reliability Performance

The Vermont Yankee facility was also compared to the other 11 units operated by Entergy Nuclear. The comparison shown in Figure 21 indicates that the Vermont Yankee facility ranks 11th out of the 12 nuclear units.

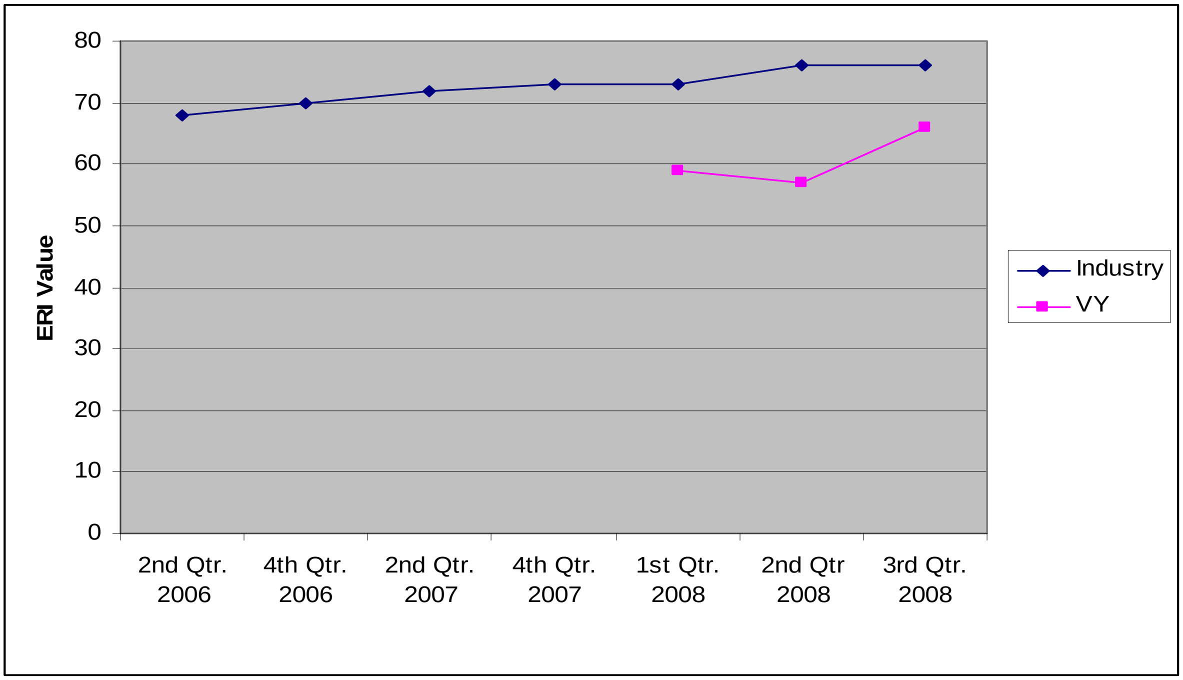

Figure 22 - Industry ERI Performance Trend

The plot in Figure 22 provides an industry overall Equipment Reliability Index (ERI) performance trend for the last seven (7) quarters. Prior to the first quarter 2008, Vermont Yankee data was not recorded as per the industry standard indicator.

Leading/Lagging ER Performance Indicators

The ER Index consists of 19 performance indicators. These indicators are divided into two groups, Leading and Lagging. Lagging indicators are those which measure past performance. Leading indicators provide insight to future performance. The following graphs compare the Vermont Yankee facility to the Industry.

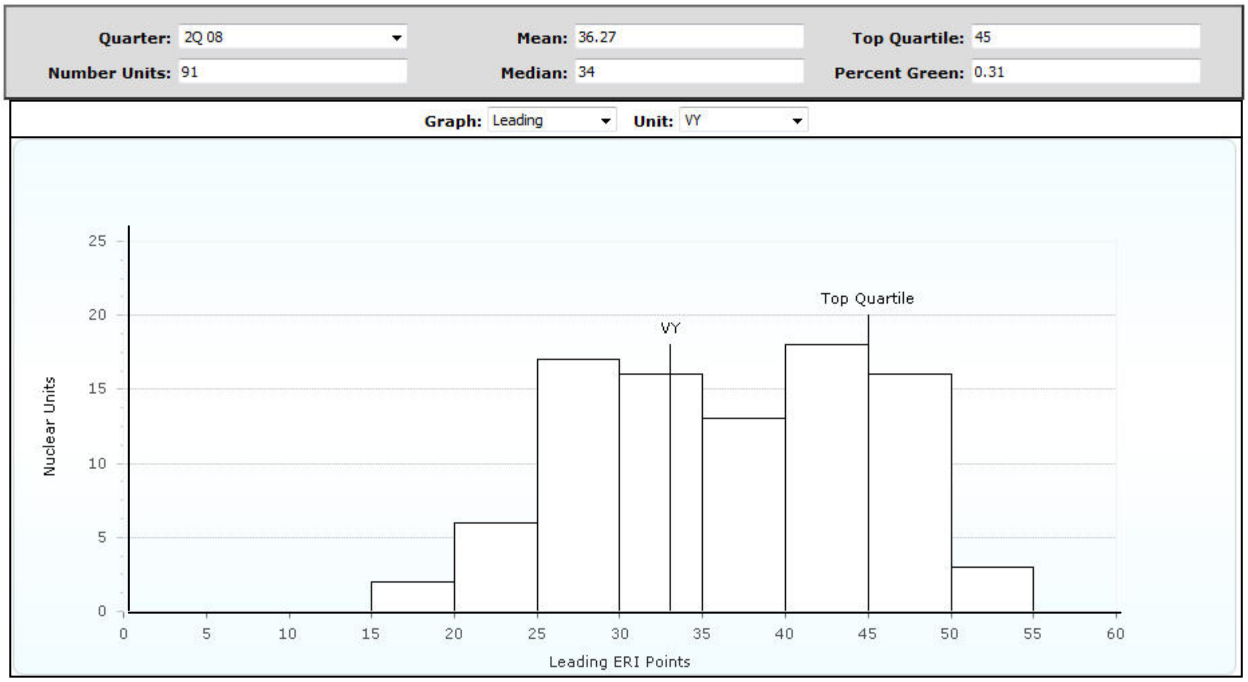

Figure 23 – ENVY vs. Industry Leading ERI Indicators

The graph in Figure 23 depicts the overall values for the Leading ERI performance indicators for the Industry, as well as the score for Vermont Yankee. The Industry average score for the Leading performance indicators is 36. Vermont Yankee ranks just below the Industry average at 33 points, ranking them in the 3rd quartile. The specific areas for improving the ERI leading indicators at ENVY are:

- Total Maintenance Backlog – 0 Points (Max value is 4)
- Age of Red and Yellow Systems – 0 Points (Max value is 6)
- Chemistry Index – 0 Points (Max value is 2)

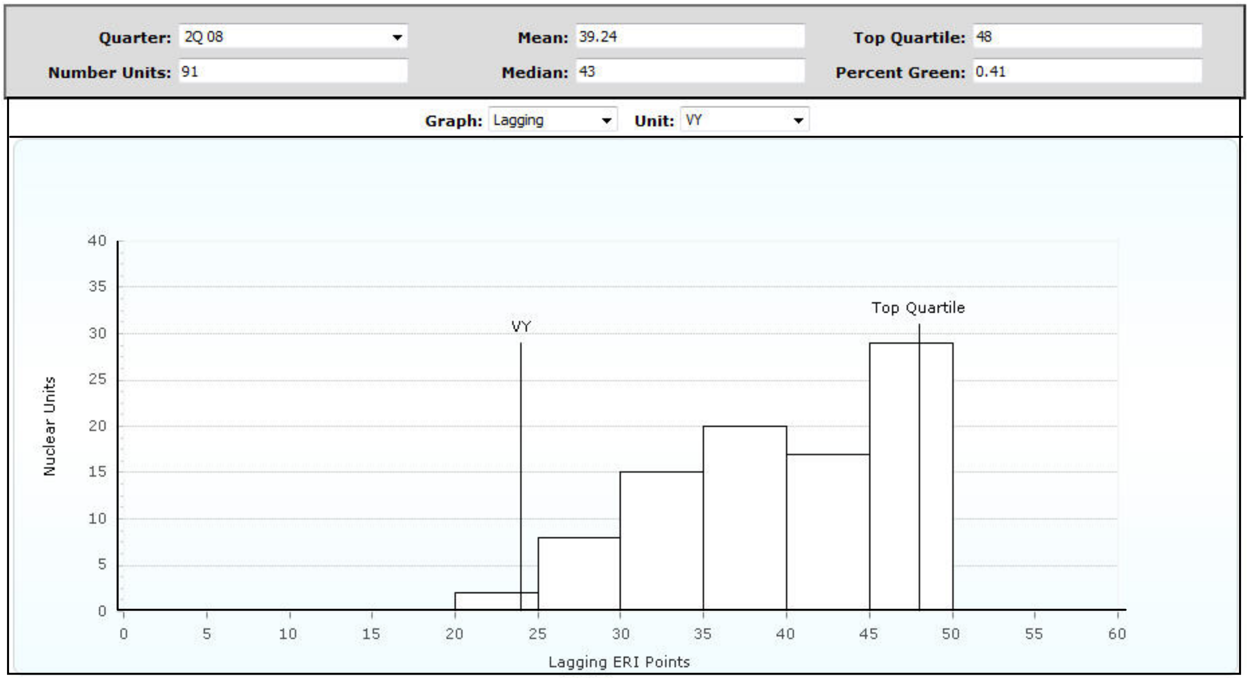

Figure 24 - ENVY vs. Industry Lagging ERI Indicators

The graph in Figure 24 depicts the overall values for the Lagging ERI performance indicators for the Industry, as well as the score for Vermont Yankee. The Industry average score for the Lagging performance indicators is 39. Vermont Yankee ranks well below the Industry average at 24 points, ranking them in the 4th Quartile. The specific areas for improving the ERI lagging indicators at ENVY are:

- Forced Loss Rate – 0 Points (Max value is 10)
- Unplanned Power Changes – 2 Points (Max value is 10)

Plant Benchmarking Report Summary

Unit Performance Summary: Generally, the benchmark data suggests that Vermont Yankee has been performing at a 'median' level. Although, there is concern that some recent ENVY indicators (Industry Performance Index and Forced Loss Rate) suggest unit performance might be falling.

Organization Summary: Generally, the benchmark data suggests that Vermont Yankee is reasonably staffed. One inconsistent area, however, seems to be in Engineering. It seems the engineering staff at the site is either high or at reasonable levels (Figure 15). However, System Managers might be understaffed based on the number of systems per system manager (Figure 17).

ERI Index Summary: At ENVY the current industry standard ER Index was reviewed & compared to the overall US Nuclear industry (91 units), the Entergy Fleet (12 units) and a group of "Sister Plants" of similar size or units that have implemented Extended Power Uprate projects. Overall the ENVY ER Index performance was as follows:

- ERI for the 2nd quarter of 2008 was 57 (of 100 possible points). The industry mean for this period is 75. ENVY is a 4th quartile performer as compared to the overall industry.
- As compared to the Entergy Fleet in the 3rd quarter of 2008, ENVY ranks 11th out of 12 units with a score of 66 points. Finally,
- As compared to the 11 "Sister Units," ENVY ranks 9th of 12 total units.

The major contributing factors for the recent degraded ER Index performance include:

- The two plant shutdowns caused by issues with the cooling tower and the Isophase bus which impacted the "Forced Loss Rate"
- Chemistry Index
- High Level of PM's deferred (38 in the 3rd quarter)
- High level of Elective Maintenance backlog.
- Average age of the plant systems designated as "Red" or "Yellow"

The ER Index performance does not meet industry standards. The ER Index at ENVY has increased from 52 to 66 over the past 3 quarters. This is an indication that management attention to ER process improvements is beginning to improve performance.

The data for ENVY, compared to the rest of the industry and the sister plant benchmarks, showed:

1. Considering the 40 to 50 contractors that were not reported in the ENVY data, ENVY meets industry staffing standards.
2. ENVY overall performance, as measured by the 2 year rolling average indicator has declined from 2006 to 2008. This decline is influenced by higher than industry forced loss rates, lower capacity factors and a greater number of recordable injuries.
3. The ENVY System Engineers have more systems per person than engineers at the benchmarked sister plants.

Section 1.4, Attachment 1

Sister Plants

Plant	Units
Brunswick	2
Cooper	1
Dresden	2
Duane Arnold	1
FitzPatrick	1
Hatch	2
Monticello	1
Nine Mile Point 1	1
Oyster Creek	1
Pilgrim	1
Quad Cities	2

Section 1.4, Attachment 2

Industry Group Subsets

Single Unit Peer Groups	Number of Units (Plants) In Peer Group
All Units (Plants)	**103 (65)**
All BWR Units	**34**
Single Unit Sites[1] (BWRs & PWRs)	**26**
Units < 850 MW[1,2] (BWRs & PWRs)	**12**
Units < 650 MW[1,2] (BWRs & PWRs)	**6**
Entergy Fleet[2]	**11**
Sister Units (Plants)[2]	**11 (8)**

[1] Group represents single unit sites only.

[2] Vermont Yankee was excluded from the median and quartile calculations.

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Section 1.4, Attachment 3

Unit Performance Data Definitions

1. **Plant Overall Performance Index:** weighting of a variety of performance parameters that measure various aspects of safety and performance.
2. **Capacity Factor:** the ratio of the net electricity generated to the energy that could have been generated at continuous full-power operation
3. **Forced Loss Rate:** the percentage of energy generation during non-outage periods that a plant is not capable of supplying to the electrical grid because of unplanned energy losses, such as unplanned shutdowns or load reductions.
4. **Production Cost:** fuel costs plus direct operating and maintenance costs divided by net generation ($/MWh).
5. **High Pressure Safety System Performance:** the ratio of hours available to hours required to be available
6. **Low Pressure Safety System Performance:** the ratio of hours available to hours required to be available
7. **Emergency Power System Performance:** the ratio of hours available to hours required to be available
8. **Recordable Injuries:** the number of Occupational Safety & Health Administration (OSHA) recordable injuries
9. **Fuel Reliability:** monitors performance in preventing defects in the metal cladding that surrounds fuel measured in microcuries per second
10. **Radiation Exposure:** the amount of radiation exposure in man-rem

Section 1.4, Attachment 4

Organization Staffing Data Definitions

1. **Plant Staffing:** excluding security staffing: the sum staffing including long-term contractors in all work functions less security
2. **Maintenance & Construction Staffing:** the sum of Maintenance/Construction and Maintenance/Construction Support staffing
3. **Operations Staffing:** the sum of Operations and Operations Support staffing
4. **Radiological Protection Staffing:** the sum of ALARA, HP Applied, HP Support and Radwaste/Decontamination staffing
5. **System/Plant Engineering:** the sum of engineering staff often referred to as Systems, Component and/or Maintenance engineers
6. **Total Engineering Staffing:** the sum of Computer Engineering, Design/Drafting, Modification Engineering, Technical Engineering, Reactor Engineering, Plant Engineering and Project Management staffing
7. **Systems/System Manager:** the number of plant systems divided by the number of system manager
8. **Management Staffing:** the sum of managers defined as those who supervise at least one first line supervisor or above
9. **Safety Staffing:** the sum of QA, QC, Nuclear Safety Review, Licensing and Emergency Preparedness staffing

Section 1.4, Attachment 5

Equipment Reliability Index

ERI Sub-Indicators	Indicator Type
Forced Loss Rate (Industry definition - 18 mo running average)	Lagging
Unplanned Power Reductions per 7000 hrs Critical (NRC Indicator)	Lagging
Post Refuel Outage Performance (60 Days)	Lagging
Unplanned LCO Entries (Scram De-rate & < 72 hours in last 3 months)	Lagging
Operator Work Around	Lagging
Critical Component Failures (No. of Critical Failures in last 3 months)	Lagging
Safety System Unavailability (NRC Indicators - MSPI)	Lagging
System Health Improvement Effectiveness	Leading
Corrective Maintenance Backlog (Non-Outage)	Lagging
Total Maintenance Backlog	Leading
Deferral of PMs	Leading
Maintenance Feedback (Percentage of PM's with feedback)	Leading
Timely Completion of PM's (1st Half of Grace)	Leading
Work Week Schedule Stability (Average of last 3 months)	Leading
Work Week Schedule Adherence (Average of last 3 months)	Leading
Long Term Planning (Qualitative status of overall process at station)	Leading
Age of Red and Yellow Systems	Leading
Chemistry Effectiveness Index (Index at or above industry average)	Leading
PM Program Bases (Status of Overall Station Process)	Leading

Appendix C: Reliability Significance Definitions

Reliability Significance Definitions

Throughout the report, the NSA team commented on ENVY levels of performance regarding numerous managerial and technical areas. Definitions of terms used by NSA team members are as follows

Good Practice: Managerial or technical area that was determined to be consistent with industry good practices that support plant and equipment reliability.

Meets Industry Standards: Managerial or technical area that was determined to be consistent with industry standard practices that support plant and equipment reliability.

Watch Area: Pertains to issues identified for which management is aware, however without appropriate focus, there exists a potential for future effect on plant and/or equipment reliability.

Challenge: Increased management focus or additional corrective actions are recommended. If this does not occur, future plant and/or equipment reliability could be affected.

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Appendix D: Regulatory Assessment Matrix

	Section 3(a)(1): Electrical System: Back-up or stand-by electrical system				Section 3(a)(2): An Emergency System: the Emergency Core Cooling System		Section 3(a)(3): A Mechanical System: The condensate/ feed water system	
	Diesel Generators	Batteries	Vernon dam tie	All associated electrical connections and controls	High pressure coolant injection system (HPCI)	Low Pressure Injection System	Condensate/ Feed Water System	Condenser
Vertical Audit								
4(1) Initial conditions	▪ Was loading an issue in 1992? ▪ Why was loading missed in 1992 EDSFI? ▪ Look at corrective actions identified in 1992 EDSFI and subsequent inspections. How were the corrective actions implemented? ▪ Look at findings in CDBIs and corrective actions; relate back to EDSFI. ▪ Look at license event reports (LERs) for these areas and how they were resolved.		N/A	▪ Look at corrective actions identified in 1992 EDSFI. ▪ How were those corrective actions implemented?	yes	Done in 3(a)(4) - RHR	yes	yes
4(2) Procurement			N/A		yes	Done in 3(a)(4) - RHR	yes	yes
4(3) Installation			N/A		yes	Done in 3(a)(4) - RHR	yes	yes
4(4) Operation			▪ Look at corrective actions identified in NRC IR-2004008 and in other inspections. ▪ How were those corrective actions implemented?		yes	Done in 3(a)(4) - RHR	yes	yes
4(5) Testing					yes	Done in 3(a)(4) - RHR	yes	yes
4(6) Inspection					yes	Done in 3(a)(4) - RHR	yes	yes
4(7) Maintenance					yes	Done in 3(a)(4) - RHR	yes	yes
4(8) Repairs					yes	Done in 3(a)(4) - RHR	yes	yes
4(9) Modifications					yes	Done in 3(a)(4) - RHR	yes	yes
4(10) Redesign					yes	Done in 3(a)(4) - RHR	yes	yes
4(11) Seismic Analysis (See Note 1)					yes	Done in 3(a)(4) - RHR	yes	yes
4(12) Training					yes	Done in 3(a)(4) - RHR	yes	yes
4(13) CAPs					yes	Done in 3(a)(4) - RHR	yes	yes
5(a)(2) Horizontal	If required	If required	If required	If required	If required	If required	If required	If required
5(a) Notes on methodology	See above	See above	See above	See above	Baseline: ▪ HPCI Report (2000) ▪ CDBI (2006)		Baseline: ▪ Report (2004) ▪ Checkworks Testimony	Baseline: ▪ Report (2004) ▪ Checkworks Testimony
5(b) Add'l Inquiries								
Notes	Note 1: Seismic analysis applies to those systems for which a full review is being performed. Review existing bases for seismic analysis to assure that loads are within the initial design envelope. Modifications should also be verified for seismic adequacy.							

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Section 3(a)(4): The primary containment system						Section 3(a)(5): A heat removal system			Section 3(a)(6): Cooling system dependent upon Connecticut River water		Section 3(a)(7): An underground piping system that carries radionuclides
Dry well shell	Torus supports	Residual Heat Removal System (RHR)	Isolation valves	Containment Spray	Adequate Suction	Normal cooling towers	Emergency-related cooling towers	Alternate cooling system	Alternate cooling system	Emergency service water	
▪ Coating ▪ Thickness ▪ Weld inspections ▪ Type A, B, C tests (review adequacy of these tests including trends determined in isolation valve testing)	▪ Fitzpatrick	yes	▪ Type C Tests (review adequacy of programs)	Done in 3(a)(4) - RHR		yes	yes	done in 3(b) (SWS)	done in 3(b) (SWS)	yes	See Note 3
		yes		Done in 3(a)(4) - RHR		yes	yes	done in 3(b) (SWS)	done in 3(b) (SWS)	yes	
		yes		Done in 3(a)(4) - RHR		yes	yes	done in 3(b) (SWS)	done in 3(b) (SWS)	yes	
		yes		Done in 3(a)(4) - RHR		yes	yes	done in 3(b) (SWS)	done in 3(b) (SWS)	yes	
		yes		Done in 3(a)(4) - RHR		yes	yes	done in 3(b) (SWS)	done in 3(b) (SWS)	yes	
		yes		Done in 3(a)(4) - RHR		yes	yes	done in 3(b) (SWS)	done in 3(b) (SWS)	yes	
		yes		Done in 3(a)(4) - RHR		yes	yes	done in 3(b) (SWS)	done in 3(b) (SWS)	yes	
		yes		Done in 3(a)(4) - RHR		yes	yes	done in 3(b) (SWS)	done in 3(b) (SWS)	yes	
		yes		Done in 3(a)(4) - RHR		See note 2	yes	done in 3(b) (SWS)	done in 3(b) (SWS)	yes	
		yes		Done in 3(a)(4) - RHR		yes	yes	done in 3(b) (SWS)	done in 3(b) (SWS)	yes	
		yes		Done in 3(a)(4) - RHR		yes	yes	done in 3(b) (SWS)	done in 3(b) (SWS)	yes	
		yes		Done in 3(a)(4) - RHR		yes	yes	done in 3(b) (SWS)	done in 3(b) (SWS)	yes	
		yes		Done in 3(a)(4) - RHR		yes	yes	done in 3(b) (SWS)	done in 3(b) (SWS)	yes	
		If required		If required		If required	If required	If required	If required	If required	
						Root cause report on cooling towers					
						Note 2: Tower modifications to be examined include (1) whether 1987 replacement of fill in seismic tower is sufficient and (2) whether new 2006 fiberglass beams in West Tower may plug suction.					Note 3: The panel is informed that there are no underground piping systems carrying radioactivity at Vermont Yankee

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Section 3(b): Additional Systems				Section 3(c): Generic Systems Issue: Cable separation - Separation of safety systems		Section 2(2): Deviations, exemptions or waivers from new reactor regulatory requirements
Main Transformer	Service Water System (SWS)	Management and Corporate Review	Sister Plant Review	Physical Separation	Electrical Separation	
	See Note 4	NSA Scope of Work, dated 10/14/08	NSA Scope of Work, dated 10/14/08	NSA Scope of Work, dated 10/22/08	NSA Scope of Work, dated 10/22/08	Two systems shall be chosen for which vertical slices inspections are being performed. The current Standard Review Plans applicable to those systems shall be reviewed and compared with the Vermont Yankee Design Basis Documents for those systems. Differences from current requirements shall be identified, and an assessment of these differences on reliability shall be rendered.
yes	yes					
yes	yes					
yes	yes					
yes	yes					
yes	yes					
yes	yes					
yes	yes					
yes	yes					
yes	yes					
yes	yes					
yes	yes					
yes	yes					
yes	yes					
If required	If required					
Root cause report on transformer fire	Buried Pipe and Tank Inspection Program (BPTIP)					
Note 4: Since the panel is informed there are no undergroundpiping systems carrying radioactivity, the Panel designates the Service Water System, which has buried piping, to be evaluated. The Buried Pipe and Tank Inspection Program (BPTIP) will be evaluated as part of the review of SWS.						

Appendix E: List of Acronyms

ACRONYMS

ACRONYMS	DESCRIPTION
AAC	Alternate AC
ACE	Apparent Cause Evaluation
AFI'S	Areas for Improvement
AO	Auxiliary Operator
AOG	Augmented Off Gas
AOV	Air Operated Valve
ASLB	Atomic Safety & Licensing Board
ASTM	American Society for Testing & Materials
BOP	Balance of Plant
BTP	Branch Technical Position
BWR	Boiling Water Reactor
CARB	Corrective Action Review Board
CASS	Caustic Austenitic Stainless Steel
CDBI	Component Design Basis Inspection
CM	Configuration Management
CR	Condition Report
CRD	Control Rod Drive
CRG	Condition Review Group
CS	Core Spray
CT	Cooling Tower
DPICS	Department Process Improvement Coordinator
DR	Discrepancy Report
EAL	Emergency Action Levels
EBOP	Emergency Bearing Oil Pump
ECCS	Emergency Core Cooling System
ECR	Emergency Change Request
EDG	Emergency Diesel Generator
EDSFI	Electrical Distribution System Function Inspection
EMPAC	Electronic Maintenance Planning & Control
ENOS	Entergy Nuclear Operating
ENVY	Entergy Nuclear Vermont Yankee
EP	Emergency Preparedness
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
ERI	ER Index
ERO	Emergency Response Organization
ERWG	Equipment Reliability Working Group
ESP	Engineering & Support Personnel
ESS	Engineered Safeguards System
FAC	Flow Accelerated Corrosion
FIN	Fix It Now
FME	Foreign Material Exclusion

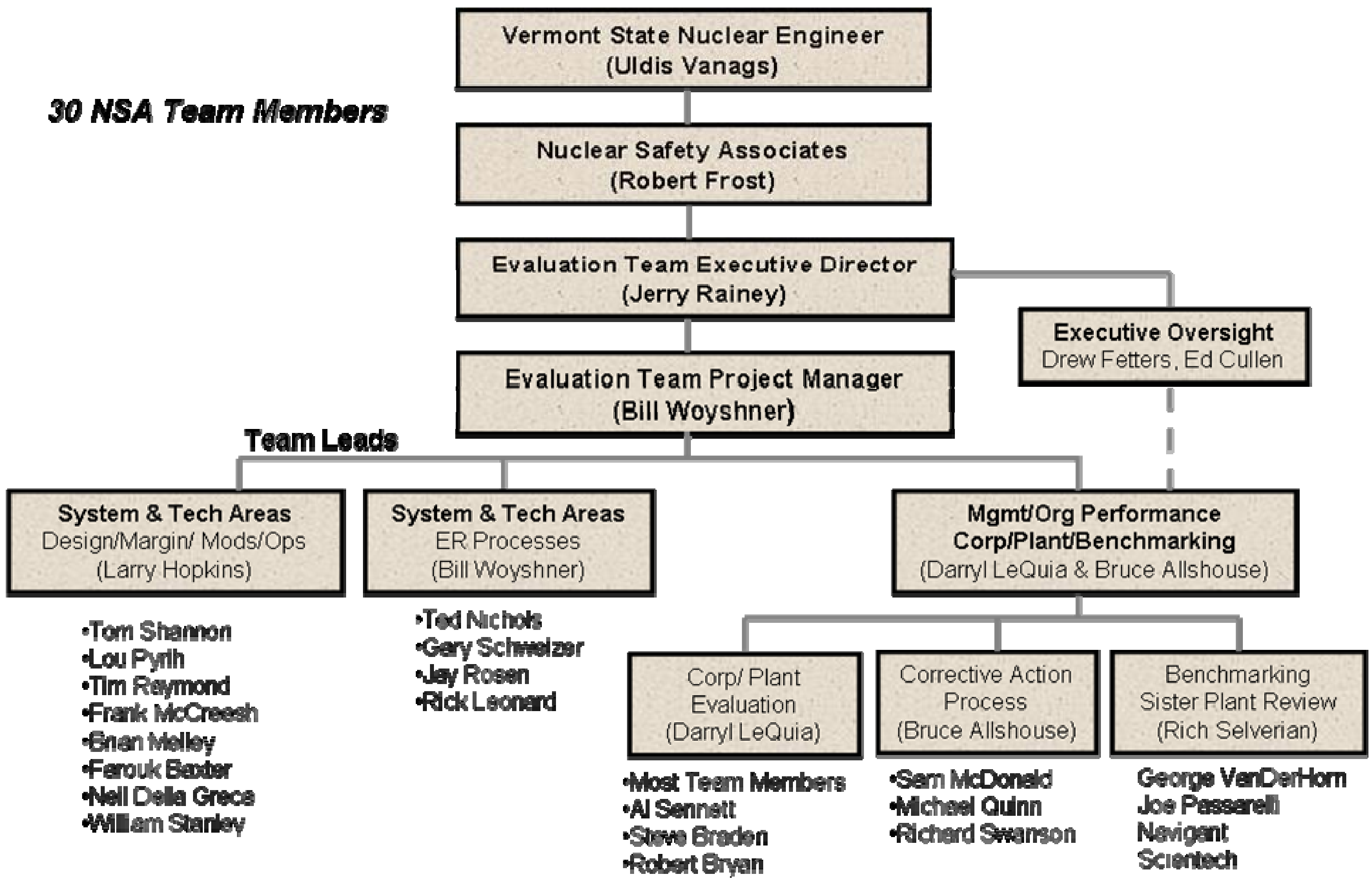
ACRONYMS	DESCRIPTION
FRP	Fiberglass Reinforced Plastic
FSS	Field Support Supervisor
FWH	Feedwater Heater
GOES	Governance Oversight Execution and Support
HP	Human Performance
HPCI	High Pressure Coolant Injection System
HPER	Human Performance Error Review
I&C	Instrumentation & Controls
IAS	Indus Asset Suite
IEEE	Institute of Electrical & Electronic Engineering
ILRT	Integrated Leak Rate Test
INPO	Institute of Nuclear Power Operations
IR	Infrared
IR	Integrated Report
ISFSI	Interim Spent Fuel Storage Initiative
ISO	Isophase
iTi	Integrated Technologies Inc.
LCO	License Condition for Operation
LERs	Licensee Event Reports
LLRT	Local Leak Rate Test
LOCA	Loss of Cooling Accident
LOOP	Loss of Offsite Power
LPCI	Low Pressure Cooling Injection
LPRM	Low Power Range Monitor
LTAM	Long Term Asset Management
LTCA	Long Term Corrective Actions
MCC	Motor Control Center
MOV	Motor Operated Valve
MRFFS	Maintenance Rule Functional Failures
MRM	Management Review Meeting
MRP	Motor Replacement Program
MSIV	Main Steam Isolation Valve
Mwt	Megawatts Thermal
NRR	Nuclear Regulation Review
NSA	Nuclear Safety Associates
NSSS	Nuclear Steam Supply System

ACRONYMS	**DESCRIPTION**
NUMARC	Nuclear Management & Resource Council
O&P	Organizational & Programmatic
ODMI	Organizational Decision Making Instruction
OE	Operating Experience
OJT	On the Job Training
P&IDs	Piping & Instrument Diagrams
PCIS	Primary Containment Isolation System
PCR	Procedure Change Request
PCRS	Paperless Condition Reporting System
PdM	Predictive Maintenance Program
PI	Performance Indicators
PM	Preventive Maintenance
PPRs	Personnel Performance Reviews
PR	Preliminary Report
RCA	Root Cause Analysis
RCAR	Root Cause Analysis Report
RCIC	Reactor Cooling Injection System
RFO	Refuel Outage
RFP	Reactor Feed Pump
RG	Regulatory Guide
RHR	Residual Heat Removal
RPS	Reactor Protection System
RUPS	Rotating Uninterruptible Power Source
RX	Reactor
SARB	Safety Analysis Review Board
SBO	Station Blackout
SER	Safety Evaluation Report
SPV	Single Point of Vulnerability
SRO	Senior Reactor Operator
SRP	Standard Review Plan
SSE	Supplemental Safety Evaluation
STA	Shift Technical Advisor
SW	Service Water
TEAR	Training Evaluation and Action Request
TOC	Training Oversight Committee
TPE	Training Performance Evaluations
TRG	Training Review Group
TS	Technical Specification
TSV	Turbine Stop Valve
UFSAR	Update Final Safety Analysis Report

ACRONYMS	**DESCRIPTION**
URT	Unit Reliability Team
VAC	Volts AC
Vdc	Volts DC
WCG	Work Place Cornerstone Group
WO	Work Order
WSCI	WSC International
WWM	Work Week Manager

Appendix F: Assessment Team Organization Chart

Vermont Yankee Nuclear Plant
Reliability Assessment Team



Appendix G: Root Cause Assessment Report

The complete Root Cause Assessment Report will be provided by December 31, 2008.

Appendix H: Assessment Team Resumes

Samuel G. McDonald

Senior Consulting Engineer

AREAS OF EXPERTISE:

SUMMARY:

• Environmental, Health & Safety Management (NRC/EPA/OSHA

Regulations regarding Criticality Safety, Radiation Protection, Safeguards,

Chemical Safety, Fire Protection, Industrial Safety, Environmental

Protection, and Emergency Response/Incident Command)

Management Leadership/Strategic Planning

• Manufacturing Operations/Strategies (Total Quality/Time-Based Management/Lean)

Nuclear Safety Culture Evaluations

Light Water Reactor Fuel Fabrication

UF6 Conversion/UO2 Pelleting

Light Water Reactor Fuel Performance

Zirconium Alloy Metallurgy

Radiation Effects in Nuclear Reactor Structural Materials

Transmission/Scanning Electron Microscopy

Dr. McDonald has over 36 years experience in the nuclear industry, 34 of which were spent as an engineer and manager employed by the Westinghouse Electric Corporation's Nuclear Fuel Division and the Westinghouse Electric Company LLC Nuclear Fuel Business Unit. He retired from the Westinghouse Electric Company in July 2006. Dr. McDonald has a broad technical experience base that includes expertise in the following areas: metallurgy of zirconium and its alloys (he was the co-developer of ZIRLO, the fuel cladding used in many PWR's world-wide); radiation effects in nuclear reactor structural materials; light water reactor fuel manufacturing, including UF6 Conversion, UO2 pelleting, and zirconium alloy cladding fabrication; and light water reactor fuel performance. During the course of his technical career, Dr. McDonald received a number of engineering achievement awards, including a George Westinghouse Signature Award, the highest award granted by the Westinghouse Electric Corporation. Dr. McDonald has also managed a number of different technical organizations. He was the Technical Services Manager for the Westinghouse Columbia Fuel Fabrication Facility for over a decade; and was the Environmental, Health & Safety Manager and Lead Emergency Director for the same facility for over five years. He also managed a national recruiting/development program for entry-level engineers for the Westinghouse Electric Corporation. The objective of the latter program was to identify, recruit, train and place high-potential engineers in leadership positions throughout the corporation. During his management career, Dr. McDonald received a number of awards, one of which was an Energy Systems Business Unit Total Quality Award, the highest management award granted by the Westinghouse Electric Corporation. He was also an Auditor for the Corporation's Total Quality Achievement Award Program, a member of the Department of Commerce's Materials Technical Advisory Committee, and has participated in numerous INPO- based Peer Reviews and INPO/NRC-based Safety Culture evaluations. Dr. McDonald is a highly experienced manager who has implemented Operations Strategies such as Total Quality and Time-Based Management (Lean); successfully applied a number of continuous improvement techniques including Strategic Planning methodologies such as the Balanced Scorecard, and taught business leadership principles to several generations of engineers and managers.

EDUCATION:

- Ph.D., Metallurgical Engineering & Material Science, University of Notre Dame du Lac, 1973
- M.S., Metallurgical Engineering & Material Science, University of Notre Dame du Lac, 1970
- B.S., Metallurgical Engineering & Material Science, University of Notre Dame du Lac, 1968

CLEARANCES:

DOE L, DOE Q, NRC U

EMPLOYMENT Nuclear Safety Associates

HISTORY: *Senior Consulting Engineer* *August 2006 -Present*

Westinghouse Electric Company, LLC

Apr 2001 to July 2006 Environment, Health & Safety Manager, Columbia Site

Westinghouse Electric Company. LLC

Apr 1991 to Apr 2001 Technical Services Manager, Columbia Fuel Fabrication Facility

Westinghouse Electric Corporation

May 1989 to Apr 1991 Manager, Engineering/Manufacturing Professional Development Program

Westinghouse Electric Corporation

Jan 1987 to May 1989 Manager, Chemical Process Engineering, Columbia Plant

Westinghouse Electric Corporation
Manager, Chemical Process Development, Columbia Plant

Westinghouse Electric Corporation
Fellow Engineer, Process & Applications Engineering, Nuclear Fuel Division, Fuel Engineering Department

Westinghouse Electric Corporation
Fellow Engineer, Performance Analysis, Nuclear Fuel Division, Fuel Engineering Department

Westinghouse Electric Corporation
Fellow Engineer, Westinghouse Electric Corporation Research & Development Center

Westinghouse Electric Corporation
Principal Engineer, Westinghouse Electric Corporation Research & Development Center

Westinghouse Electric Corporation
Principal Engineer, Westinghouse Electric Corporation, Advanced Reactor Division

University of Notre Dame du Lac
Argonne National Laboratory/Argonne University Association Fellow, Argonne National Laboratory
Jan 1986 to Jan 1987

Feb 1984 to Jan 1986

Feb 1982 to Feb 1984

May 1980 to Feb 1982

Aug 1972 to May 1980

Mar 1972 to Aug 1972

May 1970 to Mar 1972

EXPERIENCE:

Nuclear Safety Associates 8/2006 to Present

Management consultant focused in Nuclear Safety related activities, including Safety Culture evaluations, Human Performance, and Emergency Response.

Westinghouse Electric Company, LLC Nuclear Fuel Business Unit, Columbia Site 4/2001 to 7/2006

Environmental, Health & Safety Manager for the Columbia Site. Responsible for assuring that all applicable regulations, licenses and permits, were understood and effectively translated into formal policies and procedures, and appropriately implemented on the shop floor. Principle regulatory bodies included NRC, EPA, OSHA and SCDHEC. Site Emergency Director, responsible for Emergency Response Organization training and event response. Developed and maintained site EH&S Strategic Plan.

Westinghouse Electric Company, LLC Nuclear Fuel Business Unit, Columbia Fuel Fabrication Facility 4/1991 to 4/2001

Technical Services Manager for the Columbia Fuel Fabrication Facility. Responsible for all Manufacturing Engineering (Chemical & Mechanical) including Process Engineering, Process Development, Facilities Engineering, and development/implementation of all capital and expense plans. Developed and maintained Columbia Plant Strategic Plan. Implemented Time-Based Management (Lean) Operations Strategy in conjunction with Plant Staff.

Westinghouse Electric Corporation Human Resources 5/1989 to 4/1991

Engineering/Manufacturing Professional Development Program (EMPDP) Manager. Responsible for recruiting, development (leadership skills), and placement of high potential entry-level engineers throughout the Westinghouse Electric Corporation.

Westinghouse Electric Corporation Nuclear Fuel Division, Columbia Fuel Fabrication Facility 1/1987 to 5/1989

Chemical Process Engineering Manager for the Columbia Fuel Fabrication Facility. Responsible for all process engineering functions at the Columbia Plant including UF6 conversion (wet and dry processes), UO2 pelleting, uranium recycle, Low Level Radiation Waste (LLRW), and all associated support processes. Also responsible for implementing Westinghouse Total Quality Strategy in the Chemical Area.

Westinghouse Electric Corporation Nuclear Fuel Division, Design Engineering, Product/Process Development & Design, Columbia Plant 1/1986 to 1/1987

Chemical Process Development Manager for the Columbia Fuel Fabrication Facility. Responsible for all technical process development and process improvement activities associated with UF6 conversion, UO2 pelleting, uranium recycle, LLRW, and all associated processes.

Westinghouse Electric Corporation
Nuclear Fuel Division, Design Engineering, Process & Applications
Engineering, 2/1984 to 1/1986

Fellow Engineer responsible for interfacing Design Engineering and Manufacturing Engineering. Key activities included technical leadership of Fuel Division quality and yield improvement efforts in the conversion, pelleting and fuel rod fabrication areas.

Westinghouse Electric Corporation Nuclear Fuel Division, Design Engineering, Performance Analysis 2/1982 to 2/1984

Fellow Engineer responsible for developing improved understanding of PWR and BWR in-reactor fuel performance as well as appropriate fuel performance models. Specific activities included fuel rod/fuel assembly failure analyses, corrosion/crud behavior, mechanical properties, creep behavior, irradiation effects, and pellet/clad interaction (PCI). Extensive interaction with customers required.

Westinghouse Electric Corporation
Corporate Research & Development
Center 8/1972 to 2/1982

Senior/Fellow Engineer responsible for the development of advanced zirconium alloy fuel cladding (ZIRLO), and the development of an improved understanding and modeling of Zircaloy-4 corrosion and mechanical properties – both out- and in-reactor. Specific activities included (1) thermo-mechanical processing of development alloys and the Zircaloys to optimize corrosion resistance in PWR environments, (2) fabrication evaluations to optimize fuel performance and manufacturability, and (3) comprehensive mechanical property testing (tensile, creep, fatigue, etc.).

Westinghouse Electric
Corporation Advanced Reactor
Division ARD) 3/1972 to 8/1972

Senior Engineer responsible for understanding void volume induced swelling in Liquid Metal Fast Breeder Reactor (LMFBR) structural materials. Served as interface person between ARD and Argonne National Laboratory on Atomic Energy Commission sponsored advanced alloy development program.

Argonne National
Laboratory Material Science
Division 6/1970 to 3/1972

Argonne Universities Association/Argonne National Laboratory Fellow responsible for studying radiation effects in LMFBR structural materials. PhD dissertation focused on the swelling behavior of 304 stainless steel.

WSC International

GERALD R. RAINEY
President

SUMMARY

Mr. Rainey is currently President of WSC International, LLC. Mr. Jerry Rainey, a proven executive professional in the utility industry has led numerous organizations in their “Pursuit of Excellence” in the fossil and nuclear power generation sectors.

DETAILS

Mr. Rainey was President and Chief Nuclear Officer for PECO Energy Co. He was elected to this position in June 1, 1998. He was responsible for the direction of all PECO Nuclear functions including Limerick Generating Station, Peach Bottom Atomic Power Station, Station Support, Nuclear Quality Assurance, and Nuclear Planning and Development. He was also responsible for providing management services and performance improvement contracts at Vermont Yankee, Millstone, Clinton, Nine Mile and Hope Creek Nuclear Stations.

Mr. Rainey was COO responsible for the Sithe Energy Project. Exelon initially purchased 49.9% of Sithe Energies, North American Business, in December 2000. The Sithe purchase was completed in December 2002 and involved 3,800 megawatts (mw) of merchant generation, 2,500 mw under construction, and another 3,700 mw of generation in various stages of advanced development, as well as Sithe’s domestic marketing and development businesses.

During this time Mr. Rainey was also President and Chief Executive Officer of AmerGen Energy, a joint venture between PECO Energy and British Energy Company. He was responsible for the direction of nuclear functions at Three Mile Island, Clinton Power, and Oyster Creek Nuclear Stations.

Mr. Rainey joined PECO Energy Company in June 1969 after earning an Associate’s Degree in Industrial Engineering from Spring Garden Institute in Philadelphia. He earned a Bachelor of Science Degree in Engineering from Widener University in 1978, attended Fuqua Business School at Duke University in 1989 and Harvard University Graduate School of Business Administration in 1996. Rainey advanced through technical, engineering, construction, and start-up positions at Eddystone and

Cromby Generating Stations before moving into nuclear operations in the Maintenance Department at Peach Bottom in 1980. He served as an I&C Engineer, and a Branch Engineer at Limerick Generating Station during the construction and start-up of Unit 1 and Unit 2. His duties included (SRO) cold license program for Unit 1 and Electrical / I&C Start-Up, Director on Unit 2 until re-assignment to Peach Bottom in April 1987, following the Peach Bottom shut down. Upon his return to Peach Bottom in 1987 he was Superintendent of Plant Services, and was later named Superintendent of Maintenance. He was named Manager of Eddystone Generating Station in 1989, and then Vice President of Nuclear Services in 1992. In 1993, he was appointed Vice President, Peach Bottom Atomic Power Station. In March 1996, he was elected to Senior Vice President, Nuclear Operations.

WSCI *nternational*

WILLIAM S. WOYSHNER

EXECUTIVE VICE PRESIDENT

RESUME

SUMMARY

Mr. Woyshner is a Mechanical Engineer with a Bachelor of Sciences degree from Drexel University (1982) and is currently Executive Vice President for WSC International's Management Consulting Services. Mr. Woyshner has specialized in the Direction and Management of organizations involved in maintenance technology and process development and implementation. He is experienced in all levels of an organization from entry level engineer to President and COO of Maintenance & Diagnostic llc, an engineering and maintenance services business startup.

Mr. Woyshner has more than 24 years of experience in Engineering, Maintenance, and Asset Management of nuclear and fossil power plants, transmission, substation and distribution, systems and equipment. He has supported electric utilities in the development of effective corrective, preventive, predictive and proactive maintenance strategies, policies and procedures, diagnostic tools and technologies, system and equipment operations, reliability centered maintenance, risk management, management and work culture, and leadership development. Mr. Woyshner's accomplishments include:

- Developed and implemented Exelon's industry leading Predictive Maintenance program for the organization's nuclear fleet of assets (all 17 units).
- Developed the key components of an industry recognized and highly effective Predictive Maintenance Program in collaboration with the Electric Power Research Institute.
- Director of EPRI Maintenance and Diagnostic Center, the industry's leading diagnostic and maintenance processes and technologies development center at PECO's Eddystone fossil fuel power plant.
- Developed an integrated framework for Equipment Reliability & Asset Management and a structured organizational readiness assessment to support the utility industry in the implementation of business centered asset and maintenance management processes and procedures.
- Developed and implemented a leadership development, process improvement and continuous improvement program for Tokyo Electric Power (TepCo) in support of the restart activities of its 17 nuclear units.

DETAILS

- **Director of Maintenance Services – EPRI Solutions - 3/02 to present**

Responsible for managing Maintenance Services for the Power Delivery market and Nuclear/Fossil Power Generation business areas. Responsibilities include managing maintenance optimization services in the areas of Preventive, Predictive and Proactive maintenance programs, maintenance Work Management and new maintenance technology applications.

- **Utility Market Director - CSI Services - 10/96 to 2/99**

Responsible for the Sales, Delivery, Management and profit /loss statement for approximately $6 million annually of Maintenance , Diagnostic and Engineering services provided to the utility industry. The services included Managing EPRI's Maintenance and Diagnostic Center, maintenance strategy optimization implementation, predictive maintenance program development, PDM program implementation, and PDM training of personnel

- **President & COO – Maintenance & Diagnostics, llc. - 1/95 to 9/96**

Responsible for the Sales, Delivery and profit /loss statement for approximately $8 million annually of Maintenance, Diagnostic and Engineering services provided to the utility and general industries. Specific duties included: Directing the EPRI Maintenance and Diagnostic Center, utility and industrial plant maintenance strategy optimization; predictive maintenance program development, implementation, and training of personnel; and, the day-to-day management of an organization consisting of 35 engineering and technical employees working in 8 different technical business units.

- **Engineering Manager-EPRI Monitoring and Diagnostic Center - 1991 to 12/1994**

Specific responsibilities included both engineering and project management for the EPRI M&D Center projects pertaining to utility predictive maintenance program development and evaluation, and demonstration and development of new applied technologies in the area of equipment condition monitoring. Performed strategic planning and implementation management for a research and development project budget of approximately $3 million per year.

- **Lead Project Engineer-EPRI/PECO Research Project - 1988 to 1991**

Lead project engineer for the research project at PECO's Eddystone Generating Station that was jointly funded by the Electric Power Research Institute and Philadelphia Electric Company. The project is to evaluate and demonstrate new M&D technologies as applied to generating stations. Specific responsibilities included cost justification, equipment condition monitoring technology evaluation, design package preparation, and system installation and start-up for these state-of-the-art technologies.

- **Canus, Inc. - 8/87 - 3/88**

Preventive Maintenance Engineer at PECO Energy's Peach Bottom Nuclear Power Station, Delta, PA. Duties included identifying mechanical, electrical and physical plant deficiencies and coordination of the groups' response for corrective efforts to meet NRC criteria for plant restart.

- **Bogan, Inc. - 7/85 - 8/87**

Preventive Maintenance Engineer at Public Service Electric & Gas' Hope Creek Nuclear Station, Salem, NJ. Duties included establishing and implementing a preventive maintenance program for all non Q-listed instrumentation and writing surveillance procedure for various instrument systems and components.

- **Cataract Engineering & Construction - 2/83 - 7/85**

Instrument Engineer at PECO Energy's Limerick Nuclear Generating Station Units 1 & 2, Sanatoga, PA Duties included establishing and implementing a preventive maintenance program for all plant instruments (Q-listed and non Q-listed). This entailed determination of instrumentation required and the frequency of maintenance and writing procedures for each individual make and model of instrumentation (electronic, pneumatic, and hydraulic) as well as various calibration procedures.

Johnson Controls, Inc. - 10/81 - 1/83

Project Engineer. Duties included calibration, start-up service, and maintenance of fossil, nuclear, and industrial instrumentation systems. Responsible for bid preparation, procurement, and technical proposal preparation and on-site project management.

EMPLOYMENT

2003 – Present	WSC International LLC	Executive Vice President
1999 -- 2003	WSC INC (EPRI)	Director of Maintenance Services
1996 – 1999	CSI/Emerson	Utility Market Director
1995 -- 1996	Maintenance & Diagnostics LLC	President & COO
1991 – 1994	WSC Inc. (EPRI - M&D Center)	Engineering Manager
1988 – 1991	WSC Inc. (EPRI - M&D Center)	Lead Project Engineer
1987 – 1988	Canus, Inc. (Peach Bottom Nuclear Plant	I&C Engineer
1981 – 1987	Various Engineering Firms	I&C Engineer

EDUCATION

Bachelor or Science - Mechanical Engineering - 1982 Drexel University, Philadelphia, PA

DREW B. FETTERS

FORMER NUCLEAR GENERATING PLANT VICE PRESIDENT

RESUME

SUMMARY

Thirty one years of diverse executive, engineering management, engineering, project management, construction and maintenance experience in connection with nuclear and fossil power stations. Demonstrated ability to deal with top management and regulatory agencies. Technical leadership for company's acquisition of nuclear plants including evaluation of current state, opportunities for improvement and pro forma future performance results to achieve earnings targets.

DETAILS

WSC International LLC. (March 2002 – Present)

Served as a consultant to WSC International. Performed assessments and made recommendations to executive management relating to the construction progress of combined-cycle gas projects in Boston area. Proposed legal strategy to prepare for future litigation as a result of contractor project delays and over runs. Currently an independent representative responsible for evaluation and recommendation to the Board subcommittee of a large generating company on the development and execution of a multi-billion dollar refurbishment project.

Vice President – Special Projects (Exelon) (March, 2001 – February 2002)

Oversight of the construction of three combined cycle units (2580 Mwe) associated with the acquisition of Sithe Energies. Prepared recommendations to Genco Executive Team for organization and integration of Sithe into Exelon.

Vice President - Nuclear Acquisitions (PECO Energy) (June, 1999 – March, 2001)

Responsible for transition of Vermont Yankee from VYNPC to AmerGen Vermont ownership, including license transfer and financial closure, as well as planning for post-closing achievement of operational excellence initiatives.

Vice President - Nuclear Development (PECO Energy) (June, 1998 – June, 1999)

Responsible for evaluating the feasibility of the Company's investment in nuclear opportunities in North America (specifically involved in Management Contracts, the AmerGen partnership and the TMI acquisition).

Vice President - Nuclear Planning & Development (PECO Energy) (May, 1997 - June, 1998)Responsible for leadership and oversight of central maintenance support (Reactor Services and Turbines), strategic planning, administration, business development and interface with the two nuclear facilities. Chairman BWROG Executive Oversight Committee.

Vice President - Station Support (PECO Energy) (September 1995 -May, 1997)

Responsible for all Nuclear support functions, including Central Maintenance, Engineering, Licensing, Fuel Management, and Emergency Planning, for Limerick and Peach Bottom power plants. BWROG Executive Oversight Committee Member.

Director - Nuclear Engineering Division (PECO Energy) (March 1994 - September 1995)

eport directly to the Vice President of Station Support Department. Responsible for nuclear plant design basis, engineering of major programs, projects and studies for PECO nuclear stations. EPRI Nuclear Power Council Member.

Director of Maintenance - Limerick Generating Station (PECO Energy) (April 1993 - March 1994)

Report directly to the Vice President of Limerick Generating Station. Responsible for the planning and execution of all Maintenance and Instrument and Control activities for Limerick, two 1100 Mwe nuclear generating units.

Project Director - Financial Information Systems Project (PECO Energy) (1992 - 1993 - 4 months)

Report directly to the Senior Vice President-Financial. Responsible for the development and execution of a project plan to replace the corporate general ledger, procurement and timekeeping systems at PECO.

NEEDS Task Force - Nuclear Group (PECO Energy) (1992 - 9 months)

Member of a task force reporting directly to the Senior Vice President - Nuclear. This group performed a comprehensive analysis of all Nuclear Group activities and made staffing and organizational recommendations to executive management.

Manager of Projects - Limerick Generating Station (PECO Energy) (1989 - 1993)

Report directly to the Vice President of Nuclear Engineering and Services Department. Responsible for establishing the priorities for all capital and expense modifications to the Limerick Station and through a staff of ten project managers assuring the timely planning, engineering and installation of all modification work.

Project Manager - Limerick Generating Station (PECO Energy) (1987 - 1989)

Reported directly to the Vice President of Nuclear Engineering Department. Responsible for coordination of all engineering activities with respect to Limerick Unit 2 construction, and responsible for coordination of engineering support for the operational Limerick Unit 1.

Project Manager - Limerick Generating Station Unit 2 (PECO Energy) (1984 - 1987)

Reported to the Division Manager of Mechanical Engineering Division. Responsible for the coordination of all engineering at Bechtel and Philadelphia Electric Company for the construction of Unit 2 and the budget and cost for the entire project.

Various Engineering Positions - (PECO Energy) (1973 - 1984)

- Directed group of approximately 30 matrixed engineers in the review and approval of preoperational tests and results for the Preoperational Test Program for Unit 1 at Limerick. Responsible for research and design of upgrades to containment structures and systems.
- Served as member of the Mark I Containment Owners Group and Member of Technical Review Committee for the 15 Mark I GE BWR's.
- Performed various structural design projects and was a construction field Engineer on Limerick Generating Station, a two-unit boiling water reactor, in the reactor building.

EDUCATION

Bachelor of Science, Civil Engineering Lehigh University - 1973

Masters of Science, Civil & Urban Engineering University of Pennsylvania - 1978

Professional Engineer State of Pennsylvania - 1979

Masters of Business Administration St. Mary's College of California - 1984

Project Management Professional Certification 1994

CLEARANCE

Nuclear Plant Access (1973 – 2002)

WSCI nternational

EDWARD J. CULLEN, JR.

Former Vice President and General Counsel

BIOGRAPHY

SUMMARY

Mr. Cullen received a B. A. degree from Iona College, New Rochelle, New York, in 1969 and received a Juris Doctor degree from Villanova University in 1972. In 1997, he received a certificate from the Columbia University Graduate Business School for completion of the Columbia Senior Executive Program. He is admitted to the bar of the Supreme Court of Pennsylvania and the U.S. Court of Appeals for the Third Circuit.

From 1972 through 1987, Mr. Cullen was assistant general counsel of Philadelphia Electric Company "PECO". In that capacity his primary responsibility was the commercial and regulatory legal work relating to the Company's nuclear, fossil and hydro generating facilities. Early in this period he was intimately involved in the federal and state permitting and licensing activities for the Peach Bottom, Limerick and Fulton nuclear facilities. These activities included AEC (now NRC) construction permit and operating license proceedings, US Environmental Protection Agency and Army Corps of Engineers permitting, Delaware and Susquehanna River Basin Commission proceedings and Pennsylvania Department of Environmental Resources permitting. He also was responsible for advising the Company on the various commercial transactions involved in plant construction. Also during this period he advised the Company on air and water permitting issues for its non- nuclear facilities and was the primary lawyer involved in approval of the Peach Bottom Unit 1 Decommissioning Plan.

In 1988, Mr. Cullen was appointed Associate General Counsel and Head of the Corporate / Operations group within PECO's legal department. In this position he was responsible for managing legal services to all PECO's operating groups and handling legal issues relating to corporate structure and development. He continued to handle substantive legal issues in the commercial, regulatory and nuclear licensing areas. For example, in 1993 he was the primary lawyer for PECO in the negotiation of the contract for the acquisition of the partially irradiated core of the Shoreham nuclear station and for the licensing and permitting of that fuel from Shoreham to PECO's Limerick Generating Station. Following this transaction, Mr. Cullen was appointed Deputy General Counsel of PECO Energy and continued to head the Corporate / Operations Group. In this position, he was the primary lawyer negotiating the formation of AmerGen Energy Company with British Energy to acquire ownership of existing U.S. nuclear plants and successfully closed the first such acquisition.

In 2000, following the close of the PECO – Unicom merger which formed Exelon Corporation, Mr. Cullen was appointed Vice President and General Counsel of Exelon Generation Company (“ExGen”), which was the owner and operator of all the nuclear and non-nuclear generating assets previously owned by PECO and Unicom, and was responsible for marketing at wholesale the output of these units. In this position, he was the chief legal officer of ExGen. During this time he was involved directly in the preparation of the NRC license renewal application for Peach Bottom Units 2&3, which was to become the template for license renewal applications for other EXGEN and AmerGen units.

In 2002, following an Exelon reorganization, Mr. Cullen was appointed Vice President and Deputy General Counsel of Exelon Corporation and head of the Corporate and Commercial group within the legal department. He was responsible for coordinating legal services provided from the Philadelphia and Chicago offices to all divisions and subsidiaries of Exelon on a wide range of legal issues. In this position he continued to be responsible for managing regulatory and commercial legal services to ExGen and acted as general counsel for ExGen.

Since his retirement from Exelon in late 2006, he has continued to keep current on issues affecting the electric utility industry through continuing contacts with people active in the industry. He has been active in providing Pro Bono services to various organizations and individuals.

LAWRENCE HOPKINS

NUCLEAR PLANT MANAGER AND DIRECTOR OF OPERATIONS

RESUME

SUMMARY

Over 30 Years of experience in the nuclear industry managing and leading Nuclear Plant Operations for PECO Energy and AmerGen. Experience in nuclear plant engineering, training and regulatory and licensing as well. Provided management consulting for performance improvement as a consultant for Tokyo Electric Power Company and Florida Power & Light. Managed Nuclear Plant Operations at Limerick Generating Station and Nine Mile Point Nuclear Plant and served as Plant Manager at Nine Mile Point Nuclear Generating Station. Major accomplishments during Mr. Hopkins career included:

- Implementing operations, organizational, and process changes necessary to support reduced outage duration. After four successive outages of greater than 100 days, durations of 53 days, 35 days and 22.8 days were accomplished.
- Leading a successful transition from two separate unit organizations to a combined site organization as Plant General Manager at Nine Mile Point with continuing improved plant performance.

DETAILS

Tokyo Electric Power Company

- Provide ongoing assessment and evaluation of both corporate and nuclear plant performance
- Developed and implemented plans for nuclear plant monthly Performance Review Meetings
- Developed and implemented nuclear operations performance improvement iniatives for all units

Florida Power and Light - Turkey Point Nuclear Plant

- Developed and implemented nuclear operations performance improvement initiatives
- Performed an assessment of the nuclear operations training programs in preparation for the INPO Training Accreditation Evaluation
- Performed a root cause and developed corrective actions for operations training program probation recovery

First Energy Nuclear Operating Company

- Conducted assessment in preparation for INPO Corporate Evaluation
- Evaluated Organization Effectiveness, Operational Focus, Performance Improvement, Radiation Protection, Work/Outage Management and Chemistry

Constellation Nuclear - Nine Mile Point Nuclear Power Station

Plant General Manager, Units 1 & 2 (August 2001 - July 2004)

- Responsible for safe, efficient operation and testing of the units

- Responsible for Operations, Radiation Protection, Chemistry, and the Corrective Action/Self Assessment Programs
- Emergency Director
- Station Operations Review Committee (SORC) Chairman
- Major Accomplishment: Led a successful transition from two separate unit organizations to a combined site organization as Plant General Manager with continuing improved plant performance.
- Member of Oyster Creek Power Station Nuclear Safety Review Board

Plant General Manager, Unit 1

- Responsible for safe, efficient operation, testing and maintenance of unit
- Responsible for Operations, Radiation Protection, Chemistry, Maintenance, Work Management and Technical Support
- Emergency Director
- Station Operations Review Committee (SORC) Chairman
- Major Accomplishment: Implemented plant modifications to eliminate long standing equipment and design problems resulting in improved plant reliability.

PECO ENERGY COMPANY - Nine Mile Point Nuclear Power Station

Plant General Manager (1999-2004)

- Responsible for safe, efficient operation, testing and maintenance of unit
- Responsible for Operations, Radiation Protection, Chemistry, Maintenance, Work Management and Technical Support
- Emergency Director
- Station Operations Review Committee (SORC) Chairman
- Major Accomplishment: Implemented changes to improve station operations performance standards. INPO November 2000 Plant Evaluation Report Executive Summary stated, "Unit 1 has made substantial progress in meeting these standards" (operations performance standards).

Assistant to Station Vice President – Limerick Generating Station (1998-1999)

- Responsible for evaluation of the "safety conscious work environment"
- Responsible for the recovery of a troubled plant modification to comply with an NRC Confirmatory Order
- Major Accomplishment: Developed and implemented recovery strategies to complete a regulatory required plant modification (thermo-lag reduction) for Units 1 and 2 to comply with an NRC Confirmatory Order.

Training Recovery Manager (1995-1998)

- Responsible for recovery of accredited training programs with emphasis on operations training
- Responsible for development of plan for accreditation renewal of the operations training programs after being on probation.

- Major Accomplishment: Worked with site training director to implement changes necessary to restart the accredited training programs and restore accreditation renewal of the operations training programs.

Training Director

- Responsible for development and conduct of all INPO accredited training programs
- Major Accomplishment: Developed and implemented innovative methods to improve training efficiency and effectiveness.

Senior Operations Manager (1989-1995)

- Responsible for plant operations and operations support functions
- Plant Operations Review Committee (PORC) Chairman
- Emergency Director
- Major Accomplishment: Implemented operations organizational and process changes necessary to support reduced outage duration. After four successive outages of greater than 100 days durations of 53 days, 35 days and 22.8 days were accomplished.

Assistant Superintendent Operations (1986-1989)

- Responsible for plant operations and shift personnel.
- Responsible for operations related unit startup activities for Unit 2

Plant Engineering Support Manager (1979-1986)

- Responsible for development of plant engineering support group which included reactor and System Engineers
- Responsible for operating and surveillance test procedure development, start-up test implementation and review, system start-up and troubleshooting, and system performance monitoring

Nuclear Group Licensing and Regulatory Engineer (1977-1979)

- Responsible for licensing amendments
- Responsible for preparing and approving responses to NRC violations and licensee event reports

System Engineer -Eddystone Fossil Generating Station (1972-1977)

- Responsible for plant system performance, monitoring, and troubleshooting.
- Responsible for plant startup activities for two 400 Mwe oil fired units

EMPLOYMENT

2004 – Present	WSC International	Management Consultant
1999 – 2004	AmerGen (Nine Mile Point)	Plant General Manager
1998 -- 1999	PECO Energy (LGS)	Assistant Vice President

1995 – 1998	PECO Energy	Director of Training
1989 – 1995	PECO Energy	Senior Nuclear Operations Manager
1986 – 1989	PECO Energy	Assistant Ops Superintendent
1979 – 1986	PECO Energy	Plant Engineering Manager
1977 – 1979	PECO Energy	Licensing & Regulatory Engineer
1972 – 1977	PECO Energy	System Engineer

EDUCATION

2001 Institute of Nuclear Power Operations (INPO) Nuclear Plant Manager Course

1988 Senior Reactor Operator License – Limerick Unit 2

1985 Senior Reactor Operator License – Limerick Unit 1

1981 Senior Reactor Operator Certification – Limerick Generating Station

1972 Bachelor of Science; Mechanical Engineering, University of Delaware

1972 Bachelor of Arts; Economics, University of Delaware

DARRYL LE QUIA

SENIOR MANAGEMENT CONSULTANT

RESUME

SUMMARY

Mr. LeQuia brings a disciplined approach to the recovery, operation, maintenance and management of nuclear power stations. He brings 35 years of experience gained while in the United States Navy, the United States Nuclear Regulatory Commission (NRC), and while working in progressively higher management positions at 5 commercial BWR/PWR stations. He was also involved in recovery activities at the Three Mile Island and Peach Bottom Atomic Power Station, following regulatory actions.

DETAILS

Consultant, WSC International, (2003-Present)

1/2008-06/2008: Florida Power & Light-Turkey Point: Assessed Maintenance Department and Work Management performance. Coached personnel to raise standards and improve performance. Developed PI's to track and trend Human Performance relative to maintenance standards. Result: Successful INPO evaluation July 2008.

12/2004-3/2005 Project Manager: Organized, coordinated and led a 15 man team to conduct an "INPO Style" corporate-wide assessment of First Energy and its three stations. The assessment focused on progress, gaps and recommendations associated with transitioning from individual sites to an integrated "Fleet" operation.

4/2003-11/2007: Tokyo Electric Power Company (TEPCO). Reviewed and assessed recovery plans following a regulatory mandated shut down of TEPCO's 17 reactors. Conducted on-site inspections and assessed existing conditions and readiness for restart. Following restart activities, I continued with quarterly assessment trips and also provided management training, in Leadership Development Exchange series for several hundred management personnel. Training areas included: Use of Peer Groups, Process for Process Improvement and Developing a Questioning Mind. Result: TEPCO returned all units to service. Completed training for several hundred of TEPCO's line and support personnel to raise standards.

Director, Maintenance & Support Service, PECO Energy, (PBAPS & LGS) (1993-2002)

In 1993, he promoted to Director of Maintenance. He was responsible for all Mechanical/Electrical/I&C/Planning and Predictive Maintenance for the station. In this new role, he continued to raise performance standards at PBAPS by utilizing lesson's learned from the shutdown period to improve performance in the operational phase.

In 1995, he transferred to PECO's Limerick Generating Station (LGS) to continue to make improvements for the company. At LGS, he was responsible for: Safety, Fire Protection, Security, Business Services, Document Services and Facilities Maintenance. During this period, he also led a team to England to conduct benchmarking with British Energy.

To further broaden his business and management skill, Darryl became involved with the acquisition of other nuclear stations in 1999. He conducted Due Diligence reviews of Clinton Station and TMI for AmerGen, a joint venture between PECO Energy and British Energy. He also successfully served as the Transition Team Leader for the support areas during acquisition of TMI and Oyster Creek Stations. Ultimately, he served as the Director Planning and Integration for Exelon's acquisition of Sithe Energies, Inc. As the Sr. Project Manager, he was responsible for the design and integration of all transition related activities. The $1.5 Billion merger and acquisition was completed on time and under budget.

Superintendent, Plant Services, PECO Energy (1988-1993)

In 1987, the NRC shut down PECO Energy's Peach Bottom Atomic Power Station (PBAPS). Mr. LeQuia was recruited by PECO to assist with the recovery of PBAPS. As the Superintendent of Plant Services, he was responsible for the Radiological Controls, Radioactive Waste and Chemistry Programs. To effectively recover these programs, he developed and implemented Recovery Plans, presented Recovery Plans and performance indicators to Sr. Management and the NRC, developed long-term system lay-up initiatives with EPRI for investment protection, and lead a team to Japan for benchmarking with Tokyo Electric Power Co.

Regional Inspector, NRC (1986-1987)

In 1986, Darryl became an NRC Inspector in Region 1. He was responsible for monitoring compliance to regulations for 10 different stations. He was assigned to inspect poor performing stations and help raise performance to industry standards.

Health Physics Supervisor, Wisconsin Electric Power Co. (1983-1986)

Served for as a Sr. Radiological Controls Supervisor for Wisconsin Electric's Point Beach Station.

Quality Assurance Engineer, General Public Utilities TMI (1980-1983)

In 1980, after 8 years with the Naval Nuclear Power Program, Darryl began his commercial reactor experience at General Public Utility's-Three Mile Island Nuclear Station (TMI). There he was involved for 3 years with the cleanup and dismantling of TMI's crippled Unit-2 reactor.

EMPLOYMENT

2003 – Present	Senior Management Consultant	WSC International
1993 – 2001	PECO Energy (PBAPS & LGS)	Director of Maint. & Support Service
1988 -- 1993	PECO Energy	Superintendent Plant Services
1987 – 198	PECO Energy	Health Physics Technician

1986 – 1987	US NRC	Regional Inspector
1983 – 1986	Wisconsin Electric Power Co.	Health Physics Supervisor
1980 – 1983	General Public Utilities -TMI	Quality Assurance Engineer
1972 – 1980	U.S. Navy Submarine Service	Supervisor, Lead Eng. Lab Technician

EDUCATION

Graduate studies, Health Physics; Georgia Institute of Technology; 1987

Bachelor of Professional Studies, Water Treatment Technology; Elizabethtown College; 1983

Graduate; U.S. Naval Nuclear Power Program, 1973

Professional Development and Achievements:

- Board of Directors, Pottstown/Upper Perkiomen Valley United Way
- Representative for Limerick Generating Station, Tri-County Chamber of Commerce
- General Electric Executive Senior Reactor Operator Course, 1990
- Three Mile Island Management Certification Course, 1997
- Limerick Generating Station Management Certification Course, 1998
- American Nuclear Society (1986-1991)
- Former DOE "Q" Clearance; Navy "Secret"

BRUCE ALLSHOUSE

EXECUTIVE VICE PRESIDENT – WSC INTERNATIONAL

RESUME

SUMMARY

Mr. Allshouse is an experienced Manager with a diverse utility background and is Executive Vice President for WSC International's Management Consulting Services. He has worked in Nuclear and Fossil plants, Human Resources, and Transmission Maintenance. His areas of expertise include nuclear maintenance, project management, process reengineering, organizational design, and change management.

Mr. Allshouse is also actively involved with supporting Tokyo Electric Power Company with implementing business and technical process improvements within their Nuclear Organization.

Mr. Allshouse's experience in Nuclear Generation includes positions as the Director of Maintenance at Peach Bottom Atomic Power Station, the Corporate Training Director for all of Nuclear, and the Support Manager at Limerick Nuclear Generating Station where he implemented many of the organizational changes made as a result of the Peach Bottom shut down. He also held various positions in Fossil Generation, Engineering, and Human Resources within PECO Energy Co. He led or participated in four major organizational redesign projects in nuclear, fossil and corporate organizations.

In 1993, Mr. Allshouse was selected to help redesign the entire Nuclear Organization; a major improvement initiative to gain efficiency and decrease costs as part of PECO's continuing recovery efforts after the Peach Bottom shut down. In 1999, he led a cross-functional team of employees from Limerick, Peach Bottom, and the recently purchased Three Mile Island Nuclear Plant to design organizational changes, develop synergies, and identify best practices for implementation at all three sites and corporate headquarters. Subsequently, he led similar efforts at Oyster Creek and Clinton Nuclear Power Plants after they were purchased.

DETAILS

Executive Vice-President, WSC International (2003 – Present)

Senior Project Manager, Wayne, PA (1999 - 2000)

Develop organizational structure, staffing and work processes for newly acquired nuclear plants. (Staff 17, Budget $3M)

- Created structure and staffing plan for first nuclear plant acquisition; estimated operations and maintenance savings of $25M/year; accomplished within 90 days.
- Completed organizational design and staffing for second nuclear plant acquisition in less than 30 days; reduced workforce from 750 to 600 (20%).

Maintenance Director, Delta, PA (1996-1999)

Managed the planning, execution and testing of all nuclear plant equipment. (Staff 230, Budget $28M)

- Completed all maintenance activities under budget ($28M); included major plant shutdown for refueling.
- Improved Craft/Management Trust Index by 17 percentage points based on annual Employee Value Survey. Revised work management process and reduced work cycle time from 90 days to 15 days.

Human Resources Director, Philadelphia, PA (1994-1996)

Directed staffing, training and succession planning functions for entire corporation. (Staff 60, Budget $17M)

- Consolidated 10 decentralized training groups of 100 into centralized department of 40, saving $6M/year.
- Redesigned and automated staffing process; reduced time to fill positions from 6 months to 2 months.
- Created and implemented succession planning process for entire corporation; all company divisions completed within 1 year.

Director of Corporate Labs, Valley Forge, PA (1993-1994)

Provided chemistry, metallurgical and calibration services for PECO Energy and external companies. (Staff 65, Budget $11M)

- Negotiated and signed first contract for lab work with another utility; generated $500K revenue/year.
- Redesigned customer interface process and reduced calibration cycle time from 30 days to 10 days.
- Negotiated resource sharing agreement with customers and reduced equipment requirements by $500K.

Nuclear Training Director, Norristown, PA (1993-1994)

Delivered centralized training programs for nuclear generating stations. (Staff 40, Budget $7M)

- Standardized lesson plans and increased instructor class time from 30% to over 60%.
- Reduced instructors from 65 to 40 saving $2.5M/year.
- Reduced annual training facility operating costs from $1M to $650K in two years.

Administration Superintendent, Pottstown, PA (1989-1993)

Managed all support functions at nuclear plant including Emergency Planning, Document Control, Fire Protection, Industrial Safety and Facilities. (Staff 60, Budget $11M)

- Replaced 125 vendors with fixed price service contracts saving over $1 million per year.
- Created and staffed Industrial Safety Group; reduced site accident rate by 25%.
- Automated document control function, improving customer satisfaction and reducing staff by 50%.

Reengineering Special Projects (1991-1992)

Participated in re-engineering efforts on three major company projects.

- Led team of consultants and company employees who redesigned corporate laboratory function; decentralized and downsized by one-third.
- Led corporate-wide team that developed project management process for entire company and identified information technology platform.
- Redesigned nuclear department structure, staffing and work processes with 7 other employees; developed implementation plans that led to 20% reduction in staffing.

Various Supervisory and Management Positions (1969-1989)

Managed groups in Engineering, Transmission Maintenance, Nuclear Licensing and Computer Operations.

EMPLOYMENT

2003 – Present	Business Process Improvements	Consultant
PECO Energy:		
1999 – 2000	Organizational Design Project Manager	Director
1996 – 1999	Nuclear Plant Maintenance	Director
1994 – 1996	Human Resources Director	Director
1993 – 1994	Nuclear Training Director	Director
1993 – 1994	Corporate Labs Director	Director
1989 – 1993	Administration Superintendent	Superintendent
1991 – 1992	Reengineering Special Projects	Senior Engineer
1969 – 1989	Various engineering positions	Engineer

EDUCATION

B.S. Electrical Engineering, Lafayette College, 1969

CLEARANCE

Nuclear Plant Assess (1975 – 2003)

RICHARD S. SELVERIAN

Former Nuclear Plant Senior Manager

BIOGRAPHY

Mr. Selverian is president and founder of Integrated Personnel Solutions, an innovative staffing management company providing a full range of staffing services for small- to large-sized organizations.

Mr. Selverian draws on more than 20 years of experience as an executive in strategic planning, finance, human resources, and operations organizations to provide unique solutions to business problems.

In the area of strategic and business planning, Mr. Selverian has been responsible for a host of projects including designing and facilitating a strategic planning process for a large transportation company; designing and managing a strategy and business planning effort for seven (7) startup or expanding energy service companies as part of a venture capital portfolio; and developing the first business plan for many operating and support service organizations within a fortune 500 company.

Additionally, Mr. Selverian has extensive process and systems management experience. He has successfully led several reengineering and redesign processes in many disciplines including nuclear power engineering, training and development, staffing, and mergers and acquisitions. His process-management initiatives have included a range of projects from developing a thermal performance modeling system resulting in $5 million per year of additional revenue to designing and managing an interactive voice response/intranet system for 10,000 employees.

Mr. Selverian has lectured extensively on change management at The Wharton School of the University of Pennsylvania, on performance measurement at numerous industry conferences, and on "Learning the Language of Business" for human resource professionals.

Mr. Selverian is also the author of "Outsourced Staffing," an essay examining the concept of outsourced staffing and providing guidance when heading down that road.

Mr. Selverian graduated with honors from Columbia University with a Masters in Business Administration (MBA) and holds a Bachelor of Science (BS) degree in Engineering from Swarthmore College.

FAROUK D. BAXTER

ELECTRICAL SYSTEMS EXPERT

RESUME

SUMMARY

Mr. Baxter is an industry recognized expert in nuclear plant electrical power systems, with more than forty years professional experience associated with nuclear power plants worldwide and more than thirty years involvement with national and international nuclear standards development

Capabilities include:

- Safety System Inspections and Engineering Quality Oversight
- Conceptual Planning, Feasibility Studies, and Special Projects
- Expert Testimony, Second Opinions, and Interpretations
- Operability Determinations, Power Upgrades, and Licensing Renewal.
- New Plant Combined Operating License Reviews

DETAILS

- PROFESSIONAL EXPERIENCE: (CONSULTING) (1992-Present)

US NUCLEAR REGULATORY COMMISSION.

Wolf Creek - Safety System Engineering Inspection (SSEI). 1998

Callaway - Safety System Engineering Inspection (SSEI). 1998

Browns Ferry - Safety System Engineering Inspection (SSEI). 1998

D.C. Cook - Expanded System Readiness Review (ESRR). 1999

Wolf Creek - Safety System Engineering Inspection (SSEI). 2000

Grand Gulf - Safety System Design & Performance Capability Inspection (SSDPCI). 2000

Browns Ferry - Safety System Design & Performance Capability Inspection (SSDPCI). 2002

Davis Besse - Corrective Action Team Inspection (CATI). 2003

Three Mile Island - Safety System Design & Performance Capability Inspection (SSDPCI). 2003

Dresden - Quad Cities - License Renewal Application (LRA) Audit. 2003

Limerick - Safety System Design & Performance Capability Inspection (SSDPCI). 2003

Palo Verde - AIT Follow-up Inspection. 2004

Brunswick - Triennial Fire Protection Inspection (TFPI), 2004

Kewaunee - Pilot Design/Engineering Inspection, 2005

Harris - Triennial Fire Protection Inspection (TFPI), NFPA-805 Pilot, 2005

Grand Gulf - Component Design Basis Inspection (CDBI), 2006

Hatch - Component Design Basis Inspection (CDBI), 2006

Diablo Canyon - Component Design Basis Inspection (CDBI), 2006

Comanche Peak - Component Design Basis Inspection (CDBI), 2006

Oyster Creek - Component Design Basis Inspection (CDBI), 2007

Fermi - Component Design Basis Inspection (CDBI), 2007

Clinton - Component Design Basis Inspection (CDBI), 2007

Fermi – COLA Review, 2008

Victoria County – COLA Review, 2008

Regulatory Guide 1.89 – Proposed Revision, 2008

UTILITY WORK ASSIGNMENTS

Perry - Diesel Generator Latent Issues

D. C. Cook - Assessment of Emergency Diesel Generators and Electrical Systems

Point Beach

- Systems Assessment Team. Reviewed and assessed CCW, SW, and electrical protective relaying for adequacy in anticipation of NRC inspection.

Nine Mile Point 1 & 2

- Development of Transmission System Interface Specification and Procedure.
- Independent Technical Oversight (ITO) Team. Review of engineering products, such as design changes, temporary modifications, calculations; and their effective integration with technical specification, licensing design basis, and operating requirements.

Vermont Yankee

- Design Basis Document Validation.
- Self-assessment Team investigating programmatic shortcomings.
- NRC AE Inspection, Expert Panel.
- Principal author and editor for FSAR (Chapter 8) Improvement Project.
- Development and review of calculations, studies and design basis documents.
- Applicability of IEEE 279-1968 requirements to Vermont Yankee.
- Evaluation of employee concerns allegation.

Brunswick 1 & 2

- 3rd. Party Assessment Team investigating the technical adequacy of over 100 design change packages.

Salem 1 & 2 and Hope Creek

- Applicability of IEEE 279-1971 requirements to the Salem Generating Station.
- Assessment of Battery Design Margin.

Byron Station - Preparatory audit in preparation for NRC EDSFI Inspection.

LaSalle Station

- Investigation of failures of electrical equipment resulting in numerous plant trips. Determination of root cause, with recommendations to management.

Millstone 1

- Assessment and applicability of the application of Single Failure Criterion.
- Author of FSAR (Chapter 8) rewrite.

Millstone 2 - Mini-audit in preparation for NRC EDSFI Inspection.

Millstone 3 - 10CFR50.54F Vertical Slice Review Team.

Seabrook

- Independent Review Team investigating breakdown of programmatic controls resulting in NRC violations.
- 10CFR50.54F Independent Vertical Slice Review Team Investigating the Emergency AC System.

Pilgrim

- Development of MOV torque and heater sizing calculations.
- Preparatory audit for NRC Risk Based Assessment on the EDG and HPCI Systems.

Beaver Valley 2 - Safety System Functional Evaluation (SSFE) Team.

Maine Yankee

- Independent Safety Assessment Team (ISAT), Expert Panel.

- Audit of vendor facilities to assure commercial dedication of switchgear.
- Investigation into cable separation violation causal factors.

Indian Point 2

- Investigation and Root Cause Analysis of problem involving unexplained fuse actuations of emergency diesel generator auxiliaries.

Calvert Cliffs 1 & 2

- Independent assessment of potential problems with cables installed in tray and duct.

Monticello

- Calculation Assessment

Prairie Island

- Calculation Assessment

- PROFESSIONAL EXPERIENCE: (EMPLOYMENT) (1962-1992)

Yankee Atomic Electric Company (1969-1992)

- Specialist - Electrical Engineering, Plant Support Department.
- Principal Engineer, Plant License Renewal Department.
- Lead Electrical Engineer, Maine Yankee Project.
- Engineering Manager, Seabrook Project.
- Manager of Electrical Engineering.
- Senior Electrical Engineer, Electrical and Control Engineering Department.

Commonwealth Associates, Inc. (1968-69)

Atomic Power Constructions, Ltd. (1966-1968)

Bechtel India Ltd. (1964-66)

Tata Power Company. (1962-64)

EMPLOYMENT

2003 – Present	WSC International LLC	Executive Vice President
1999 -- 2003	WSC INC (EPRI)	Director of Maintenance Services
1996 – 1999	CSI/Emerson	Utility Market Director
1995 -- 1996	Mantenance & Diagnostics LLC	President & COO

1991 – 1994	WSC Inc. (EPRI - M&D Center)	Engineering Manager
1988 – 1991	WSC Inc. (EPRI - M&D Center)	Lead Project Engineer
1987 – 1988	Canus, Inc. (Peach Bottom Nuclear Plant	I&C Engineer
1981 – 1987	Various Engineering Firms	I&C Engineer

EDUCATION

Bachelor of Technology Degree (B.Tech.) in Electrical Engineering. Indian Institute of Technology (IIT), Kharagpur, India. (1962).

Registered Professional Engineer (PE) in Massachusetts.

Advanced Management Training Programs at Babson College, Bentley College, Boston University, and Electric Council of New England.

PROFESSIONAL AFFILIATIONS AND HONORS

Senior Life Member IEEE, Member IEEE Power Engineering Society Nuclear Power Engineering Committee, Member IEEE Power Engineering Society Nuclear Power Engineering Committee Subcommittee on Auxiliary Power Systems, IEEE, Power Engineering Society Distinguished Service Award - 2002.

SELECTED PUBLICATIONS

- "IEEE 603/IEEE 308 Ad Hoc Committee Report." Paper presented at the 1978 Symposium on Nuclear Power Systems held in Washington D.C.
- "The Dangers of Bypassing Thermal Overload Relays in Nuclear Power Plant Motor Operated Valve Circuits." Paper presented at the IEEE Power Engineering Society Winter Meeting held in New York in 1980. IEEE Transactions on Power Apparatus and Systems, Volume PAS-99, No. 6, November/December 1980.
- "Automatic Trip of Off-Site Power Supplies on Experiencing Abnormal Voltage Variations on Nuclear Power Plant Emergency Buses." Paper presented at the 1980 EEI Systems Planning Meeting held in New Orleans.
- "Automatic Disconnection from the Preferred Power Supply due to Degraded Grid Relaying Actuation, Arguments For and Against." Paper/Panel Discussion presented at the 1997 Nuclear Science Symposium, Albuquerque, NM and published in IEEE Transactions on Nuclear Science, Vol. 45, August 1998.
- "Questionable System Grounding Practices at Nuclear Power Plants." Paper published in IEEE Transactions on Energy Conversion. Vol. 17, No. 2, June 2002.
- "Misinterpretation and Disregard for the Single Failure Criterion." Paper presented to the IEEE's Nuclear Power Engineering Committee at Meeting 04-01 in January 2004.
- "The Next Nuclear Accident - A Perspective on Energy Sufficiency." Column in the IEEE Power & Energy Magazine, January/February 2006.

ANIELLO (NEIL) L. DELLA GRECA, PE

ELECTRICAL SYSTEMS EXPERT

RESUME

SUMMARY

With forty years of engineering experience and more than thirty in the nuclear field, Mr. Della Greca currently provides part-time consulting services to the US Nuclear Regulatory Commission (NRC), where he held the position of Senior Reactor Inspector in the Electrical Branch until he retired in December 2004. In addition, since March 2007, he provided consulting services to InfoZen, Inc. in the update of the US NRC Standard Review Plan. Also, since November 2005, he has provided training for an industrial safety course at the Business & Industry Training Center of the Gloucester County College.

While working at the NRC, he led and conducted numerous engineering and operational team inspections, including safety system design performance, electrical distribution system functionality, instrumentation and controls, fire protection, problem identification and resolution, and maintenance. In addition, he led and performed many event response inspections and participated in various special assignments, including a review of Rosemount transmitter performance issues, non-condensable gas concerns at Pilgrim, and electro-magnetic/radio frequency interference at Millstone. As a Senior Inspector, he was also responsible for mentoring and training junior inspectors in technical as well as nuclear regulatory requirement areas. While at the NRC, he received twelve awards and several other commendations for my performance in various assignments and activities.

Prior to joining the NRC, Mr. Della Greca worked for approximately 15 years at Stone & Webster Engineering where he held progressively responsible positions in the Electrical and I & C Sections. As a Controls Engineer, he designed and reviewed the instrumentation and controls of the electrical and most balance-of- plant systems for the Nine Mile Point 2 (NMP2), River Bend, and North Anna 3 Projects. In addition, he coordinated with General Electric the design of the nuclear steam systems. In the supervisory capacities, he had technical and administrative responsibilities for various projects, including development of failure modes and effects analysis for the NMP2 project, design of plant modifications for the Oyster Creek station, development and implementation of the NMP2 maintenance and environmental qualification programs. In addition, he was assigned as Consultant to Ansaldo, an Italian Architect-Engineer, where he implemented programs for environmental qualification of equipment and electrical separation using US standards and regulatory requirements. He also assisted in the design and environmental qualification testing of the reactor protection system for an experimental sodium cooled reactor.

Prior to joining Stone & Webster Engineering, he designed switchgear equipment and system protection and controls for General Electric Company. He also designed industrial instrumentation and controls for Pennwalt.

DETAILS

- Consultant - US Nuclear Regulatory Commission, Rockville, MD (10/05 TO 9/07)
  Led assessment teams at various nuclear facilities to evaluate their readiness to respond to design basis threats and, more recently, to evaluate licensees' program for control and accounting of special nuclear materials.

- Senior Consultant - INFOZEN, INC, Rockville, MD (3/07 to 6/07)
  Provided consulting services in the update of Chapters 7 and 8 of the US NRC Standard Review Plan. The two chapters address review requirements for instrumentation and control systems and electrical systems at nuclear power plants.

- Trainer - GCC Business & Industry Training Center, West Deptford, NJ (9/05 to Present)
  Provided training for the Association of Reciprocal Safety Councils, Inc. (ARSC) Basic Orientation Plus industrial safety course. In addition, translated the course in Italian and provided training to Italian speaking industrial workers.

- Senior Reactor Inspector, US Nuclear Regulatory Commission, King of Prussia, PA (12/88-12/04)

  Led and participated in numerous multi-disciplinary inspections of nuclear facilities throughout Region I. The inspections encompassed many different areas of plant activities, including design, installation, testing, maintenance, operations, and fire protection.

  As Team Leader, ensured that inspections were correctly focused and team findings properly characterized. As Team Member, evaluated system design and performance, instrumentation and controls, adequacy of set-point calculations, environmental qualification of equipment, impact of design changes and plant modifications, maintenance and test activities, and plant operating and emergency procedures. As Electrical Specialist, evaluated electrical design calculations, including load, short circuit, and voltage drop analyses, equipment protection and breaker coordination studies. In addition, evaluated plant events to confirm root cause of event and acceptability of problem resolution. As Senior Inspector, provided mentoring and instructions to junior engineers in technical as well as regulatory requirement areas.

  Familiar with current NRC inspection program and regulations.

- Senior Controls Site Engineer, Stone & Webster Engineering Corp. (9/73-12/88)
  Senior Controls Site Engineer Nine Mile Point – Unit 2 (7/86 – 12/88)

  Responsible for the review of plant design modifications and their impact on environmental qualification of safety-related equipment. Also, performed equipment life extension calculations and provided technical assistance in the development of programs for maintenance, surveillance and dedication of commercial grade spare parts. Conducted failure modes and effects analysis of plant systems.

Consultant, Progetto Elementi Combustibili (5/85 – 7/86)

Advised Ansaldo, SPA (Genova, Italy), on the interpretation of NRC requirements and US Standards relative to equipment qualification and electrical separation criteria. In addition, developed an overall qualification program for the P.E.C. (a sodium cooled reactor); trained engineering personnel in the review of equipment qualification documents; and assisted Esacontrol (an Italian Testing Laboratory) in the preparation of plans and procedures for qualification testing of safety-related equipment.

Assistant to Project Engineer, Nine Mile Point – Unit 2 (5/82 – 5/85)

Assisted the Project Engineer in the development, planning and administration of the NMP2 overall qualification effort. Duties included preparation and administration of budgets and schedules, writing project procedures, training personnel and coordination of equipment qualification activities. Also, served as NMPC's representative on a multi-plant Technical Review Committee and assisted General Electric in the development of its harsh environment qualifications program, ensuring that it adequately met current NRC requirements and NMP2 design criteria.

Lead Controls Engineer, Oyster Creek (12/79 – 5/82)

Directed the I&C group in the execution of engineering tasks for the upgrading of various plant systems. Responsibilities included administration of budget and schedules, and preparation and technical review of design, procurement, and installation documents. Design included development of appropriate calculations.

Principal/Controls Engineer, Nine Mile Point – Unit 2 (1/77 – 12/79)

As Principal, was responsible for the administration, planning and development of system failure modes and effects analysis. As Controls Engineer, prepared instrument loop, logic and elementary diagrams for BOP and NSSS plant systems, including water and steam systems, plant and heater drains, normal and emergency electrical power systems, and reactor protection. In addition, prepared instrument set-point calculations for systems assigned.

Controls Engineer, North Anna - Unit 3 and River Bend (9/73 – 12/76)

Developed elementary diagrams for plant systems, including protection and control of emergency diesel generators, normal and emergency electrical power systems, and fire protection.

- Electrical Design Engineer, General Electric Co. Switchgear Division., Philadelphia, PA (1/68-9/73)
  Designed low and medium voltage switchgear equipment for commercial, industrial and power generation facilities. Duties included system analysis, design of protection schemes and preparation of control schematics and wiring diagrams.

- Electrical Design Engineer, Pennwalt Chemical Corp. – Stokes Div., Philadelphia, PA (6/66 to 1/68)
  Designed instrumentation and controls for industrial projects, including impregnating chambers and freeze-dryers. Duties entailed preparation of control schematics and panel layouts, equipment installation and startup.

- Administrative Assistant, COMPREHENSIVE DESIGNERS, Philadelphia, PA (11/61 – 6/66)
  Responsible for the inception and maintenance of a computerized engineering information retrieval system, review of engineering releases, and preparation of test data and source programs.

EDUCATION

University of Pennsylvania –Philadelphia, PA, BSEE – 1966

University of Naples, Italy – Education (2 years)

Numerous NRC, Stone & Webster, and General Electric Continuing Education Courses

REGISTRATIONS

Professional Engineer – Commonwealth of Pennsylvania

Bachelor or Science - Mechanical Engineering - 1982 Drexel University, Philadelphia, PA

RICK LEONARD

NUCLEAR INDUSTRY SENIOR EQUIPMENT RELIABILITY ENGINEER

RESUME

SUMMARY

Mr. Leonard has over 28 years experience in the Nuclear and fossil industry associated with Equipment Reliability projects. Over 20 years of his career was dedicated to various technical and engineering positions at the Peach Bottom Nuclear Power plant in the areas of condition monitoring technology applications, Predictive Maintenance, and component and System Engineering. For the past eight years Mr. Leonard has provided equipment reliability consulting services to organizations worldwide with a specific focus on supporting the Electric Power Research Institute with the Large Electric Motor User Group; authoring and or supporting the development of many technical reports related to this topic.

DETAILS

Senior Project Manager/Director

- Performing EPRI RMT Assessments at various nuclear facilities worldwide.
    - Includes the use of Remote Monitoring Technologies (RMT) to decrease workers radiation exposure and increase workforce efficiencies.
- Planned, coordinated and scheduled EPRI project ‘Location Tracking in Nuclear Power Plants.’
    - Involves the real-time tracking of personnel, equipment (radwaste storage casks), and specific tools in NPP’s.
    - Integration of real-time dose reading with real-time personnel location for RP continuous monitoring.
- Planned, coordinated and scheduled EPRI project ‘Tools for Large Electric Motors.’
    - Included the use of Electro Magnetic Induction (EMI) analysis and Partial Discharge (PD) analysis for early detection of motor insulation faults and diagnosis.
    - Included the development of a cost effective EMI detector.
- Planned, coordinated and scheduled EPRI project ‘Advanced Predictive Maintenance for Large Electric Motors.
- Providing consultations to ARAMCO in areas of Predictive Maintenance (PdM), equipment reliability, and troubleshooting of malfunctioning equipment.
- Providing consultations to Saudi Electric Company in areas of Predictive Maintenance, equipment reliability, and troubleshooting of malfunctioning equipment
    - Including consultations on the specifications and installations of plantwide rotating equipment monitoring system.
- Providing consultations to SWCC in the selection of water pipe monitoring equipment and the diagnosis of impending pre-stressed piping failures.

Maintenance & Diagnostics LLC – Project Manager (1995 – 2000)

Project Manager

- Planned, coordinated and scheduled EPRI project 'Predictive Maintenance for Large Electric Motors.
    - Included ten electric utilities throughout the U.S.
- Established motor-based PdM programs throughout the U.S. and India.
- Defined budget and resource needs for various other EPRI projects.

Philadelphia Electric Company – Nuclear Engineering (1985 – 1995)

- Routinely supervised a cross-functional and multi-talented company and contract work force in the troubleshooting of failed and/or malfunctioning plant rotating, and critical safety and generation equipment.
- Initiated the start of several Predictive Maintenance (PdM) programs company-wide.
- Obtained training in vibration analysis, thermography, acoustics emissions, ultrasonics, ultrasonic flow measurements, and large-scale data acquisition.

Philadelphia Electric Company – Instrumentation & Controls (1980 – 1985)

- Performed installation, calibration, and troubleshooting of plant control and safety systems.
- Obtained training in all plant systems, associated equipment, and procedures, including those critical to safety shutdown.
- Obtained First Class FCC license for the maintenance of radio equipment requiring FCC compliance.

EDUCATION

- BA Degree, Business – Organizational Management, Eastern University.
- BS Degree, Electrical Engineering, Johns Hopkins University.
- AA Degree, Electronics Engineering, Harford Community College
- AA Degree, Broadcast Communications, Harford Community College
- Obtained training in EMI Signature Analysis and Partial Discharge
- Obtained training towards Certified Energy Engineering Certification.
- Presently working towards an MBA at Warwick University, UK

FRANK X. MC CREESH, P.E.

CABLE SEPARATION EXPERT

RESUME

SUMMARY

Twenty-one years of experience in the disciplines of fire protection and civil engineering with expertise in design and analysis various fire suppression and protection systems, performance of analyses, studies and inspections of these systems and program development.

DETAILS

Principle Engineer, Triad Fire Protection Engineering Corp. (1997 to Present)

Provide fire protection technical consulting and expertise to several clients both in the nuclear power industry and commercial building industry. Projects included:

- NRC Region 2 Inspector as the fire protection specialist for fire protection triennial inspections. Plants inspected include the H. B. Robinson Plant, North Anna Station, St. Lucie Plant and Vogtle Station.
- Performed nuclear oversight plant inspections for the Exelon fleet of nuclear plants as well as for the PSEG nuclear plants.
- Compiled comprehensive program for inspection, testing and maintenance of fire protection systems and equipment at numerous PSEG fossil power stations.
- Performed due diligence assessments of the fire protection programs at Oyster Creek, Vermont Yankee and Nine Mile Point Nuclear Generating Stations.
- Lead Fire Protection Auditor for selected generating facilities
- Performed hydraulic and water hammer analysis for power plant piping systems
- Supported Susquehanna Steam Electric Station for an NRC Inspection of their fire protection program.
- Provided training for installation of Thermo-Lag fire barrier systems.
- Performed assessment and modification to the detection and suppression systems at the Susquehanna Nuclear Station.
- Provided analysis support of 10CFR50 Appendix R fire protection issues and provided fire protection evaluations to resolve open issues
- Performed modifications to convert numerous automatic CO2 systems to manually actuated systems.
- Designed and qualified electrical raceway fire barrier systems to enable various power stations to meet the requirements of 10CFR50, Appendix R.
- Performed comprehensive fire detection study for Peach Bottom Atomic Power Station for selected high risk areas.
- Compiled a computer based combustible loading analysis for Salem Generating Station and Peach Bottom Atomic Power Station.
- Performed hydraulic calculations to assess adequacy of Limerick Generating Stations fire suppression systems.

- Project Manager and Chief Engineer for the installation and testing of an emergency power generator for a major commercial facility.
- Performed technical and management oversight function for thermo-lag upgrade project at Limerick Generating Station.
- Reviewed and provided recommendations for the Mecatiss Project at Oyster Creek Generating Station. Performed structural condition assessment for the Philadelphia Fire Academy's Burn Building Training facility to meet the requirements of NFPA 1403 for fire department live fire training.

Manager, Fire Protection & Civil/Structural Engineering Group, Exelon Nuclear (1989-1997)

Supervised activities involved with the Fire Protection Program, including combustible loading control, suppression system analysis, NRC interface, regulatory compliance reviews, containment analysis, penetration seal design and qualification, fire barrier system analysis and training issues.

Senior Engineer, Fire Protection and Civil Engineering Group, Exelon Nuclear

Support of the Fire Protection Program at Peach Bottom and Limerick Generating Stations. Supported outages to address fire protection issues.

Group Supervisor-Appendix R Modification Group, Nuclear Plant Engineering, Pennsylvania Power & Light Co. (1987-1989)

Supervised several fire protection engineers to support Susquehanna's compliance with 10CFR50 Appendix R. Presented results to the NRC staff and received approval and complimentary remarks on conservative approach top structural analysis. Authored numerous deviation requests for alternate compliance to Appendix R for Susquehanna's fire protection program.

Project Engineer-Nuclear Plant Engineering and Transmission Engineering, Pennsylvania Power & Light Co. (1978-1987)

Performed numerous engineering tasks in support of the start up of the Susquehanna Nuclear Plant and the design construction of transmission lines to support the electric supply from the station.

EMPLOYMENT

1997 to Present	Triad Fire Protection Engineering Corp.	Principal Engineer
1989-1997	Exelon Nuclear	Manager-Fire Pr
1987-1989	Pennsylvania Power & Light Co.	Group Supervisor/Project Engineer

EDUCATION

Drexel University, Bachelor of Science, Civil Engineering, 1978

President of Engineering Class, 1978

Registered Professional Engineer

Professional Member, National Fire Protection Association

Professional Member, American Society of Civil Engineers

Professional Member, Structural Engineering Institute

Resources provided by Triad Fire Protection Engineering Corp.

BRIAN W. MELLY, P.E.

CABLE SEPARATION EXPERT

RESUME

SUMMARY

Twenty-nine years of experience in the discipline of fire protection engineering. Specialized in fossil, industrial and nuclear fire protection engineering and has considerable expertise in the develop of fire hazards analyses, design and evaluation of fire suppression systems, fire protection inspections and audits, code analyses and interpretations, computer fire recreation analysis, pre-fire planning and hydraulic modeling of water supply systems. Currently involved in the testing of building smoke management systems, fire wrap upgrades and CO2 system conversions.

Performed structural steel survivability analyses for several utilities to predict performance of the fire barrier support steel in a fire. Integrated effects of Thermo-Lag 330-1 material burning into the structural steel survivability methodology.

Involved in identification and resolution of Thermo-Lag fire barrier issue for several utilities. Efforts included formulation and implementation of a comprehensive resolution strategy designed to achieve regulatory compliance without wholesale replacement of existing barriers. Prior to involvement in the Thermo-Lag issue, Mr. Melly was active in the development of methodologies for the analysis of complex water supply networks. Participated in the development of a hydraulic analysis software program that simplifies the analysis of water supply systems.

Computer modeling to predict and model the growth of fire and its effect in buildings. Reviewed and evaluated all currently available fire protection hydraulic software for the Society of Fire Protection Engineers. Presented his review nationwide and has been published in the society of Fire Protection Engineers and National Fire Protection Association.

Developed and trained all NRC Regional Inspectors on performance of the Triennial Fire Protection Inspection. Participated in 8 risk informed triennial fire inspections for the NRC in Regions 2 and 3. Mr. Melly was a member on the USNRC's fire modeling PIRT panel.

DETAILS

Experience includes:

- Design and evaluation of sprinkler and CO2 fire suppression and detection systems
- Design and evaluation of underground and aboveground fire water distribution systems
- Installation supervision and acceptance testing of fire suppression, fire detection and fire water distribution systems.
- System analyses, system design specifications and purchase specification development for fire suppression systems
- Pre-fire planning
- Experienced in CFAST, FDS, CONTAM, EXIT89, FIREST3, BRANZFIRE, SAFIR
- Fire protection inspections and audits

- Penetration Seal Program evaluations
- Model building code and fire code analysis and interpretation
- Computer aided fire modeling
- Computer aided network hydraulic modeling
- IPEEE Five methodology evaluations
- Thermo-Lag, Darmatt, E-Mat, FS-195, MECATISS< HYMEC, MT Fire barrier evaluations
- Design Basis Documents (DBD) Program, system and topical development
- Fire protection program reviews

EMPLOYMENT

1994 to Present Triad Fire Protection Engineering Corp. Vice President Engineering/Principal Engineer

1989-1994 PECO Energy Senior Engineer

1984-1989 Professional Loss Control Senior Fire Protection Engineer/Project Manager

1978-1984 Black & Veatch Consulting Engineers Fire Protection Design Engineer

EDUCATION

University of Maryland, BS, Fire Protection Engineering, 1978

University of Maryland, M.E., Fire Protection Engineering, 2005

University of Kansas, MBA, Graduate Courses, 1980-81

National Bureau of Standards, Fire Modeling Courses

Society of Fire Protection Engineers-Fire Hazards Analysis Courses

Registered Professional Engineer – Delaware and Pennsylvania (registered in discipline of Fire Protection Engineering)

Society of Fire Protection Engineers – Member Grade

National Fire Protection Association – Member Grade

Resources provided by Triad Fire Protection Engineering Corp.

EDWARD (TED) M. NICHOLS, II

NUCLEAR PLANT DIRECTOR OF ENGINEERING

RESUME

SUMMARY

Mr. Nichols has over twenty five years of operating plant experience in the commercial nuclear power industry focused on plant engineering and maintenance. This experience consisted of twenty one years with Northeast Utilities and Florida Power and Light and four years as a consultant serving the commercial nuclear power industry. In addition, his engineering career started with seven years as a marine engineer in the U.S. Merchant Marine.

Mr. Nichols has held several key technical and management positions during his twenty five years in the electric power industry. Much of this experience has focused on equipment reliability associated with plant operations, system readiness and asset management.

DETAILS

Executive Consultant

Currently, Mr. Nichols is an Executive Consultant providing consultant services to the power generation industry and recent or ongoing clients include Bruce Power-Canada, Yucca Mountain Project, and Harvard Medical School.

Vice President - Altran Solutions Corp 2005 - 2007

Mr. Nichols was the Vice President and Managing Director of Altran Solutions Corp which provided engineering consulting services to the commercial nuclear power industry. In this role Mr. Nichols provided overall financial and management direction/responsibility for the company. Altran Solutions consisted of 95 employees located in six offices in the US and Canada (headquarters; Boston, Mass). Services provided consisted of technical support, materials failure analysis, I&C/digital upgrades and plant operations support (Operations, Maintenance and Engineering).

Senior Consultant – Polestar 2003 - 2005

Senior consultant with Polestar Inc. providing consulting services focused on equipment reliability and asset management services in support of the commercial nuclear utility industry. Assignments included due diligence teams, independent reviews/assessments, representing minority owners of nuclear power plants, team leader for an EPRI sponsored five station Equipment Reliability/Long Term Planning Benchmarking Project as well as providing technical and programmatic consulting services to utilities in support of assessing and improving system engineering, maintenance effectiveness and equipment reliability strategies.

Plant Engineering Manager - Seabrook Station 1998-2003

Assigned as the Plant Engineering Manager responsible for 75 System Engineers, component engineers and maintenance engineers at Seabrook Station. In this position, he was responsible for developing and implementing of a station equipment reliability recovery plan based on the guidelines in INPO AP-913. This assignment consisted of developing a Station Plant Health Committee, a station equipment reliability vision and associated roles, responsibilities and expectations for the system, component engineers and maintenance engineers. Key elements of this program included the implementation of a PMO Project focused on establishing PM technical requirements utilizing streamlined RCM methodologies/ CBM practices. In addition, comprehensive programmatic processes focused on improving plant performance and reducing O&M costs were implemented such as System Engineering (system health reports/performance monitoring), maintenance planning and scheduling (AP-928). Comprehensive change management plans were developed to assist in the implementation of the technical elements along with the organizational and behavioral elements of this program. Results of this effort contributed to repeat INPO 1 evaluations (since 1999) and significant improvements in equipment reliability and overall station performance.

Condition Based Maintenance Manager 1994-1998

Served as the Condition Based Maintenance Manager for Northeast Utilities consisting of 80 engineers, technicians and contractors. In this position, he was responsible for the development and implementation of a corporate equipment reliability strategy covering Millstone Station, Connecticut Yankee and Seabrook Station. This assignment consisted of developing a comprehensive maintenance program based on preventive, predictive and condition monitoring processes and methodologies for three nuclear sites and a number of fossil and hydro units. A significant responsibility for this position consisted of developing and implementing a change management strategy that would support the successful technical implementation and behavior changes required to establish a safe, reliable and cost effective maintenance strategy. During this period Mr. Nichols participated in the development of AP-913"Equipment Reliability Process" as well as working on the INPO Equipment Reliability Working Group (ERWG)

System Engineering Supervisor – Connecticut Yankee Nuclear 1984-1993

Mr. Nichols served in leadership positions as System Engineering Supervisor, Maintenance Engineering Supervisor and Project Manager at Connecticut Yankee (Haddam Neck Plant) Atomic Power Company. In these roles he developed engineering and maintenance capabilities focused on insuring equipment reliability supported safe and economical power operations. In addition, Mr. Nichols performed in leadership roles in support of Refuel Outages, extended plant shutdown and recovery efforts, maintenance improvement programs, regulatory and INPO interface.

EDUCATION

- BS – Marine Engineering, Maine Maritime Academy
- "Senior Nuclear Plant Managers Program" - Institute of Nuclear Power Operations 2002
- Executive Education – General Management Development Program
  Harvard Business School 2002

LOU PYRIH

NUCLEAR PLANT ENGINEERING VICE PRESIDENT & DIRECTOR

RESUME

SUMMARY

Over forty years of experience in engineering, licensing, engineering management, and project management of nuclear, fossil, and hydroelectric generating stations. Design reviews of nuclear plant systems and spent nuclear fuel storage facilities. Preparation of nuclear plant Safety Analysis Reports and Environmental Reports for nuclear plant license applications to the Nuclear Regulatory Commission (NRC), and testimony in hearings before the Atomic Safety and Licensing Board (ASLB). Development of management processes for engineering, design of modifications, performance of 10CFR50.59 reviews, and configuration management.

Performed as vice chairman of Nuclear Review Board with oversight responsibility for safety of all nuclear operations. Provided consulting services in litigation, including expert testimony. Performed site selection studies for nuclear generating stations. Performed reservoir, open channel, and conduit hydraulic analyses for pumped storage hydroelectric plant and water supplies for nuclear and fossil-fueled generating stations.

Senior Technical Consultant (2001 to present)

- Bruce Power, Units 3 &4. Performed a due diligence restart readiness assessment for client.
- Exelon, Peach Bottom Atomic Power Station, Units 2 and 3, Dresden, Units 2 and 3, and Quad Cities, Units 1 and 2. Participated in preparation of the License Renewal Applications.
- Kozloduy Nuclear Power Plant, Kozloduy, Bulgaria. With local partner, Risk Engineering, provided consulting services on all aspects of Configuration Management to Bulgarian nuclear utility.

Senior Management Consultant (1993 to 2000)

- Alliant Energy, Duane Arnold Energy Center. Participated in assessment of feedwater heater shell wall degradation. The assessment included a review of industry experience related to feedwater heater shell wall thinning, Duane Arnold's design and operating parameters and recommended an inspection program, including heater prioritization and timing for Duane Arnold.
- Consumers Energy Company (CEC), Fossil and Hydro Operations. Provided management consulting services. Led the consultant team that conducted interviews with CEC management and staff, developed processes and procedures in project management, engineering management, construction management, verification and testing, and outage management.

- PECO Energy. Conducted reviews on the TN-68 spent Fuel Dry Cask Storage Draft Topical Safety Analysis Report, specifically the Thermal Evaluation chapter that assesses the technical adequacy of information, supporting analysis and calculations, and conformance of the design to regulatory requirements and design criteria.

- Consumers Energy Company, Nuclear Operations. Developed a Project Management Manual in preparation for the decommissioning of Big Rock Point Nuclear Plant.

- Carolina Power and Light Company. Provided consulting services in the development of implementation plans for improvements in its Configuration Management Program.

- Carolina Power and Light Company, Brunswick Nuclear Plant. Performed an independent assessment on the effectiveness of engineering improvement initiatives.

- Commonwealth Edison, Braidwood Nuclear Plant, Cooling Pond. Provided consulting services for their litigation with local authorities regarding taxation. Provided expert testimony in court case for Commonwealth Edison.

- PECO Energy. Provided consulting services on oversight monitoring of Configuration Management Program performance.

- For several nuclear utilities, provided independent reviews of their responses to Nuclear Regulatory Commission's (NRC) 10CFR50.54(f) information request on design basis maintenance and configuration management.
- Northeast Nuclear Energy Company, Millstone Nuclear Power Plant; Connecticut Yankee, Haddam Neck; and Public Service of New Hampshire, Seabrook plants. Team leader for the team that developed a configuration assurance process for the oversight organization
- Public Service Electric & Gas Company. Developed an engineering assurance program for Nuclear Design Engineering.

Director of Nuclear Engineering Division, Philadelphia Electric Company (1961 to 1991)

- Managed a division of 260 engineers, designers, and support staff. Responsible for performing design basis reconstitution, multi-discipline studies, safety analyses and design, construction support, and start-up testing of modifications at four 1000 MW nuclear units. Major accomplishments included the formulation of procedures and processes for a newly organized division and the provision of leadership to accept a new corporate vision and teamwork.

- Managed a division of 200 engineers and support staff. This division was responsible for performing multi-discipline studies, safety analyses, safety assessments, engineering, construction support, and start-up testing on all modifications at three operating 1000 MW nuclear units. Also performed independent design reviews, start-up test, and readiness reviews for one 1000 MW nuclear unit under construction. Major accomplishments included providing engineering support for two nuclear units that were restarting from NRC ordered shutdown.

- Chief Mechanical Engineer, managed a division of 145 engineers and clerical staff. This division was responsible for all mechanical, civil, chemical, nuclear, and architectural engineering aspects of the company's fossil, nuclear, hydroelectric generating stations and transmission and distribution substations.
- Engineer in Charge of Power Plant Services Section. Managed a section of 26 engineers. This section was responsible for auxiliary plant systems, chemical water treatment, and waste management; engineering for nuclear and fossil generating stations, including engineering and project management of the installation of a S02 scrubber at a two unit supercritical steam fossil-fired generating station.

- Nuclear and Environmental Section. Managed a section of 20 engineers. This section was responsible for nuclear plant licensing, including preparation of Safety Analyses Reports and Environmental Reports to the NRC, the application of reactor core analysis codes, excore nuclear fuel management, design and implementation of environmental monitoring programs for nuclear, fossil, and hydro-electric plants and performance of Probabilistic Risk Assessments (PRAs).

- Supervising Engineer, managed two different branches in the Mechanical Engineering Division with responsibilities for nuclear steam supply systems, nuclear plant licensing, and fuel management.
- Mechanical Engineering Division. Progressed through positions with responsibilities for pump turbines in hydroelectric installation and project engineering for start-up of pumped storage plant; site selection studies for nuclear plants; water supplies and circulating water systems at various generating stations; studies of water storage reservoirs in the Delaware River Basin; nuclear steam supply systems at two nuclear plants; and project engineering for modifications at a 2000 MW nuclear plant

EDUCATION

B.S., Mechanical Engineering, Drexel University, 1960

M.S., Mechanical Engineering, Drexel University, 1965

M.B.A., LaSalle University, 1983

Additional Courses:

- Executive Development Program, Cornell University, 1985

JAY ROSEN

SENIOR CONSULTANT

RESUME

SUMMARY

Mr. Rosen has more than 20 years of experience in Plant Maintenance, Condition Based Maintenance, Corrective and Proactive Maintenance, Reliability Centered Maintenance, Diagnostic Tools and Technologies, Instrumentation and Controls, Engineering and Project Management, and Facilities Engineering. Mr. Rosen's accomplishments include:

- Development of an Integrated PM/RCM Program and System to Address more than 12000 distinct maintenance activities as part of a Comprehensive RCM Program
- Installation of a wireless LAN and sensor system for remote Condition Monitoring, System and Equipment Health Assessment.
- Development and Implementation of an Environmental Qualification Program of a Multi-Unit Nuclear Plant
- Development and Integration of an Electronic Documentation Management System to Manage more than 2500 Diagnostic and Maintenance Procedures

DETAILS

As a Project Manager for CSI Services, Mr. Rosen managed a Maintenance Optimization project for PECO Energy that realized $300,000 per year in maintenance cost reductions. He was also responsible for development and implementation of an Optimized Maintenance program at Southern Natural Gas Company and developed a maintenance plan for Princeton University's Cogeneration facility.

At Philadelphia Electric Company, Mr. Rosen was the Environmental Qualification and Preventive Maintenance Coordinator and was responsible for ensuring Maintenance/I&C department compliance with EQ and PM station requirements. His duties included originating Non-Conformance Reports and Engineering Work Requests, maintaining the PM and EQ sections of the CHAMPS and PIMS databases and implementing the Predictive Maintenance and Reliability Centered Maintenance (PM/RCM) programs at the Limerick Generating station.

At Public Service Electric and Gas Company, Mr. Rosen was I&C staff engineer responsible for developing and implementing the department's preventive maintenance program and database development for instrument valve lineups. He supervised approximately 40 subcontracted procedure writers and spare parts evaluators. He also managed department computer tracking systems.

Mr. Rosen has been associated with EPRI and EPRI Solutions since 1998. He has been involved in maintenance assessments (Nuclear, Fossil and T&D), establishing and implementing PdM programs, implementing RCM programs, and installation of wireless sensor systems.

Mr. Rosen is currently providing subcontracted services to the Electric Power Research Institute in the areas of Condition Based Maintenance. These services include project development, on-site project management, scheduling, project report development, and maintenance program assessments.

Mr. Rosen is also providing consulting and Engineering support activities to various Utilities and Engineering/Consulting firms in the areas of Condition Based maintenance, Leadership and Process Improvement.

EMPLOYMENT

2001 – Present	R&P Consulting, LLC	Director and Operations Manager
2001 – 1998	CSI Services	Sr. Consultant
1995 – 1998	Applied Management Engineering	Consultant
1989 – 1995	Peco Energy (Exelon)	System Engineer
1987 – 1989	Canus Corporation	Consultant
1984 – 1987	PSE&G	I&C Staff Engineer

EDUCATION

- BS Degree, Electrical Engineering – Minor Physics, Wilkes University, Wilkes-Barre, PA, USA

GARY SCHWEIZER

SENIOR MANAGER OF NUCLEAR PLANT MAINTENANCE & ENGINEERING

RESUME

SUMMARY

Over twenty five years of achievement as an engineer and manager. Recently provided engineering management consulting support for the Limerick Generating Station ISFSI Project. Previously held positions within Exelon in Maintenance, Engineering and Work Management, which included Site Supervisor PBAPS for Nuclear Maintenance Division, Manager of Component Engineering at Limerick Generating Station, Manager of Work Management Limerick Generating Station responsible for all on-line and refueling activities, and Senior Manager Corporate Maintenance/Work Control Exelon.

DETAILS

Limerick Station Independent Spent Fuel Storage Project (02/07 thru 06/08)

Provided Engineering & Project Management Consulting for the planning and installation of the Limerick Generating Station (LGS) Independent Spent Fuel Storage Installation (ISFSI) project. Also coordinated material rail deliveries of storage modules and installation of modules on ISFSI Pad. Involved in planning and execution of ISFSI first fuel campaign at LGS.

Exelon, Peach Bottom Station Generator Modification/Rewind Project (05/04 – 12/06)

Consulting Project Manager -Developed bid specification for Exelon fleet wide generator rewind project from design phase through contract award. Subsequently provided assistance to Peach Bottom Generating Station to assist in planning and implementation of their first generator rewind.

First Energy Nuclear Operating Company (10/03 – 02/04)

Conducted assessment in preparation for INPO Corporate Evaluation. .Evaluated Organization Effectiveness, Operational Focus, Performance Improvement, Radiation Protection, Work/Outage Management and Chemistry.

PECO Energy Nuclear (12/00 -10/03)

Senior Manager Maintenance & Work Control (12/00 -10/03)

Responsible for providing governance and oversight of Maintenance & Work Control (M/WC) activities at 4 Nuclear Sites. Including standardizing Maintenance and Work Management processes and practices. Responsible for Corporate M/WC staff; providing direction, coaching, etc. of personnel. Developed and maintained business plan and budget for the organization., and managed/performed numerous Self assessments at all 4 nuclear sites and corporate support office.

Manager of LGS Outage & On-line Work Management (11/96 -12/00)

LGS Outage & Online Maintenance Manager- Supervised personnel involved in LGS Work Management processes. Managed processes for performing all on-line maintenance work and preparing for and executing Refuel Outages at LGS, including risk management.

Manager of Component Engineering – Limerick Station (05/93 -11/96)

Manager of Component Engineering - Started up Component Engineering and new Site Engineering branch. Supervised Component Engineering personnel who were responsible for improving component reliability, Predictive Maintenance Program and numerous Engineering programs (e.g., ISI and IST).

Nuclear Maintenance Division Branch Head & PBAPS Site Supervisor (05/89 – 05/93)

Supervisors responsible for performing Electrical and Mechanical Maintenance at fossil and nuclear stations.

Point of contact for all NMD activities on site at PBAPS. Supervised Technical Staff and Craft daily during non-outage time. Supervised daily activities of NMD Planners.

Engineer Supervisor – Limerick Generating Station (06/88 -05/89)

Supervised a group of Engineers and Technical Assistants providing technical support for mechanical and electrical maintenance craft. Additionally assisted in preparation of Reliability Centered Maintenance Program

for Limerick Generating Station (LGS) and Peach Bottom Atomic Power Station (PBAPS).

Maintenance Engineer - PECO Generation (01/79 -06/88)

Provided engineering support functions for Maintenance mechanical and electrical work groups (at all Fossil, Hydro, and Nuclear Stations). Specific projects included:

- Preparing the vendor work scope
- Pricing requirements for a multi-million dollar Eddystone Boiler Outage
- Muddy Run Hydro Turbine Seal Replacement/Wicket Gate Repair Outage
- Peach Bottom Atomic Power Station Outage support.
- Wrote and reviewed controlled work procedures (Preventive Maintenance and Corrective Maintenance)
- Assisted in resolution of Quality Control concerns.

EDUCATION

B.S.E.E., Drexel University 1979

Training/Certifications

- Designated alternate for Non-Reactor Core Alterations
- Kepner Tregoe Problem Solving and Decision Making
- Dynamics of Selection Interviewing, Put It in Writing,
- Interaction Management, MARC, Conflict Management,
- Assessment of First Line Supervisors, Leadership
- Exchange Series for Second Level Supervisors,
- Management Accounting Game, and Component Engineer Certification

THOMAS E. SHANNON, P.E.

FORMER ENGINEERING DIRECTOR/VICE PRESIDENT

RESUME

SUMMARY

Over forty years of experience, including twelve years with WorleyParsons, in nuclear, fossil, hydro, and transmission and distribution for utility projects. Management and participation in numerous projects involving compliance with Federal Regulations, and ASME and ANSI Codes and Standards for nuclear power plants. Design reviews of nuclear plant systems, including consideration of compliance with 10CFR50, Appendix A, General Design Criteria, and applicable regulatory guides. Management of Power Uprate and License Renewal projects. Management of engineering organizations performing multi-discipline engineering studies, safety analysis, and design of power plant modifications for operating plants and design reviews of Architect/Engineering designs for nuclear plants under construction. Preparation of component specifications, safety analysis reports, procurement documents, management assessment task plans, and preparation, review, approval, and implementation verification and validation of corrective action requests.

DETAILS

Consultant, WorleyParsons - 2008 to present

Helped develop a feasibility study for new nuclear units in Russia and other Eastern European countries. Participated in the evaluation of various reactor alternatives for the Russian sites.

Manager Business Development, WorleyParsons - 2003 - 2007
Responsible for the development and implementation of the strategic business development plan for the nuclear services sector. Responsible for developing new nuclear services clients in the US and Canada. Responsible for the development of joint ventures to meet the needs of an emerging nuclear market.

Project Manager, WorleyParsons. – 2002-2003
Responsible for the development of Configuration Control process for a four unit nuclear site in Bulgaria. Interfaced with the client to ensure complete implementation of the Configuration Control Process. Evaluated their compliance with the process.

Project Manager, Parsons E&C – 2000-2002
PECO Energy, Peach Bottom License Renewal Project. Responsible for Parsons' execution of the License Renewal Project. Directed the Parsons' team preparing Aging Management Reviews and Time-Limited Aging Analysis. Directed the preparation of the license renewal application for submittal to the Nuclear Regulatory Commission (NRC). Interfaced on a daily basis with the customer to ensure customer expectations were met.

Nuclear Services Project Manager, Parsons E&C – 1997-2000

Duties included nuclear services business development for the Vice President of Nuclear Services. Primary emphasis is on maintaining relationships with existing clients and developing relationships with new clients.

Business Utilities Manager, Asta Engineering. – 1995-1997

Primary responsibilities included nuclear services business development. In addition, performed an independent assessment of the effectiveness of engineering improvement initiatives to Carolina Power and Light Company, Brunswick Nuclear Plant. Provided independent reviews for several nuclear utilities for their responses to NRC's 10CFR50.54(f) information request on design basis maintenance and configuration management. Member of the Asta team which developed a configuration assurance process for the oversight organization of Northeast Nuclear Energy Company's Millstone Nuclear Power Station, Connecticut Yankee, and Seabrook plants.

Senior /Engineer, PECO. - 1994

Civil and Mechanical Engineering. Managed a multi-discipline organization responsible for the civil and mechanical engineering design work in the support of the operation of four 1,000 MW nuclear units, as well as the development and maintenance of various long range programs and studies in support of these units.

Instituted PECO Project Work management and Program Review Procedures. Successfully turned around negative customer feedback to positive with work backlog reduced from 300+ items to 30 items during first nine months of implementation..

Senior Project Manager, PECO – 1993-1994

Thermal Lag Project. Responsible for the development of the project plan to address NRC concerns with thermal lag fire barriers at two nuclear stations. Overall responsibility for scope, schedule, budgeting, and staffing.

Senior Project Manager, PECO. – 1991-1994

Nuclear Power Uprate Project. Responsible for a multi-discipline, multi-company project to increase the rating of four 1,000 MW nuclear units by 5 percent. The project encompassed all facets of plant design and operation as well as close coordination with the original equipment suppliers and regulatory agencies.

Performed detailed financial analysis and feasibility studies.

Developed detailed project plans, addressing resources, funding work breakdown structures and integrated project schedules for two year $62 million project. Project completed on schedule and under budget.

Presented results of analysis to corporate management and regulators for approval resulting in NRC licensing of four units to operate at the 05 percent power level with the units currently operating at that level.

Manager, PECO. – 1991

Special Projects, Production Department. Responsible to the Senior Vice-President for corporate compliance with the recently enacted Clean Air Act as well as for all dealings with the Susquehanna and Delaware River Basin Commissions.

Provided corporate representation on Emission Control Task Force for two 850 MW coal-fired, jointly-owned units. The primary responsibility of the task force was to develop the Clean Air Act Compliance Strategy for those units. Strategy included the technical, economic, and decision analysis as well as the specification development and bid evaluation for the strategy ultimately selected.

Also responsible for the development of a corporate project management program for implementation throughtout the company. Worked extensively with the Project Management Institute and various utility and non-utility companies to develop a foundation for that program.

Assistant Manager, PECO. – 1989-1991

Engineering Division. Responsible for the engineering support for the company's fossil, hydro, and combustion turbine units as well as the entire transmission and distribution system. During this period, the division was completely restructured to provide more focused and customer-oriented support.

Participated in a major reorganization of the engineering division, restructuring to Fossil and Transmission and Distribution Division.

Instituted goals and objectives for the new division, with appropriate performance indicators for 160 assigned engineering and engineering support staff.

Initiated quality management (QM) teams to improve efficiency, division became corporate model for teams in transition. QM teams instituted many cost savings measures including standardized substation designs resulting in no less than $1.0 million annually with a cumulative savings in excess of $5 million.

Project Manager, PECO. – 1987-1989

Peach Bottom, Nuclear Engineering and Construction Department. Responsible for all of the nuclear engineering and construction department's activities relative to the restart of two 1,000 MW nuclear un its from an NRC-ordered shutdown. This function included budgeting, planning, organizing, controlling, monitoring and reporting for all of the engineering and construction work necessary to restart these units. Organized onside team of 60+ engineers to analyze and correct deficiencies.

Generated project plans included task plans, integrated schedules, and root cause analysis to ensure deficiencies properly identified, corrected, and precluded from recurring.

Presented results of engineering analysis to corporate management and NRC; received NRC approval for restart of the units received.

Engineer-In-Charge, PECO. – 1985-1987

Power Plant Services Section, Engineering Division. Managed a section responsible for the design of modifications to SO2 scrubber projects, chemical and auxiliary plant systems for all nuclear and fossil/hydro generating stations, as well as radwaste engineering for four 1,000 MW nuclear units. The section consisted of approximately 40 engineers and support personnel. Also during this time, participated in the reorganization of the engineering division into the nuclear engineering division.

Supervising Engineer, PECO. – 1983-1985

Managed the nuclear steam supply branch of the Mechanical Engineering Division responsible for the modifications of the NSSS for two 1,000 MW operating nuclear units and for the design for the NSSS for two 1,000 MW nuclear units under construction.

Engineer, Senior Engineer/Project Engineer, PECO. – 1971-1983

Progressed through position in mechanical engineering division with responsibilities for major redesign of pump turbines in a hydroelectric installation site selection studies for nuclear plants, deisgn of circulating water systems for various generating stations, project engineering for modifications at 12,000 MW nuclear plant as well as the design of the nuclear steam supply system and various balance of plant systems of a 2,000 MW nuclear plant under construction.

United States Army. – 1968-1971

EMPLOYMENT

2003 – Present	WorleyParsons	Consultant
2003-2007	WorleyParsons	Manager Business Development
2002-2003	WorleyParsons	Project Manager
2000-2002	Parsons E&C	Project Manager
1997-2000	Parsons E&C	Nuclear Services Project Manager
1995-1997	Asta Engineering	Business Utilities Manager
1994	PECO	Senior Manager
1991-1994	PECO	Senior Project Manager
1991	PECO	Manager

1989-1991	PECO	Assistant Manager
1987-1989	PECO	Project Manager
1985-1987	PECO	Engineer-In-Charge
1983-1985	PECO	Supervising Engineer
1971-1983	PECO	Engineer. Senior Engineer/Project Engineer

EDUCATION

Bachelor or Science - Mechanical Engineering - 1968 Drexel University, Philadelphia, PA

M.B.A., Drexel University, 1975

REGISTRATIONS/AFFILIATIONS

Professional Engineer, Pennsylvania

RICHARD N. SWANSON

CONSULTANT

RESUME

SUMMARY

Mr. Swanson has thirty-six years of experience with organizations in capital-intensive technical industries. He has made key contributions in a number of dramatic performance turn-arounds, both as a consultant and senior line manager. Budgetary responsibilities have included annual budgets of $245 million (capital) and $74 million (expense), with organizations of up to 550 employees. Line management responsibilities have included engineering, regulatory strategy & compliance, project management, construction, performance oversight, quality assurance, nuclear safety assessment, and risk assessment. Programmatic responsibilities have included managing technical programs, managing projects (from one to over one hundred projects), designing and performing readiness assessments, diagnosing organizational weaknesses, designing and conducting independent program assessments, designing and mentoring self-assessments, analyzing and improving processes, developing and implementing organization-wide improvement strategies, and leading and coaching event investigation teams.

Clients have included nuclear and fossil generating companies; uranium enrichment facilities; electric transmission and distribution companies; manufacturing organizations; DOE; and the NRC.

DETAILS

Root Cause, Event Investigations, Employee Concerns & Performance Assessment

◊ One of a small team of instructors training NRC inspectors how to evaluate licensee Root Cause Assessments (US NRC)

◊ Led, coached, and supported numerous independent investigations into employee allegations & concerns (including SCWE), “root cause,” events, management performance (US DOE (Yucca Mountain Project); Shaw AREVA MOX Services, LLC; Proto-Power Corp; InfraSource, Inc;

Portsmouth GDF; Paducah GDF; Beaver Valley, Perry, Davis-Besse; Millstone Station; PG&ENEG

Operations Division; Baltimore Gas & Electric Co. Transmission Dep’t.; Susquehanna; Indian

Point 2; BNFL Fuel Solutions Corporation; D C Cook; LaSalle; Salem; Hope Creek; Palisades; Big

Rock Point; Pilgrim) ◊ Mentored plant evaluators re: Event Investigations and Root Cause Assessments (Millstone Station, D C Cook; LaSalle)

◊ Trained and Mentored managers and departments responsible for operational performance assessments (NMC, Portsmouth Gaseous Diffusion Plant; Millstone Nuclear Oversight;

Palisades; Big Rock Point; Salem; Hope Creek)

◊ Established station-wide, on-going quarterly operational performance assessment program (Palisades, Big Rock Point)

◊ Evaluated and enhanced operational performance assessment programs and practices (Point Beach, Susquehanna, GE Nuclear Energy Division; Portsmouth Gaseous Diffusion Plant; Millstone Station; St. Lucie; Palisades; Big Rock Point; Salem; Hope Creek; Turkey Point)

◊ Evaluated, enhanced, and implemented operational assessment methods, integration, and planning as General Manager/Director of QA (Palisades; Big Rock Point; Salem; Hope Creek)

◊ Evaluated various aspects of "readiness to restart" (DC Cook, Crystal River, Millstone Station, LaSalle, Palisades, Big Rock Point, Pilgrim)

◊ Analyzed Performance Indicators used by management, provided recommendations for improvement (Susquehanna; D C Cook; LaSalle; St. Lucie; Millstone; Portsmouth Gaseous Diffusion Plant; Salem; Hope Creek; Palisades; Big Rock Point)

◊ Authored, implemented "Principles of Nuclear Oversight" (Crystal River; Palisades; Big Rock Point; Salem; Hope Creek)

◊ Advised self-assessment teams (Indian Point 2, St. Lucie, Palisades, Big Rock Point)

• Process & Program Analysis

◊ Advice, process development &/or management of regulatory agency team inspections (D C Cook; Quad Cities; Millstone 2 & 3; Crystal River; Portsmouth Gaseous Diffusion Plant; Palisades; Big Rock Point; Salem; Hope Creek; Pilgrim).

◊ Process development & support for regulatory inspection management, information management, and implementation plan for Independent Corrective Action Verification Program (inspection involving more than 15 NRC inspectors and 45 inspecting engineers for 25+ weeks per unit) (Millstone 2 & 3)

◊ Defined, planned, implemented self-assessment programs, achieved measurable line performance improvement (Palisades; Big Rock Point; Salem; Hope Creek)

◊ Process analysis (Work Control; Design Processes; Corrective Action Programs; others)

(Portsmouth Gaseous Diffusion Plant; Salem; Hope Creek; D C Cook; LaSalle; Millstone Station;

St. Lucie; Crystal River; Indian Point 2; Pilgrim)

◊ Evaluated oversight processes & interfaces (Susquehanna, GE Nuclear Energy Division;

Portsmouth Gaseous Diffusion Plant; Millstone Station; St. Lucie; Palisades; Big Rock Point;

Salem; Hope Creek; Turkey Point)

◊ Extensive consulting assignments re: restart strategy & processes, licensing department performance improvement, licensing department interfaces (Cooper, D C Cook; Crystal River;

Millstone Units 3 & 2; LaSalle; Pilgrim)

◊ Authored, implemented "Principles of Regulatory Interface (Cooper, D C Cook, Crystal River)

• Technical Support

◊ Managed Engineering Department before, during and after plant was 'watch listed'; received four consecutive "SALP 1" NRC evaluations in engineering and tech support (Pilgrim)

◊ Detailed review and advice re: major design information submittals required by 10CFR50.54(f)

(Millstone 2; Millstone 3; Braidwood; Byron; Dresden; LaSalle; Quad Cities; Zion).

◊ Developed & implemented numerous engineering programs (e.g., Erosion/corrosion; Equipment Qualification; Appendix R/Fire Protection; Heavy Loads; Long-Term Equipment Layup; Safety Enhancement Program; Risk Assessment; Design Review Board) (Pilgrim; Salem; Hope Creek)

◊ Project Manager/Project Engineer for various technical, critical path projects (e.g., Drywell

Restoration following elevated temperatures; Feedwater Heater Replacement)

◊ Enhanced QA effectiveness, overhauled oversight assessment methods, integration, and

planning as General Manager/Director of QA (Palisades; Big Rock Point; Salem; Hope

Creek)

◊ Turned around QA Department performance while reducing complement 17% (Consumers

Power)

◊ Strategic advice regarding QA Department deployment, implementation, and focus (US DOE

(Yucca Mountain Project))

◊ Construction Manager for second US Boiling Water Reactor Recirc Pipe replacement (Pilgrim)

◊ Established and implemented capital budget management strategy (PSE&G; Boston Edison)

◊ Achieved measurable improvements in project accountability, tightened project controls for project organizations with annual budgets in excess of $150 Million (Hope Creek; Salem; Pilgrim)

◊ Established regulatory affairs department, defined and implemented regulatory strategy and processes, repaired deteriorated relations with NRC (Pilgrim)

◊ Past member of Power Ascension Executive Review Board (Pilgrim)

◊ Past member of Management Safety Review Committee (Palisades; Big Rock Point)

EDUCATION

Babson College, Wellesley, Massachusetts; MBA, Finance.

Northeastern University, Boston, Massachusetts; MS, Engineering Management & Operations Research.

U.S. Naval Nuclear Power Program.

U.S. Naval Academy, Annapolis, Maryland; BS, Operations Analysis.

Licensed Professional Engineer (Mechanical).

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