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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 229 TO FACILITY OPERATING LICENSE NO. DPR-28

ENTERGY NUCLEAR VERMONT YANKEE, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

Proprietary information pursuant to
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1.0 INTRODUCTION

1.1 Application

By application dated September 10, 2003, as supplemented by letters dated October 1, and October 28 (2 letters), 2003; January 31 (2 letters), March 4, May 19, July 2, July 27, July 30, August 12, August 25, September 14, September 15, September 23, September 30 (2 letters), October 5, October 7 (2 letters), December 8, and December 9, 2004; February 24, March 10, March 24, March 31, April 5, April 22, June 2, August 1, August 4, September 10, September 14, September 18, September 28, October 17, October 21 (2 letters), October 26, October 29, November 2, November 22, and December 2, 2005; January 10, and February 22, 2006 (References 1 through 46), Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy or the licensee), requested changes to the Facility Operating License and Technical Specifications (TSs) for the Vermont Yankee Nuclear Power Station (VYNPS).

The proposed changes would increase the maximum steady-state reactor core power level from 1593 megawatts thermal (MWt) to 1912 MWt, which is an increase of approximately 20%. The proposed increase in power level is considered an extended power uprate (EPU).

1.2 Background

VYNPS is a boiling-water reactor (BWR) plant of the BWR/4 design with a Mark-I containment. The Nuclear Regulatory Commission (NRC or Commission) licensed VYNPS on February 28, 1973, for full-power operation at 1593 MWt (i.e., the current power level).

The VYNPS site is located in the town of Vernon, Vermont, on the west bank of the Connecticut River, on the pond formed by the Vernon Dam and Hydroelectric Station. As shown in VYNPS Updated Final Safety Analysis Report (UFSAR) Table 2.2.1 (Reference 50), in the year 2000, the population was estimated to be 9,919 within a 5-mile radius of the site, 23,954 within a 10-mile radius, and 193,746 within a 25-mile radius.

The construction permit for VYNPS was issued by the Atomic Energy Commission (AEC) on December 11, 1967. The plant was designed and constructed based on the proposed General Design Criteria (GDC) published by the AEC in the *Federal Register* (32 FR 10213) on July 11, 1967 (hereinafter referred to as "draft GDC"). The AEC published the final rule that added Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "General Design Criteria for Nuclear Power Plants," in the *Federal Register* (36 FR 3255) on February 20, 1971 (hereinafter referred to as "final GDC").

Differences between the draft GDC and final GDC included a consolidation from 70 to 64 criteria. As discussed in the NRC Staff Requirements Memorandum for SECY-92-223, dated September 18, 1992 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML003763736), the Commission decided not to apply the final GDC to plants with construction permits issued prior to May 21, 1971. At the time of promulgation of Appendix A to 10 CFR Part 50, the Commission stressed that the final GDC were not new requirements and were promulgated to more clearly articulate the licensing requirements and practice in effect at that time. Each plant licensed before the final GDC were formally adopted was evaluated on a plant-specific basis, determined to be safe, and licensed by the Commission.

As discussed in Appendix F of the VYNPS UFSAR, the licensees for VYNPS have made changes to the facility over the life of the plant that may have invoked the final GDC. The extent to which the final GDC have been invoked can be found in specific sections of the UFSAR and in other VYNPS design and licensing basis documentation.

Unique design features at VYNPS include the alternate cooling system (ACS) and the alternate alternating current (AAC) power source. These features are discussed below.

During the original plant licensing, the hypothetical loss of the Vernon Dam was postulated. This led to the design and implementation of the ACS, which is a closed-cycle cooling system. VYNPS has two cooling towers, each with eleven cells. The design of the ACS utilizes one cooling tower cell and the associated fan, a cooling tower water basin (deep basin), and the residual heat removal service water (RHRSW) pumps and piping. The deep basin is sized to provide a 7-day supply of water. The ACS is not classified as an engineered safeguards system and is not designed to accept the consequences of a design basis loss-of-coolant accident (LOCA). It is also not designed to meet single failure criteria. The ACS is used for those events where the service water (SW) pumps are not available, which could occur during flooding of the intake structure, if the Vernon Dam fails and the river level falls, or due to a fire in the intake structure which disables the SW pumps. The ACS is designed to provide adequate heat removal for these postulated events to achieve and maintain safe shutdown when the normal SW system (i.e., pumping from the Connecticut River) is lost. The evaluation of the ACS under EPU conditions is discussed in Safety Evaluation (SE) Sections 2.5.3.3 and 2.5.3.4.

VYNPS has an AAC power source for coping with a Station Blackout (SBO) event in order to meet the requirements in 10 CFR 50.63. Under this approach, VYNPS relies on the Vernon Hydroelectric Station to provide power to an emergency bus until offsite or onsite alternating current (AC) power is available. The evaluation of the impact of the proposed EPU on the plant's ability to cope with and recover from an SBO event is discussed in SE Section 2.3.5.

1.3 Licensee's Approach

The licensee's application for the proposed EPU was prepared following the guidelines contained in General Electric (GE) Licensing Topical Report (LTR) NEDC-33004P-A, "Constant Pressure Power Uprate," Revision 4, dated July 31, 2003 (Reference 51). The constant pressure power uprate (CPPU) LTR (CLTR) was approved by the NRC in an SE dated March 31, 2003 (Reference 52).

Attachment 4 to Reference 1 contains GE Report NEDC-33090P (proprietary) which is the Power Uprate Safety Analysis Report (PUSAR) for VYNPS. This report summarizes the results of the safety analyses and evaluations performed by GE specifically to justify the proposed EPU for VYNPS. The report follows the generic content and format using the CPPU approach to uprating reactor power, as described in the CLTR. A non-proprietary (i.e., publicly available) version of the PUSAR is contained in Attachment 6 to Reference 1.

As described in Section 1.2 of the PUSAR, an increase in the electrical output of a BWR is accomplished primarily by generation and supply of higher steam flow to the turbine-generator. Most BWRs, as originally licensed, have as-designed equipment and system capability to accommodate steam flow rates at least 5% above the original rating. In addition, continuing

improvements in the analytical techniques have resulted in a significant increase in the design and operating margin between the calculated safety analyses results and the current plant licensing limits. The available margins in the calculated results, combined with the as-designed excess equipment, system, and component capabilities: (1) have allowed many BWRs to increase their thermal power ratings by 5% without any nuclear steam supply system (NSSS) hardware modifications, and (2) provide for power increases up to 20% with some non-safety hardware modifications.

For VYNPS, the method for achieving higher steam flow necessary for the proposed 20% EPU would be accomplished by retaining the existing maximum extended load line limit analysis (MELLLA) power/flow map and increasing core flow (and power) along the MELLLA upper boundary line as shown in Figure 1-1 in the PUSAR (Reference 1, Attachment 4, page 1-12). The current MELLLA power/flow map was approved in VYNPS Amendment No. 219 dated April 14, 2004 (Reference 53). As discussed in Section 2.1 of the PUSAR, the additional energy requirements for CPPU are met by an increase in bundle enrichment, an increase in the reload fuel batch size, and/or changes in fuel loading pattern to maintain the desired plant operating cycle length.

The proposed CPPU approach would not increase the reactor operating pressure or the current licensed maximum core flow. CPPU operation would not involve increasing the maximum reactor vessel dome pressure because the plant, due to modifications to non-safety power generation equipment, would have sufficient pressure control and turbine flow capabilities to control the inlet pressure conditions at the turbine. Table 1-2 of the PUSAR provides a summary of the reactor thermal-hydraulic parameters for current licensed thermal power (CLTP) plant operating conditions and CPPU conditions (Reference 1, Attachment 4, page 1-11).

The licensee plans to implement the EPU in one step (i.e., the proposed 20% increase will occur in a single power ascension program). The licensee completed the modifications necessary to implement the EPU during the refueling outage in fall 2005. Subsequently, the plant will be operated at 1912 MWt starting in Cycle 25 (i.e., during the operating cycle following the fall 2005 outage).

1.4 Plant Modifications

The licensee has determined that plant modifications are necessary to implement the proposed EPU. A discussion of the EPU modifications is provided on pages 17 through 31 of Attachment 1 to Reference 24. The following is a list of these modifications:

Main Turbine Diaphragm Replacement

The 8th stage diaphragms of the low pressure turbines were replaced during the fall 2005 refueling outage to upgrade the turbine to accommodate the increased steam flow for EPU conditions. This change increases the structural integrity of the diaphragms.

Main Turbine Cross-Around Relief Valves and Discharge Piping

The main turbine cross-around relief valves and discharge piping have been modified to accommodate the increase in pressure and flow in the cross-around piping to the low pressure turbine.

Main Generator Stator Rewind

The main generator stator was rewound in place. In addition, the generator hydrogen coolers were replaced with upgraded coolers.

Main Condenser Tube Staking

Additional support staking of the main condenser tubes was performed to minimize potential effects of flow-induced vibration (FIV).

Feedwater Heater 4A/B Shell Side Relief Valve

Because the EPU will increase the feedwater mass flow through the heat exchanger tubes by 20%, the relief valves on the shell side must be capable of relieving pressure to avoid overpressure of the heat exchangers in case of an internal failure. The cause of the overpressure would be the failure of the heat exchanger tubes. The relief valves on feedwater heaters 4A/B have been replaced with higher capacity relief valves.

Steam Dryer Strengthening

Strengthening of the steam dryer was performed to reduce the effect from FIV.

Isolation Phase Bus Duct Cooling

A modification was made to the isophase bus duct cooling system to provide additional cooling capacity associated with the EPU.

High Pressure Feedwater Heater Replacement

The four high pressure feedwater heaters have been replaced to accommodate the EPU increased flow and pressure conditions and to provide more erosion-resistant material.

Residual Heat Removal Service Water (RHRSW) Piping Modification

A modification was made to the RHRSW pump motor bearing cooling water supply line to recover SW flow that is currently being discharged during ACS mode of operation of the pumps. Because the EPU will increase the decay heat rate and increase evaporative losses from the ACS deep basin, the return of the cooling water from the bearing oil coolers is necessary to maintain the 7-day deep basin water inventory design requirement.

Nuclear Steam Supply System (NSSS)/Torus Attached Piping Supports

Main steam line supports in the drywell and the reactor core isolation cooling (RCIC) line support in the RCIC room were upgraded based on EPU temperature considerations. These changes reestablished the design margins for the piping and support configurations.

Condensate Demineralizer Filtered Bypass

Operation at EPU conditions with increased condensate/feedwater flow will require operation of the five condensate demineralizer vessels. During backwash and precoat operations when one demineralizer is removed from service, the remaining four demineralizers do not have the capacity for full condensate flow, thus requiring a bypass flow path around the demineralizers and increasing the potential for debris to be passed from the condenser to the reactor. The new bypass filter provides the means of limiting debris passage by filtering the bypassed flow during demineralizer backwash and precoat operations.

Feedwater System Suction Pressure Trip and Reactor Recirculation (RR) System Runback

The proposed EPU requires that the three currently installed reactor feedwater pumps (RFPs) and the three currently installed condensate pumps (CPs) be operating to achieve the uprated power level (i.e., 1912 MWt). In the pre-EPU configuration for this operation, upon a trip of a CP, the suction pressure to the RFPs would drop such that the three RFPs would trip based on a single 150 psig low pressure suction trip. Therefore, this modification provides a staggered sequential time delay tripping of the RFPs such that suction pressure could recover to preclude tripping of all the RFPs. In addition, at EPU conditions with the trip of a CP or RFP, the steam/feedwater flow mismatch would result in a reactor trip on low level if power/steam flow were not rapidly reduced to levels that could be supported by the operating pumps. Therefore, a modification has been added which will provide a rapid RR pump runback on low feedwater flow following a RFP or CP trip during operation at high power levels. In addition, as discussed in Reference 42, a modification to trip the "B" RFP upon a trip of a CP has been added to provide additional margin to preclude inadequate RFP suction pressure and preserve feedwater flow.

Cooling Tower Fans/Motors

On 21 of the 22 cooling towers, the cooling tower fan blades and motors have been replaced with higher efficiency blades and higher horsepower motors to provide for cooling tower plume control (environmental and aesthetic issues).

Safety Relief Valve (SRV) Monitor Power Feed Relocation to New Breaker

Based on licensee evaluations conducted for the EPU, the breaker that fed the SRV monitor panel was found to be not environmentally qualified to the new environment. Accordingly, this modification rerouted the power feeding the panel to a new breaker that is located in a mild environment.

Grid Stability

The licensee's grid stability study identified several changes required for the grid to accept the uprated power. The modifications made were as follows:

- Increased the million volt-ampere (MVA) rating on the VYNPS - Northfield 345 kV line from 896 MVA to a minimum rating of 1075 MVA.
- Increased the MVA rating on the Ascutney-Coolidge 115 kV line from 205 MVA to 240 MVA.
- Added 60 MVA of shunt capacitors at the VYNPS 115 kV bus.
- Added a second primary protection scheme on the VYNPS north bus.
- Added a second primary protection scheme on the VYNPS main generator.
- Replaced the VYNPS 381 breaker to provide independent pole tripping.
- Added out-of-step protection for the VYNPS generator.

Main Turbine - High Pressure Flow Path

The modifications associated with the main turbine high pressure flow path include replacement of the rotor and diaphragms; new control cams, camshafts, and hydraulics; new control valve settings, and turbine control and setpoint changes.

Instrumentation and Control Changes

The changes in various plant parameters at EPU conditions (e.g., steam flows) will require various instrumentation and control setpoint and calibration changes including the following:

- Electronic pressure regulator (mechanical hydraulic pressure control system for the turbine generator) setpoint change;
- Main steam line high flow setpoint change;
- Neutron monitoring setpoint changes (average power range monitor flow-biased scram and rod block monitors);
- Rod worth minimizer setpoint; and
- Turbine first stage pressure setpoint.

The NRC staff's evaluation of the licensee's plant modifications, within the scope of the areas of review, is provided in Section 2.0 of this SE.

1.5 Method of NRC Staff Review

The NRC's staff's review of the VYNPS EPU application is based on NRC Review Standard RS-001, "Review Standard for Extended Power Uprates," (Reference 54). RS-001 contains guidance for evaluating each area of review in the application, including the specific GDC used as the NRC's acceptance criteria. Since the guidance in RS-001 is based on the final GDC and VYNPS was designed and constructed based on the draft GDC, Entergy submitted supplements to the EPU application dated October 1 and October 28, 2003 (References 2 and 4), which provided a matrix that cross-references the draft GDC to the final GDC and a matrix that cross-references the sections of the PUSAR that apply to each of the areas of review contained in RS-001. In addition, in a supplement dated January 31, 2004 (Reference 5), Entergy provided a revision to the template SE in RS-001 replacing the numeric values of the final GDC with the corresponding VYNPS design criteria and draft GDC that constitute VYNPS's current licensing basis. Related changes to VYNPS plant-specific design criteria were also incorporated in the revised template. Minor changes to the template were provided by Entergy in supplements dated July 2, 2004, and August 1, 2005 (References 9 and 31).

The NRC staff reviewed the licensee's application to ensure that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) activities proposed will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public. The purpose of the NRC staff's review is to evaluate the licensee's assessment of the impact of the proposed EPU on design-basis analyses. The NRC staff evaluated the licensee's application and supplements. The NRC staff also performed audits of analyses supporting the EPU and performed independent calculations, analyses, and evaluations as noted below.

In areas where the licensee and its contractors used NRC-approved or widely accepted methods in performing analyses related to the proposed EPU, the NRC staff reviewed relevant material to ensure that the licensee/contractor used the methods consistent with the limitations and restrictions placed on the methods. In addition, the NRC staff considered the effects of the changes in plant operating conditions on the use of these methods to ensure that the methods are appropriate for use at the proposed EPU conditions. Details of the NRC staff's review are provided in Section 2.0 of this SE.

Audits of the analyses supporting the proposed EPU were conducted by the NRC staff and its contractors in relation to the following topics:

- steam dryer structural integrity analyses (see SE Section 2.2.6)
- reactor neutronic and thermal/hydraulic analyses (see SE Section 2.8.7)

Independent confirmatory calculations, analyses, and evaluations were performed by the NRC staff and its contractors in relation to the following topics:

- reactor vessel pressure-temperature limits (see SE Section 2.1.2)
- LOCA mass and energy release (see SE Section 2.6.1)

- long-term containment temperature response for a LOCA (see SE Section 2.6.5)
- emergency core cooling system (ECCS) performance (see SE Section 2.8.5.6.2)
- lattice physics (see SE Section 2.8.7)
- alternative source term dose analyses (see SE Section 2.9.2)
- risk of crediting containment accident pressure (see SE Section 2.13)

1.6 Engineering Inspection

As discussed in an NRC inspection report dated December 2, 2004 (Reference 55), the NRC conducted a team inspection in accordance with Temporary Instruction (TI) 2515/158, "Functional Review of Low Margin/Risk Significant Components and Human Actions" (Reference 56), at VYNPS during the period from August 9 through September 3, 2004. The inspection was the first of four pilot inspections to be conducted at different plant sites to assist the NRC in determining whether changes should be made to the Reactor Oversight Process to improve the effectiveness of its inspections and oversight in the design/engineering area.

In selecting samples for review, the engineering inspection team focused on those components and operator actions that contribute the greatest risk to an accident that could involve damage to the reactor core. Additional consideration was given to those components and operator actions impacted by the proposed EPU license amendment. In addition, inspection samples were added based upon operational experience and issues previously identified by the NRC's technical staff during the course of its VYNPS EPU review. A complete listing of all components, operator actions, and operating experience issues reviewed by the inspection team is contained in Attachment A to Reference 55.

For each sample selected, the engineering inspection team reviewed design calculations, corrective action reports, maintenance and modification histories, and associated operating procedures, and performed walkdowns of material conditions (as practical). The team identified eight findings of very low safety significance (i.e., Green as defined in the NRC's Reactor Oversight Process), one unresolved item, and one minor finding. Based on the technical areas covered in RS-001, the NRC Headquarters staff determined that four of the inspection team findings would require Entergy to submit supplemental information to the NRC to support the EPU amendment request. The staff requested this information through requests for additional information (RAIs) and conference calls with Entergy. Entergy provided the additional information as described in the relevant portions of Section 2.0 of this SE. The four findings and how they relate to the EPU review are discussed below.

Station Blackout (SBO)

Finding: The team identified a non-cited violation of 10 CFR 50.63, "Loss of all alternating current power," because the licensee had not completed a coping analysis for the period of time the alternate AC source (the Vernon Hydro-Electric Station) would be unavailable, and had not demonstrated by test the time required to make the alternate source available for an SBO involving a grid collapse. This finding applies to current plant operation as well as EPU operating conditions.

EPU Review: EPU Review Standard RS-001, SE Section 2.3.5, "Station Blackout," requires that the NRC staff reach a conclusion that the licensee has adequately evaluated the effects of the proposed EPU on SBO and demonstrate that the plant will continue to meet the requirements of 10 CFR 50.63 following implementation of the proposed EPU. In order for the staff to reach this conclusion, Entergy needs to demonstrate that VYNPS meets the requirements in 10 CFR 50.63. The resolution of this issue is discussed in SE Section 2.3.5.

Appendix R Timeline for RCIC Initiation

Finding: The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," because from June 2001 to September 2004, the licensee did not adequately coordinate between the operations department and the engineering organization regarding procedure revisions that increased the length of time required to place the RCIC system in service from the alternate shutdown panels.

EPU Review: EPU Review Standard RS-001, SE Section 2.11, "Human Performance," requires the staff to conclude that the licensee has appropriately accounted for the effects of the proposed EPU on the available time for operator actions. The engineering inspection team found that the timeline for operator actions to place RCIC in service during an Appendix R scenario had been impacted due to procedure changes and that the licensee had not incorporated these changes into the VYNPS Safe Shutdown Capability Analysis (SCCA). However, the team found that at the current power level, during an Appendix R scenario, the operators have sufficient time to place RCIC in service from the alternate shutdown panels prior to reactor water level reaching the top of active fuel. At the proposed EPU power level, the team concluded that the margin was reduced such that the ability to place RCIC in service from the alternate shutdown panels prior to reactor water level reaching the top of active fuel was questionable. The resolution of this issue is discussed in SE Section 2.11.

Periodic Testing of Motor-Operated Valves (MOVs)

Finding: The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," because the licensee had conducted MOV diagnostic tests using procedures that did not include acceptance limits, which were correlated to and based on applicable (stem thrust and torque) design documents. Additionally, MOV diagnostic testing had been conducted solely from the motor control centers using test instrumentation that had not been validated to ensure its adequacy.

EPU Review: EPU Review Standard RS-001, SE Section 2.2.4, "Safety-Related Valves and Pumps," requires that the NRC staff reach a conclusion that the licensee has adequately evaluated the effects of the proposed EPU on its MOV programs related to Generic Letters (GLs) 89-10, 96-05, and 95-07, and the lessons-learned from those programs for other safety-related power-operated valves. The engineering inspection team found that the licensee did not manage NRC commitments and conditions documented in the SE for the GL 96-05 MOV

periodic verification program. The resolution of this issue is discussed in SE Section 2.2.4.

Condensate Storage Tank (CST) Temperature

Finding: The team identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," because the licensee had neither established the correct CST temperature limit for use in the plant transient analyses nor translated the CST temperature limit into plant procedures.

EPU Review: EPU Review Standard RS-001, SE Section 2.6.5, "Containment Heat Removal," requires the NRC staff to review the containment heat removal systems assessment provided by the licensee and conclude that the licensee has adequately addressed the effects of the proposed EPU. This review includes the effects of the proposed EPU on the analyses of the available net positive suction head (NPSH). The engineering inspection team found that the licensee used non-conservative CST temperatures in calculations for current plant conditions as well as for the EPU analyses. Although available NPSH margin was lowered, adequate NPSH for the core spray pumps remained due to conservatisms that existed in other aspects of the licensee's NPSH analysis. The resolution of this issue is discussed in SE Section 2.6.5.

2.0 EVALUATION

2.1 Materials and Chemical Engineering

2.1.1 Reactor Vessel Material Surveillance Program

Regulatory Evaluation

The reactor vessel material surveillance program provides a means for determining and monitoring the fracture toughness of the reactor vessel beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the reactor vessel. The NRC staff's review primarily focused on the effects of the proposed EPU on the licensee's reactor vessel surveillance capsule withdrawal schedule. The NRC's acceptance criteria are based on (1) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary (RCPB) be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage; (2) draft GDC-33, insofar as it requires that the RCPB be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant; (3) draft GDC-34 insofar as it requires that the RCPB be designed to minimize the probability of rapidly propagating type failures; (4) 10 CFR Part 50, Appendix H, which provides for monitoring changes in the fracture toughness properties of materials in the reactor vessel beltline region; and (5) 10 CFR 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix H. Specific review criteria are contained in Standard Review Plan (SRP) Section 5.3.1 and other guidance provided in Matrix 1 of RS-001 (Reference 54).

Technical Evaluation

The NRC's regulatory requirements related to the establishment and implementation of a facility's reactor vessel materials surveillance program and surveillance capsule withdrawal schedule are given in 10 CFR Part 50, Appendix H. Two specific alternatives are provided with regard to the design of a facility's reactor vessel surveillance program which may be used to address the requirements of Appendix H to 10 CFR Part 50.

The first alternative is the implementation of a plant-specific reactor vessel surveillance program consistent with the requirements of American Society for Testing and Materials (ASTM) Standard Practice E 185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." In the design of a plant-specific reactor vessel surveillance program, a licensee may use the edition of ASTM Standard Practice E 185, which was current on the issue date of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) to which the reactor vessel was purchased, or later editions through the 1982 edition.

The second alternative provided in Appendix H to 10 CFR Part 50 is the implementation of an integrated surveillance program (ISP). An ISP is defined in Appendix H to 10 CFR Part 50 as occurring when, "the representative materials chosen for surveillance for a reactor are irradiated in one or more other reactors that have similar design and operating features."

The licensee discussed the impact of EPU on the reactor vessel material surveillance program in Section 3.2.1 of Attachment 4 to Reference 1. This section indicates that VYNPS will participate in the BWR Vessel and Internals Project (BWRVIP) ISP and will comply with the withdrawal schedule specified for the surrogate surveillance capsules that now represent VYNPS.

The BWRVIP ISP was submitted for NRC staff review and approval in proprietary topical reports BWRVIP-78, "BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan," dated December 22, 1999, and BWRVIP-86, "BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan," dated December 22, 2000. Additional information necessary to establish the technical basis for, and proposed implementation of, the BWRVIP ISP was provided in letters from the BWRVIP to the NRC dated December 15, 2000, and May 30, 2001. The NRC staff approved the proposed BWRVIP ISP in an SE dated February 1, 2002. However, the NRC staff's SE required that plant-specific information be provided by BWR licensees who wish to implement the BWRVIP ISP for their facilities. The plant-specific information must demonstrate that each reactor has an adequate dosimetry program and that there is an adequate arrangement for sharing data between plants. In an amendment request dated March 26, 2003, the licensee addressed the VYNPS plant-specific information required in the NRC staff's February 1, 2002, BWRVIP ISP SE. The NRC staff approved the amendment request in a letter dated March 29, 2004 (VYNPS Amendment No. 218).

In the SE for VYNPS Amendment No. 218, the NRC staff evaluated the plant-specific information provided by the licensee to demonstrate the BWRVIP ISP can be implemented at VYNPS. The NRC staff concluded that the plant-specific information demonstrated that there is an adequate dosimetry program and an adequate arrangement for sharing data between plants. Since the licensee has provided the plant-specific information requested in the NRC

staff's SE for the proposed BWRVIP ISP, the licensee has demonstrated the compliance of VYNPS with the ISP requirements of Appendix H to 10 CFR Part 50.

As part of the proposed implementation of the BWRVIP, no further surveillance capsules will be removed and tested from the VYNPS reactor vessel since VYNPS is not a host ISP plant for providing surveillance capsules. As indicated in the test matrix of BWRVIP-86-A, reactor vessel weld and plate surveillance materials from Susquehanna Unit 1 have been selected from among the existing plant surveillance programs to represent the corresponding limiting plate and weld material in the VYNPS reactor vessel. The two remaining capsules will continue to reside in the VYNPS reactor vessel in case they are needed in the future as a contingency. The peak neutron fluence at 33 effective full-power years (EFPY) and 4.827×10^8 megawatt-hours (EPU conditions at the end of the current VYNPS license term) at the 1/4 thickness (T) is 2.35×10^{17} neutrons per centimeter squared (n/cm^2). Since this fluence value is less than that projected to be received by the representative surveillance materials from Susquehanna Unit 1, the withdrawal schedule for the BWRVIP ISP does not need to be changed and the BWRVIP ISP will provide adequate surveillance data to monitor the impact of neutron radiation on the VYNPS reactor vessel at EPU conditions.

Appendix H of 10 CFR Part 50 requires that an ISP used as a basis for a licensee-implemented reactor vessel surveillance program be reviewed and approved by the NRC staff. The ISP to be used by the applicant is a program that was developed by the BWRVIP. The licensee will apply the BWRVIP ISP as the method by which the VYNPS reactor vessel will comply with the requirements of 10 CFR Part 50, Appendix H. The BWRVIP ISP identifies capsules that must be tested to monitor neutron radiation embrittlement for all licensees participating in the ISP, and identifies capsules that need not be tested (standby capsules). These untested capsules were originally part of the licensee's plant-specific surveillance program and have received significant amounts of neutron radiation.

In the most recent staff-approved version of the ISP, the reactor vessel surveillance capsules from VYNPS have not been designated for removal and testing to support the ISP. However, as addressed in 10 CFR Part 50, Appendix H, Section III (C)(1)(d) and in the staff-approved BWRVIP ISP, maintaining adequate contingencies to support potential changes to the program is an important part of any ISP. As discussed in the NRC's SE for VYNPS Amendment No. 218, the licensee will revise UFSAR Section 4.2.6 to state, in part, that:

> The Vermont Yankee Nuclear Power Station is not a host ISP plant for providing surveillance capsules; however, the remaining two VYNPS materials surveillance capsules will continue to reside in the reactor in case they are needed in the future as a contingency.

Based on the licensee's commitment to maintain the capsules in the reactor vessel, the licensee has satisfied the contingency of 10 CFR Part 50, Appendix H, Section III (C)(1)(d).

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the reactor vessel surveillance withdrawal schedule and concludes that the licensee has adequately addressed changes in neutron fluence and their effects on the schedule. The NRC staff further concludes that the reactor vessel capsule withdrawal schedule is appropriate to ensure that the material surveillance program will continue to meet the requirements of 10 CFR Part 50,

Appendix H, and 10 CFR 50.60, and will provide the licensee with information to ensure continued compliance with draft GDC-9, 33, and 34 in this respect following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the reactor vessel material surveillance program.

2.1.2 Pressure-Temperature Limits and Upper-Shelf Energy (USE)

Regulatory Evaluation

Appendix G of 10 CFR Part 50 provides fracture toughness requirements for ferritic materials (low alloy steel or carbon steel) materials in the RCPB, including requirements on the USE values used for assessing the safety margins of the reactor vessel materials against ductile tearing and requirements for calculating pressure-temperature (P-T) limits for the plant. These P-T limits are established to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests. The NRC staff's review of P-T limits covered the P-T limits methodology and the calculations for the number of EFPY specified for the proposed EPU, considering neutron embrittlement effects and using linear elastic fracture mechanics. The NRC's acceptance criteria for USE and P-T limits evaluations are based on: (1) draft GDC-9, insofar as it requires that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage; (2) draft GDC-33, insofar as it requires that the RCPB be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant; (3) draft GDC-34 insofar as it requires that the RCPB be designed to minimize the probability of rapidly propagating type failures; (4) draft GDC-35 insofar as it requires that service temperatures for RCPB components constructed of ferritic materials ensure the structural integrity of such components when subjected to potential loadings;
(5) 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB; and (6) 10 CFR 50.60, which requires compliance with the requirements of 10 CFR Part 50, Appendix G. Specific review criteria are contained in SRP Section 5.3.2 and other guidance provided in Matrix 1 of RS-001.

Technical Evaluation

USE Value Calculations

Appendix G of 10 CFR Part 50 provides the NRC's criteria for maintaining acceptable levels of USE for the reactor vessel beltline materials of operating reactors throughout the licensed lives of the facilities. The rule requires reactor vessel beltline materials to have a minimum USE value of 75 ft-lb in the unirradiated condition, and to maintain a minimum USE value above 50 ft-lb throughout the life of the facility, unless it can be demonstrated through analyses that lower values of USE would provide acceptable margins of safety against fracture equivalent to those required by Appendix G of Section XI to the ASME Code. The rule also mandates that the methods used to calculate USE values must account for the effects of neutron irradiation on the USE values for the materials and must incorporate any relevant reactor vessel surveillance capsule data that are reported through implementation of a plant's 10 CFR Part 50, Appendix H, reactor vessel materials surveillance program.

By letter dated April 30, 1993, the Boiling Water Reactor Owner's Group (BWROG) submitted a topical report entitled, "10 CFR 50, Appendix G Equivalent Margins Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels," to document that BWR reactor vessels could meet the margins of safety against fracture equivalent to those required by Appendix G of the ASME Code for Charpy USE values less than 50 ft-lb. In a letter dated December 8, 1993, the NRC staff concluded that the topical report demonstrates that the materials evaluated have the margins of safety against fracture equivalent to Appendix G of the ASME Code, in accordance with Appendix G of 10 CFR Part 50. In this report, the BWROG derived through statistical analysis the initial USE values for materials that originally did not have documented Charpy USE values. Using these statistically derived Charpy USE values, the BWROG predicted the end-of life (40 years of operation) USE values in accordance with Position 1.2 in Regulatory Guide (RG) 1.99, Revision 2. According to this RG, the decrease in USE is dependent upon the amount of copper in the material and the neutron fluence predicted for the material. The BWROG analysis determined that the minimum allowable Charpy USE in the transverse direction for base metal and along the weld for weld metal was 35 ft-lb.

General Electric (GE) performed an update to the USE equivalent margins analysis, which is documented in Electric Power Research Institute (EPRI) Report TR-113596, "BWR Vessel and Internals Project BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines (BWRVIP-74)", dated September 1999. EPRI Report TR-113596 provides a bounding Charpy USE for BWR plants for 54 EFPY. The analysis in EPRI TR-113596 determined the reduction in the unirradiated Charpy USE resulting from neutron radiation using the methodology in Position 1.2 in RG 1.99, Revision 2. Using this methodology and a correction factor of 65% for conversion of the longitudinal properties to transverse properties, the lowest irradiated Charpy USE at 54 EFPY for all BWR/3-6 plates is projected to be 45 ft-lb. The correction factor for specimen orientation in plates is based on NRC Branch Technical Position MTEB 5-2. Using the RG methodology, the lowest irradiated Charpy USE at 54 EFPY for shielded metal arc welds is projected to be 51.1 ft-lb. The value for the BWR/3-6 plates is greater than the 35 ft-lb minimum allowable and thus will meet the margins of safety against fracture equivalent to those required by 10 CFR Part 50, Appendix G. The value for the shielded metal arc weld is greater than the 50 ft-lb criteria in 10 CFR Part 50, Appendix G. EPRI Report TR-113596 indicates that the percent reduction in Charpy USE for the limiting BWR/3-6 plates and shielded metal arc welds are 23.5% and 39%, respectively. Therefore, to demonstrate that beltline materials meet the criteria specified in the report, licensees must demonstrate that the projected percent reduction in Charpy USE for their beltline materials are less than those specified for the limiting BWR/3-6 plates and the shielded metal arc welds. Licensees also have to show that the actual percent reduction in Charpy USE for their surveillance weld and plate are less than or equal to the values projected using the methodology in Position 1.2 in RG 1.99, Revision 2. Beltline materials that meet these criteria will meet the margins of safety against fracture equivalent to those required by Appendix G to 10 CFR Part 50.

The licensee discussed the impact of the EPU on the Charpy USE values for the reactor vessel beltline materials in Section 3.2.1 of Attachment 4 to Reference 1. This section indicates that projected percent reduction in Charpy USE for the plates is less than 23.5% and the projected percent reduction in Charpy USE for the shielded metal arc welds is less than 39%. However, the actual decreases in Charpy USE for the surveillance plate and shield metal arc weld are greater than the values predicted using RG 1.99, Revision 2. As discussed below, the results from the surveillance test data are not necessary for evaluating the impact of neutron irradiation on the reactor vessel beltline materials, at this time.

RG 1.99, Revision 2 has two methods for determining the percent reduction in Charpy USE. In Position 1.2, the percent reduction in Charpy USE is determined from Figure 2 in the RG which is based on the neutron fluence and the amount of copper in the material. In the second method, identified as Position 2.2, the percent reduction in Charpy USE is determined from surveillance data. RG 1.99, Revision 2 indicates surveillance data may be used for determining the Charpy USE when two or more credible surveillance data sets become available from the reactor. Since only one data set from a plate and a weld is presently available, RG 1.99, Revision 2 would recommend that the Charpy USE be determined using Position 1.2. Using Figure 2 in RG 1.99, Revision 2, the staff determined that the percent reduction in Charpy USE at a neutron fluence of $2.35 \times 10^{17} n/cm^2$ (neutron fluence at end of the current license with EPU conditions) was 9.5% for the plate material and 8.0% for the shielded metal arc weld material. The analysis in EPRI TR-113596 utilized an unirradiated Charpy USE in the longitudinal direction of 91 ft-lb for BWR/3-6 plates and 84.5 ft-lb for shield metal arc welds. The value for the plates is the lowest value from the database and is less than the lower 95/95 confidence value. The value for the shielded metal arc welds is the value corresponding to the lower 95/95 confidence value. Since these values are statistically determined with at least 95/95 confidence, the values may be used in the evaluation of Charpy USE. The Charpy USE for plate material in the transverse direction would be 59 ft-lb, which is 65% of 91 ft-lb. Using these unirradiated values for the Charpy USE for the plate and the weld and the percent reduction determined using Figure 2 in RG 1.99, Revision 2, the Charpy USE at a neutron fluence of $2.35 \times 10^{17} n/cm^2$ is 53 ft-lb for the plate material and 78 ft-lb for the weld material. Since both the weld metal and plate material are projected to have Charpy USE greater than 50 ft-lb at expiration of the license at EPU conditions, the reactor vessel materials satisfy the requirements of 10 CFR Part 50, Appendix G. As discussed in Section 2.1.1, the surveillance data from Susquehanna Unit 1 will be utilized to monitor the impact of neutron radiation on the VYNPS beltline materials. In accordance with Appendix G, 10 CFR Part 50, the licensee must re-evaluate the impact of neutron radiation on Charpy USE, when the surveillance data from Susquehanna Unit 1 become available.

Pressure-Temperature Limit Calculations

Section IV.A.2 of 10 CFR Part 50, Appendix G, requires that the P-T limits for operating reactors be at least as conservative as those that would be generated if the methods of calculation in the ASME Code, Section XI, Appendix G, were used to calculate the P-T limits. The rule also requires that the P-T limit calculations account for the effects of neutron irradiation on the P-T limit values for the reactor vessel beltline materials and incorporate any relevant reactor vessel surveillance capsule data that are required to be reported as part of the licensee's implementation of its 10 CFR Part 50, Appendix H, reactor vessel materials surveillance program.

Section 3.2.1 of Attachment 4 to Reference 1 indicates that the P-T limit curves contained in the TSs remain bounding for EPU conditions. The VYNPS P-T limit curves were approved in VYNPS Amendment No. 218 dated March 29, 2004. Tables 2-1 and 2-2 of Attachment 2 to the licensee's letter dated March 26, 2003 (application associated with Amendment No. 218), provided the adjusted reference temperature (ART) values for the limiting material as 57°F at 1/4T fluence (2.20×10^{17} n/cm^2) and 48°F at 3/4T fluence (1.20×10^{17} n/cm^2). Section 2.0 of Attachment 2 to the March 26, 2003, letter states that for purposes of determining the P-T curves for the vessel core region material, VYNPS has elected to maintain the more conservative ART values previously used by VYNPS (89°F at the 1/4T point and 73°F at the

3/4T point). The licensee's submittal states that, based on RG 1.99, Revision 2, lower values of ART could have been used.

The NRC staff's assessment included an independent calculation of the ART values for both the 1/4T and 3/4T locations of the VYNPS reactor vessel beltline regions based on the revised 33 EFPY neutron fluence specified in the submittal for VYNPS for EPU conditions. The staff confirmed, using the methodology of RG 1.99, Revision 2, that the limiting beltline material was the reactor vessel plate 1-14 with an ART of 58°F at the 1/4T location and 53°F at the 3/4T location. Item 13 in Table 1, "Proposed OL and TS Changes," in Attachment 1 to Reference 1, indicates the analytical methods used in the March 26, 2003, letter are unchanged; however, the peak neutron fluence increased to 3.18×10^{17} n/cm^2. The neutron fluence methodology was determined to be consistent with the guidance in RG 1.190 as discussed in the NRC's SE for Amendment No. 218. Previously, the P-T limit curves were based on a peak vessel fluence value of 1.24×10^{18} n/cm^2 resulting in the limiting material (reactor vessel plate 1-14) having an ART of 89°F at the 1/4T location and 73°F at the 3/4T location. Since the staff has confirmed that the previous ART values bound the revised ART values for EPU conditions, the staff agrees that the P-T limit curves contained in the TSs remain bounding for EPU conditions.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the USE values for the reactor vessel beltline materials and P-T limits for the plant. The staff concludes that the licensee has adequately addressed changes in neutron fluence and their effects on the USE values for VYNPS reactor vessel beltline materials and the P-T limits for the plant. The staff concludes that the VYNPS beltline materials will continue to have acceptable USE, as mandated by 10 CFR Part 50, Appendix G, through the expiration of the current operation license for the facility. The NRC staff further concludes that the licensee has demonstrated the validity of the proposed P-T limits for operation under the proposed EPU conditions. Based on this, the NRC staff concludes that the proposed P-T limits will continue to meet the requirements of 10 CFR Part 50, Appendix G, and 10 CFR 50.60 and will enable the licensee to comply with draft GDC-9, 33, 34, and 35 in this respect following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the proposed P-T limits.

2.1.3 Reactor Internal and Core Support Materials

Regulatory Evaluation

The reactor internals and core supports include structures, systems, and components (SSCs) that perform safety functions or whose failure could affect safety functions performed by other SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the reactor coolant system (RCS)). The NRC staff's review covered the materials' specifications and mechanical properties, welds, weld controls, nondestructive examination procedures, corrosion resistance, and susceptibility to degradation. The NRC's acceptance criteria for reactor internal and core support materials are based on draft GDC-1 and 10 CFR 50.55a for material specifications, controls on welding, and inspection of reactor internals and core supports. Specific review criteria are contained in SRP Section 4.5.2 and BWRVIP-26, and Matrix 1 of RS-001.

Technical Evaluation

Reactor internals and core support materials are subject to the following degradation:

- Crack initiation and growth due to stress-corrosion cracking (SCC), intergranular stress-corrosion cracking (IGSCC) and irradiation assisted stress-corrosion cracking (IASCC);
- Crack initiation and growth due to flow induced vibration;
- Cumulative fatigue damage; and
- Loss of fracture toughness due to thermal aging and neutron embrittlement.

Cumulative fatigue damage and crack initiation and growth due to flow induced vibration are discussed in Section 2.2.2 of this SE. Crack initiation and growth and loss of fracture toughness due to thermal aging and neutron embrittlement are managed through the inservice inspection program that conforms to the requirements of 10 CFR 50.55a and the BWRVIP program. The BWRVIP inspection program supplements the inservice inspection program required by 10 CFR 50.55a. The BWRVIP program is reviewed and approved by the NRC. Section 10.7 of the Attachment 4 to Reference 1 indicates that VYNPS belongs to the BWRVIP organization, and implementation of the procedurally controlled program is consistent with the BWRVIP issued documents. The inspection strategies recommended by the BWRVIP consider the effects of fluence on the applicable components and are based on component configuration and field experience. To mitigate the potential for SCC, IGSCC and IASCC, VYNPS utilizes noble metals applications. Reactor water chemistry conditions are maintained consistent with EPRI, BWRVIP and established industry guidelines, except where technical justification in accordance with BWRVIP-94 has been documented. The licensee concludes that the current inspection program for the reactor internal components is adequate to manage any potential effects of EPU conditions because the increase in neutron fluence resulting from EPU conditions does not significantly increase the potential for degradation.

Since EPU conditions do not significantly increase the potential for degradation, the NRC staff concludes that the current inspection program is acceptable for all reactor vessel internals components except for the top guide and the steam dryer, which are discussed below.

Top Guide

Note 1 in Matrix 1 of Section 2.1 of RS-001 Revision 0 indicates that guidance on the neutron irradiation-related threshold for inspection for IASCC in BWRs is in BWRVIP report BWRVIP-26. The NRC staff's SE for BWRVIP-26 dated December 7, 2000, states that the threshold fluence level for IASCC is 5×10^{20} n/cm^2 (E > 1 million electron volts).

The licensee, in response to a staff RAI (Attachment 1 to Reference 6), indicated the following:

> Of the reactor vessel internal components, only the top guide's integrated flux will exceed 5×10^{20} n/cm^2. VY will commence inspection of critical top guide components in the refueling outage following power uprate. Enhanced Visual Testing (EVT)-1 of top guide grid beams will be performed in accordance with SIL 554 following the sample selection and

inspection frequency of BWRVIP-47 for the CRD guide tubes. In other words, VY will perform inspection of 10% of the total population of cells within twelve years, with one-half (5%) to be completed within six years. The six-year intervals at Vermont Yankee will be defined to be the same as those for the CRD guide tubes. Selection of the cells will be biased to the highest fluence areas in the top guide. However, Vermont Yankee reserves the right to modify the above inspection program should BWRVIP-26 be revised in the future.

The proposed top guide inspection program will inspect a sample of top guides in the highest fluence areas using a technique capable of detecting IASCC at a frequency consistent with industry recommendations. The NRC staff concludes that the proposed program is reasonable and provides an acceptable means to manage the potential for IASCC.

Steam Dryer

The NRC staff raised concerns during the review that the proposed EPU conditions could cause cracks left in service in the steam dryer, following refueling outage (RFO) 24 (spring 2004), to grow to a size that could affect the integrity of the steam dryer and result in the generation of loose parts, which could affect the function of other reactor internals components. In response to a staff RAI, the licensee, in Attachment 2 to Reference 9, reported that the flaws left in service were produced by IGSCC. The licensee quantitatively evaluated the largest flaw, which is located in the dryer drain channel. The crack is located in the heat-affected-zone adjacent to the weld, follows the grain boundary, and exhibits a jagged appearance typical of IGSCC. The crack is not straight and does not have characteristics of a fatigue crack.

The NRC staff's summary of the licensee's quantitative evaluation (contained in Attachment 2 to Reference 14) follows:

- IGSCC crack growth was assumed during future operation at a rate of 5×10^{-5} in/hr on each end, consistent with established BWRVIP growth rates (which is also consistent with the IGSCC rates given in NUREG-0313). This growth will be independent of any fluctuating loading since it is dependent only on the sustained loads, which in this case are the residual stresses from the dryer fabrication. The fuel cycle length at VYNPS (i.e., the time between refueling outages) is nominally 18-months (13,140 hrs). The predicted IGSCC crack growth for the next fuel cycle is then ($5 \times 10^{-5} \times 13{,}140$) or 0.66 inch at each end of the indication. This translates into a projected increase in the crack length from 12.0 inches to 13.32 inches.

- The next step was to evaluate the length at which fatigue crack growth could occur. It is well established that fatigue will only occur when the applied stress intensity factor range exceeds the threshold stress intensity factor (ΔK_{th}). For stainless steel at 550°F, this value is conservatively assumed to be 5 ksi-in$^{1/2}$.

- Strain gage data from an overseas BWR measured on the drain channel was used to determine the magnitude of the peak alternating stresses that would be present. A conservative adjustment to this peak stress for use in conjunction with the VYNPS drain channel was performed by scaling the overseas plant stress to the ratio of the square of the steam line velocity at VYNPS at EPU conditions to the square of the steam velocity at the overseas plant. The use of square of the steam line velocity is consistent with the

recommendations in Appendix N of the ASME Code, Section III that deals with the treatment of dynamic loads. Also, the exponent 2 is consistent with the average of the exponents obtained in the development of the generic fluctuating load definition.

- The results of this evaluation established that the flaw would be predicted to reach 13.32 inches after 18 months. The associated ΔK for this longer crack is below the critical ΔK_{th}. Only when the crack reaches 15.6 inches would the crack reach the ΔK_{th} at which fatigue crack extension could take place. This would be predicted to occur after 32 months of operation (i.e., longer than the 18-month fuel cycle).

The licensee's conclusion that the flaws remaining in service will not cause loose parts is based on the premise that as long as the flaws are not subjected to crack growth resulting from fatigue, they will grow at a slow enough rate during each fuel cycle that crack growth can be monitored by inservice inspection. This conclusion is based on industry experience with IGSCC flaws in BWR steam dryers. The licensee has also performed a qualitative engineering assessment of all the flaws and determined that there is additional margin in the design of the components that will prevent their failure.

In Reference 33, Attachments 1 and 10, the licensee provided commitments regarding steam dryer inspections. During RFO 24 (spring 2004), the licensee performed a baseline visual inspection of all accessible, susceptible locations of the steam dryer consistent with GE Services Information Letter (SIL) No. 644, Revision 1, "BWR Steam Dryer Integrity," dated November 9, 2004. The licensee had originally planned to conduct visual inspection of all accessible, susceptible locations of the steam dryer during RFO 25 (fall 2005), RFO 26 (spring 2007), and RFO 27 (fall 2008). This plan was based on implementation of EPU prior to RFO 25. However, since the EPU will be implemented after RFO 25, the licensee committed to perform visual inspection of all accessible susceptible locations of the steam dryer during RFO 26, RFO 27, and RFO 28 (spring 2010). During RFO 25, the licensee committed to perform a visual inspection of the steam dryer modifications, flaws left "as-is," and repairs made during RFO 24. New information on indications identified in previous inspections that were not repaired will be compared with the previous information to validate crack growth projections.

In Supplement 42 to its EPU license amendment request (Reference 43), the licensee documented the results of the steam dryer inspection during RFO 25 and its analysis of those results. In particular, the licensee found no indications in the recent steam dryer modifications (including the gussets or their weld connections) nor any changes in previous left-as-is indications. The licensee did identify about 50 new indications in the end plates used to separate the internal vane assemblies in the steam dryer. The end plates are fabricated from 3/16-inch thick Type 304 stainless steel, and are 48 inches high and 8 inches wide with a channel shape that has a 1.25-inch flange on each side. Most of the indications are tight horizontal IGSCC cracks that appear to be 1.25 inches long on the inlet flow side of the flange next to the dryer shell. The licensee did not identify any indications in the 48-inch vertical welds that hold the end plates in place on both the inlet and outlet flow sides of the vane assemblies. The licensee identified six fatigue cracks in the fillet welds where the bottom of the end plates fit into the drain trough. In that the end plates are notched into the drain troughs, the end plate to trough welds do not perform a structural function for the assembly. The licensee also reported that the previously identified steam dryer indications (including those in the end plates) had not grown in size. The licensee believed that the enhanced inspection technique applied during RFO 25 might have resulted in the identification of the additional indications in the steam dryer.

In evaluating the end plate indications, the licensee determined that there were no structural consequences from the steam dryer indications, nor was there a potential for the generation of loose parts if it is postulated that the end plate indications propagate across the entire 8-inch end plate width.

The NRC staff does not believe that IGSCC will arrest; however, the licensee can propose a revised frequency of examination based on observed crack growth. Based on the licensee's analysis, the industry experience with IGSCC, and the licensee's commitment to institute an inspection program as discussed above, the staff concludes that there is reasonable assurance that the steam dryer can be safely operated at EPU conditions with flaws discovered during the spring 2004 and fall 2005 outages.

With regard to steam dryer experience at other nuclear power plants, the NRC staff discussed with the licensee the identification of fatigue cracks in the steam dryers at Dresden Units 2 and 3 during their fall 2005 outages. The licensee reviewed the steam dryer damage that occurred in those plants at the connection between the gussets and lower cover plate. The licensee verified that the pressure loads had been properly evaluated for the gusset to cover plate connection in the steam dryer at VYNPS, and that those loads will not cause fatigue damage to its gussets or connections. Additional discussion of steam dryer modeling is provided in Section 2.2.6.2.1 of this SE.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the susceptibility of reactor internal and core support materials to known degradation mechanisms and concludes that the licensee has identified appropriate degradation management programs to address the effects of changes in operating temperature and neutron fluence on the integrity of reactor internal and core support materials. The NRC staff further concludes that the licensee has demonstrated that the reactor internal and core support materials will continue to be acceptable and will continue to meet the requirements of draft GDC-1 and 10 CFR 50.55a with respect to material specifications, welding controls, and inspection following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to reactor internal and core support materials.

2.1.4 Reactor Coolant Pressure Boundary Materials

Regulatory Evaluation

The RCPB defines the boundary of systems and components containing the high-pressure fluids produced in the reactor. The NRC staff's review of RCPB materials covered their specifications, compatibility with the reactor coolant, fabrication and processing, susceptibility to degradation, and degradation management programs. The NRC's acceptance criteria for RCPB materials are based on (1) 10 CFR 50.55a and draft GDC-1, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) draft GDC-40 and 42, insofar as they require that protection be provided for engineered safety features (ESFs) against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a loss-of-coolant accident (LOCA); (3) draft GDC-9 and 33, insofar as they require that the RCPB be designed and constructed so as to have an

exceedingly low probability of gross rupture or significant leakage; (4) draft GDC-34 insofar as it requires that the RCPB be designed to minimize the probability of rapidly propagating type failures; (5) draft GDC-35 insofar as it requires that service temperatures for RCPB components constructed of ferritic materials ensure the structural integrity of such components when subjected to potential loadings; and (6) 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB. Specific review criteria are contained in SRP Section 5.2.3 and other guidance provided in Matrix 1 of RS-001. Additional review guidance for thermal embrittlement of cast austenitic stainless steel components is contained in a letter from C. Grimes, NRC, to D. Walters, Nuclear Energy Institute (NEI), dated May 19, 2000.

Technical Evaluation

The NRC staff reviewed Section 3.5.1, "Reactor Coolant Pressure Boundary Piping" and Section 3.6, "Reactor Recirculation System" of Attachment 4 to Reference 1. The licensee stated that the plant-specific evaluation process is consistent with the methodology described in Appendix K of GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," NEDC-32424P-A, February 1999 [known as ELTR1]. This process involves comparing existing data such as temperatures, pressures, and flow rates, with corresponding data at EPU conditions to determine piping system acceptability. Existing stresses and pipe support loads are increased to evaluate EPU conditions. The licensee stated that these revised stresses and pipe support loads were evaluated and are within acceptable design limits.

According to the licensee, flow, pressure, temperature, and mechanical loading for most of the RCPB piping systems do not increase as a result of the EPU at VYNPS. The Main Steam (MS) and Feedwater (FW) systems, under EPU conditions, will experience an increase in flow due to higher steam flow from the reactor. However, the licensee stated that changes in fluid conditions experienced by the MS and FW systems are minor.

The NRC staff generally agrees with the licensee's conclusion that the RCPB materials will not be significantly affected after the EPU is implemented. However, because of the history of IGSCC in BWR RCPB piping, the staff requested additional information regarding the licensee's disposition of the reactor recirculation system (RRS) piping. The licensee responded in Reference 6 and stated that there is no significant change in temperature, pressure and flow rate for the RRS piping resulting from the EPU. According to the licensee, the RRS operating temperature will decrease by less than 1% while the operating pressure changes by less than 1% (RRS pump suction pressure decreases by less than 1% and the RRS pump discharge pressure increases by less than 1%). The staff also requested that the licensee identify the materials of construction for the RRS piping and discuss the effect of the EPU on the material. The licensee responded that all of the piping in the RRS is low carbon Type 316 stainless steel. The staff inquired as to whether the RRS contained any flaws that had been evaluated and left in place per ASME Code, Section XI. The licensee responded that there are no known flaws in the RRS piping. According to the licensee, in addition to IGSCC mitigation measures described in letters dated July 28, 1988, and July 15, 1989, VYNPS has adopted hydrogen water chemistry with noble metal chemical addition, and hydrogen injection rates will be adjusted as power is increased to maintain protection.

The NRC staff finds that the VYNPS RCPB would not be significantly affected by the proposed EPU. This finding is based on the fact that low carbon Type 316 stainless steel is resistant to IGSCC, is suitable for use in BWR RRS piping, and is classified as a category A material per NUREG-0313. The absence of any known flaws in the RRS piping that require monitoring and management further adds to the staff's confidence that the RCPB at VYNPS will continue to meet all required licensing basis requirements after the EPU is implemented.

Conclusion

The NRC staff concludes that the licensee has demonstrated that the RCPB materials will continue to perform as designed following implementation of the proposed EPU and will continue to meet the requirements of draft GDC-1, 9, 33, 34, 35, 40, and 42, 10 CFR Part 50, Appendix G, and 10 CFR 50.55a. Therefore, the NRC finds the proposed EPU acceptable with respect to RCPB materials.

2.1.5 Protective Coating Systems (Paints) - Organic Materials

Regulatory Evaluation

Protective coating systems (paints) provide a means for protecting the surfaces of facilities and equipment from corrosion and contamination from radionuclides and also provide wear protection during plant operation and maintenance activities. The NRC staff's review covered protective coating systems used inside the containment for their suitability for and stability under design-basis loss-of-coolant accident (DBLOCA) conditions, considering radiation and chemical effects. It also included reviewing the effect of spalled coatings on net positive suction head (NPSH) in the residual heat removal (RHR) and core spray (CS) system pumps. The NRC's acceptance criteria for protective coating systems are based on (1) 10 CFR Part 50, Appendix B, which states quality assurance requirements for the design, fabrication, and construction of safety-related SSCs and (2) Regulatory Guide 1.54, Revision 1, for guidance on application and performance monitoring of coatings in nuclear power plants. Specific review criteria are contained in SRP Section 6.1.2.

Technical Evaluation

In order to provide required core and containment cooling following a DBLOCA, the RHR and CS pumps should maintain an adequate NPSH. This NPSH can be significantly reduced if the debris generated by failed coatings is deposited on the pump suction strainers. The methodology used by the licensee for determining the amount of the debris generated and transported to the strainers is based on the NEDO-32686 Report, "BWROG Utility Resolution Guidance for ECCS Suction Strainer Blockage." The licensee considered two types of debris which could be generated in post-LOCA environment: protective coatings and organic materials. The protective coatings consist of inorganic zinc with an epoxy top coat. The bounding value for the amount of this coating stripped by a LOCA jet is 85 pounds-mass and it remains unchanged after the proposed EPU. The organic materials consist of paint chips. Their effect on NPSH was evaluated by the tests performed on the ECCS strainer design under the simulated LOCA conditions. The results of the tests have indicated that for the strainer approach velocity and suppression pool turbulence predicted for the post-LOCA environment in VYNPS, the paint chips will not contribute to the head loss in the pump strainers. Neither will they contribute to the head loss after EPU because the strainer approach velocity, the

suppression pool turbulence and the form of the failed coating will not change. Since the debris accumulated on the pump suction strainers will not cause increased head loss, the licensee concluded that the proposed EPU will not change the NPSH in the RHR and CS pumps.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on protective coating systems and concludes that the licensee has appropriately addressed the impact of changes in conditions following a DBLOCA and their effects on the protective coatings. The NRC staff further concludes that the licensee has demonstrated that the protective coatings will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of 10 CFR Part 50, Appendix B. Therefore, the NRC staff finds the proposed EPU acceptable with respect to protective coatings systems.

2.1.6 Flow-Accelerated Corrosion

Regulatory Evaluation

Flow-accelerated corrosion (FAC) is a corrosion mechanism occurring in carbon steel components exposed to flowing single- or two-phase water. Components made from stainless steel are immune to FAC, and FAC is significantly reduced in components containing small amounts of chromium or molybdenum. The rates of material loss due to FAC depend on velocity of flow, fluid temperature, steam quality, oxygen content, and pH. During plant operation, control of these parameters is limited and the optimum conditions for minimizing FAC effects, in most cases, cannot be achieved. Loss of material by FAC will, therefore, occur. The NRC staff has reviewed the effects of the proposed EPU on FAC and the adequacy of the licensee's FAC program to predict the rate of loss so that repair or replacement of damaged components could be made before they reach critical thickness. The licensee's FAC program is based on NUREG-1344, GL 89-08, and the guidelines in EPRI Report NSAC-202L-R2. It consists of predicting loss of material using the CHECWORKS™ computer code, and visual inspection and volumetric examination of the affected components. The NRC's acceptance criteria are based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC.

Technical Evaluation

VYNPS has a procedurally controlled FAC inspection program relying on selective component inspections to provide a measure of confidence in the condition of components susceptible to FAC. The program is based on the guidelines developed by EPRI and the American Society of Mechanical Engineers (ASME). The selection of the components for these inspections and their scope is determined by the susceptibility of these components to FAC as determined by the licensee developed criteria. The components which either indicate damage caused by FAC or show signs that such damage could occur before the next outage, are repaired or replaced. The licensee determines predicted rates of wear by FAC using the CHECWORKS™ program. This program calculates wear rates due to FAC from the operating parameters in the plant. Some of these parameters will be affected by the proposed EPU and their changes will have an impact on FAC wear rates. Increase in velocity of flow of single- or two-phase fluid (which is expected to occur in some lines) will produce higher FAC wear rates. The licensee has determined that an increase in the velocities in the main steam line and feedwater lines will

cause proportional increases in FAC wear rates. The proposed EPU will also have an effect on moisture and oxygen content, and on temperature. A change of these parameters will impact FAC in the main steam drains, moisture separator drains, and the turbine cross around system piping and will require the licensee to suitably modify the FAC inspection program for these systems. The piping in the extraction steam system at VYNPS is made from material immune to FAC. In response to an NRC staff RAI, the licensee, in Reference 6, provided information on typical expected changes due to FAC in several plant systems subsequent to EPU. After reviewing this information, the staff concurred with the licensee's assessment that the proposed EPU could cause an increase of FAC in some plant systems. Accordingly, the licensee plans to modify the inputs to the CHECWORKS™ program and introduce some changes to the FAC inspection program to account for the changes due to the EPU.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effect of the proposed EPU on the FAC analysis for the plant and concludes that the licensee has adequately addressed changes in the plant operating conditions on the FAC analysis. The NRC staff further concludes that the licensee has demonstrated that the updated analyses will predict the loss of material by FAC and will ensure timely repair or replacement of degraded components following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to FAC.

2.1.7 Reactor Water Cleanup System

Regulatory Evaluation

The reactor water cleanup system (RWCS) provides a means for maintaining reactor water quality by filtration and ion exchange and a path for removal of reactor coolant when necessary. Portions of the RWCS comprise the RCPB. The NRC staff's review of the RWCS included component design parameters for flow, temperature, pressure, heat removal capability, and impurity removal capability; and the instrumentation and process controls for proper system operation and isolation. The review consisted of evaluating the adequacy of the plant's TSs in these areas under the proposed EPU conditions. The NRC's acceptance criteria for the RWCS are based on (1) draft GDC-9, insofar as it requires that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage; (2) draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents; and (3) draft GDC-51, insofar as it requires that systems that contain radioactivity be designed with appropriate confinement. Specific review criteria are contained in SRP Section 5.4.8.

Technical Evaluation

The licensee reviewed functional capability of the RWCS to ensure that after the proposed EPU, it will meet the requirements of draft GDC-9, GDC-51, and GDC-70. The review has indicated that, although some operating system parameters will change slightly after the proposed EPU, the system will be able to perform its functions in a satisfactory manner and will meet the requirements imposed by the draft GDCs. The slight changes in operating system parameters result primarily due to increased feedwater flow. The licensee has determined that the changes due to the EPU will produce no detectable effect on system performance including

any current analysis in the program for the RWCS Motor Operated Valves, described in Generic Letter 89-10. Higher feedwater flow will result in a higher concentration of iron in the reactor water and a corresponding increase in its conductivity. The capability to remove the iron and other impurities from the processed water through the RWCS will be somewhat affected by the EPU, but the capability of the radwaste system will not be exceeded and the only concern will be a need for more frequent backwash of the filter demineralizer. The operational temperature of the water handled by the RWCS will be reduced by 1.7°F, which is too small to have any significant effect on the performance of the system. The licensee also verified that for all piping and components pressure and temperature, the ratings will remain unaffected and because of negligible changes of the system process parameters, no instrumentation setpoints will need to be readjusted. The NRC staff reviewed the licensee's evaluation and concurs that the proposed EPU will introduce only insignificant changes in the RWCS operating parameters, which will not affect satisfactory performance of its intended functions.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed EPU on the RWCS and concludes that the licensee has adequately addressed changes in impurity levels and pressure and their effects on the RWCS. The NRC staff further concludes that the licensee has demonstrated that the RWCS will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of draft GDC-9, 51, and 70. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the RWCS.

2.1.8 Additional Review Area - Reactor Vessel Feedwater Nozzle

Regulatory Evaluation

Reactor vessel feedwater (RVFW) nozzles are part of the RCPB. Therefore, RVFW nozzles must meet the regulatory requirements for RCPB materials. The regulatory requirements for RCPB materials are discussed in Section 2.1.4 of this SE. The NRC's acceptance criteria for RVFW nozzle materials are based on: (1) 10 CFR 50.55a and draft GDC-1, insofar as they require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) draft GDC-9 and 33, insofar as they require that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage; (3) draft GDC-34 insofar as it requires that the RCPB be designed to minimize the probability of rapidly propagating type failures; (4) draft GDC-35 insofar as it requires that service temperatures for RCPB components constructed of ferritic materials ensure the structural integrity of such components when subjected to potential loadings; and (5) 10 CFR Part 50, Appendix G, which specifies fracture toughness requirements for ferritic components of the RCPB.

Technical Evaluation

Section 50.55a of 10 CFR specifies ASME Code inspection requirements for RCPB materials and RVFW nozzles. As a result of cracking observed in the blend radius area of BWR RVFW nozzles, the staff issued, in a letter dated November 13, 1980, augmented inspection

requirements. The augmented inspection requirements for the RVFW nozzle blend radius area are discussed in NUREG-0619.

In a letter to the BWROG dated March 10, 2000, the NRC staff accepted an alternative to the recommendations set forth in NUREG-0619. The alternative inspection requirements for the RVFW nozzle blend radius area are contained in BWROG Topical Report GE-NE-523-A71-0594, Revision 1, dated August 1999.

Section 3.2.2 of Attachment 4 to Reference 1 indicates EPU conditions do not impact the integrity of RVFW nozzles. For the RVFW nozzle blend radius location, in addition to a stress and fatigue analysis, a fracture mechanics analysis was performed in conjunction with inner surface exams and cycle counting to assure potential crack growth is small in relation to ASME Code, Section XI limits. The ultrasonic testing (UT) inspection of the inner surface of the RVFW nozzles is based on BWROG Topical Report GE-NE-523-A71-0594, Revision 1, as an alternative to NUREG-0619.

The fracture mechanics analysis evaluates crack growth for conservative design transients. Design cycles are then monitored through plant procedure. In response to a staff RAI, the licensee (a) compared the temperatures and pressures for EPU conditions to the temperature and pressure conditions assumed in the fracture mechanics analysis for Startup/Shutdown, Hot Standby On/Off Flow and Leak Pattern Changes in Power design transients, (b) compared the temperature distribution along the nozzle wall and the heat transfer coefficients used in the fracture mechanics analysis to the temperature distribution and heat transfer characteristics expected during EPU conditions, and (c) compared the frequency of events assumed in the fracture mechanics analysis to the frequency of events expected during EPU conditions. Although VYNPS does not expect the frequency of the events to increase due to EPU conditions, VYNPS monitors these events and can adjust the frequency of the UT examination. The conservative design transients used in the fracture mechanics evaluation conservatively bound changes under EPU conditions. The thermal model used in this assessment employed heat transfer coefficients and a temperature profile that remain conservative under EPU conditions. Therefore, the licensee concluded that the cycle limits and inspection frequency are not affected by EPU conditions. Since the licensee has demonstrated that the fracture mechanics analysis bounds EPU conditions, the licensee has demonstrated that the UT inspection program for the RVFW nozzle inner blend radius is adequate for EPU conditions.

The licensee has provided composite curves in its P-T limits that satisfy the requirements of Appendix G, 10 CFR Part 50 for the beltline, nozzles, including the feedwater nozzle, and upper shell regions of the reactor vessel. The feedwater nozzles do not receive neutron radiation that is high enough to have their fracture toughness affected by the EPU. Therefore, the feedwater nozzle analysis for compliance with Appendix G, 10 CFR Part 50 is not affected by EPU.

Conclusion

The NRC staff concludes that the licensee has demonstrated that the RVFW nozzle materials will continue to be acceptable following implementation of the proposed EPU and will continue to meet the requirements of draft GDC-1, 9, 33, 34, 35, 40, and 42, 10 CFR Part 50, Appendix G, and 10 CFR 50.55a. Therefore, the NRC staff finds the proposed EPU acceptable with respect to RVFW nozzle materials.

2.2 Mechanical and Civil Engineering

2.2.1 Pipe Rupture Locations and Associated Dynamic Effects

2.2.1.1 Regulatory Evaluation

Structures, systems, and components (SSCs) important to safety at nuclear power plants could be impacted by the pipe-whip dynamic effects of a pipe rupture. The NRC staff conducted a review of pipe rupture analyses to ensure that SSCs important to safety at VYNPS are adequately protected from the effects of pipe ruptures. The staff's review covered (1) the implementation of criteria for defining pipe break and crack locations and configurations; (2) the implementation of criteria dealing with special features, such as augmented inservice inspection (ISI) programs or the use of special protective devices such as pipe-whip restraints; (3) pipe-whip dynamic analyses and results, including the jet thrust and impingement forcing functions and pipe-whip dynamic effects; and (4) the design adequacy of supports for SSCs provided to ensure that the intended design functions of the SSCs will not be impaired to an unacceptable level as a result of pipe-whip or jet impingement loadings. The NRC's acceptance criteria are based on draft GDC-40 insofar as it requires that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures. Specific review criteria are contained in SRP Section 3.6.2.

2.2.1.2 Technical Evaluation

The design basis for the original VYNPS reactor coolant pressure boundary (RCPB) piping, components and supports systems include postulated breaks in all high energy piping above 1 inch in diameter. For the proposed EPU, the reactor dome pressure remains unchanged. There is no significant increase in temperature and flow rate in RCPB piping except the FW and main steam lines (MSLs) where there is about a 20% increase in FW and steam flow rates. The licensee determined that the increase in flow rate during normal operation at EPU conditions has no effect on the mass, energy releases and the break flow velocity, since they are determined by reactor pressure (which is unchanged by EPU), the size of the pipe (which is unchanged by EPU), and reactor fluid conditions (which is only slightly different for FW). Therefore, the loads associated with the thrust at the break locations, jet impingement loadings at and away from the break locations, and asymmetric pressurization remain unchanged for the EPU conditions at VYNPS.

The licensee reviewed pipe stresses and fatigue usage factor calculations for the as-built configurations of VYNPS piping systems at EPU conditions. The loads in the piping structural evaluation include seismic loads, thermal loads, SRV discharge loads, and LOCA loads including pool swell, condensation oscillation (CO), and chugging loads. The seismic loads are not affected by the EPU. As a result of its review, the licensee determined that the EPU conditions are bounded by the design basis LOCA loads based on the test conditions defining the pool swell, CO and chugging loads for VYNPS. The licensee also determined that the parameters used to define the SRV loads are not affected by the EPU conditions and, therefore, the existing SRV loads for VYNPS remain applicable at EPU conditions. No new postulated pipe break locations were identified by the licensee.

On the basis of its review, the NRC staff finds the licensee's analysis methodology associated with the break locations and the associated dynamic effects of SRV and LOCA loads to be adequate and acceptable based on SRP Section 3.6.2.

2.2.1.3 Conclusion

The NRC staff has reviewed the licensee's piping evaluations related to determinations of rupture locations and associated dynamic effects, and concludes that the licensee has adequately addressed the effects of the proposed EPU for VYNPS. The NRC staff further concludes that the licensee has demonstrated that the ESFs will continue to meet the requirements of draft GDC-40 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the determination of pipe rupture locations and dynamic effects associated with the postulated rupture of piping.

2.2.2 Pressure-Retaining Components and Component Supports

2.2.2.1 Regulatory Evaluation

The NRC staff has reviewed the structural integrity of pressure-retaining components (and their supports) designed in accordance with the ASME Code, Section III, Division 1, and draft GDC 1, 2, 9, 33, 40, and 42. The NRC staff's review focused on the effects of the proposed EPU on the design input parameters and the design-basis loads and load combinations for normal operating, upset, emergency, and faulted conditions. The NRC staff's review covered (1) the analyses of FIV; and (2) the analytical methodologies, assumptions, ASME Code editions, and computer programs used for these analyses. The NRC staff's review also included a comparison of the resulting stresses and fatigue cumulative usage factors (CUFs) against the Code-allowable limits. The NRC's acceptance criteria are based on (1) 10 CFR 50.55a and draft GDC-1, insofar as they require that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) draft GDC-2, insofar as it requires that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of a LOCA; (4) draft GDC-9 and 33, insofar as they require that the RCPB be designed and constructed so as to have an exceedingly low probability of RCPB gross rupture or significant leakage; and (5) draft GDC-34 insofar as it requires that the RCPB be designed to minimize the probability of rapidly propagating type failures. Specific review criteria are contained in SRP Sections 3.9.1, 3.9.2, 3.9.3, and 5.2.1.1; and other guidance provided in Matrix 2 of RS-001.

2.2.2.2 Technical Evaluation

2.2.2.2.1 Nuclear Steam Supply System Piping, Components, and Supports

The RCPB piping system consists of a number of safety-related piping subsystems that move fluid through the reactor and other safety systems. The licensee evaluated the effects of EPU

conditions, including higher flow rate, temperature, pressure, fluid transients and vibration effects on the RCPB and Balance-of-Plant (BOP) piping systems and components. The components evaluated included equipment nozzles, anchors, guides, penetrations, pumps, valves, flange connections, and pipe supports (including snubbers, hangers, and struts). The licensee indicated that the original Code of record as referenced in the original and existing design basis analyses was used in the evaluation. The NRC staff finds this acceptable.

The RCPB piping systems evaluated by the licensee include the reactor recirculation, MS, MS drains, reactor core isolation cooling (RCIC), high-pressure coolant injection (HPCI), FW, reactor water cleanup (RWCU), core spray (CS), standby liquid control (SLC), residual heat removal (RHR), reactor pressure vessel (RPV) head vent line, control rod drive (CRD) piping, and SRV discharge line systems. The evaluation used the United States of America Standards (USAS) B31.1, "Power Piping" (1967 Edition), which is the VYNPS Code of record. The licensee indicated that the evaluation follows the process and methodology defined in Appendix K of General Electric Licensing Topical Report (LTR) NEDC-32424P-A (February 1999), "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," (ELTR1) and in Section 4.8 of Supplement 1 of GE Licensing Topical Report, NEDC-32523P-A (February 2000), "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate" (ELTR2). In general, the licensee compared the increase in pressure, temperature and flow rate due to the EPU against the same parameters used as input to the original design-basis analyses. The comparison resulted in bounding percentage increases in stress for affected limiting piping systems. The bounding percentage increases are compared to the design margin between calculated stresses and the Code allowable limits. The bounding percentage increases were also applied to the original calculated stresses for the piping to determine the stresses at the proposed EPU conditions. The NRC staff finds the methodology to be acceptable considering the conservatism in the application of the scaling factors for the EPU stress to loading combinations that include individual loads (i.e., dead weight and seismic loads) that are not affected by the EPU.

At EPU conditions, the flow, pressure, temperature, and mechanical loading for the RCPB piping systems either remain the same or change insignificantly (i.e, within the safety margin) except the MS and FW lines. The licensee evaluated the MS and FW lines and associated branch piping systems in accordance with the requirements of USAS-B31.1 (1967 Edition) for the effects of the EPU on piping, piping supports including the associated building structure, piping interfaces with the RPV nozzles, penetrations, flanges and valves. In Supplement 8 (July 2, 2004) of its EPU request, the licensee provided maximum calculated stresses for the limiting FW and MS piping systems at VYNPS for the proposed EPU. The maximum stresses shown in the tables are less than the Code allowable limits for both the FW and MS piping systems.

At VYNPS, the MS piping pressures and temperatures are not affected by the CPPU. [[ ]] There is no effect on the analyses for these load conditions. The licensee performed a bounding piping analysis including the effects of EPU conditions. The increase in MS flow results in increased forces from the turbine stop valve closure transient. The turbine stop valve closure loads bound the main steam isolation valve (MSIV) closure loads because the MSIV closure time is significantly longer than the stop valve closure time.

The licensee evaluated piping supports such as snubbers, hangers, struts, anchorages, equipment nozzles, guides, and penetrations by evaluating the piping interface loads due to the increases in pressure, temperature, and flow for affected limiting piping systems. The evaluation shows that there is adequate design margin between the original design stresses and Code limits for the supports to accommodate the load increase, with the exception of two pipe supports, MSH-6 (MS-35) and RMSH-14 (MS-6), that were identified as requiring modification for operation at EPU conditions. Both modifications consisted of replacing the existing pipe clamp with a new pipe clamp. Both of these modifications were implemented during the April 2004 refueling outage (RFO 24). These pipe support modifications do not affect the piping configuration; therefore, there are no piping configuration changes to reflect in the EPU analysis.

Piping systems other than FW and MS lines connecting to the RCPB do not experience an increase in flow rate at EPU conditions. The normal operating pressure and temperature of the reactor are not changed by the EPU. The licensee evaluated these systems by reviewing the original design basis analysis of record. The review shows that there is adequate design margin between the original design stress and the Code limits. Therefore, these piping systems continue to comply with the USAS B31.1 Code and are acceptable to operate at EPU conditions.

In Supplement 8 (July 2, 2004) of its EPU request, the licensee indicated that the piping vibration monitoring program for VYNPS addresses FIV of the critical piping systems that will experience increased flow during EPU operating conditions. The piping steady state vibration program for EPU operation follows the guidelines of ASME OM-S/G-2000 Code, Part 3, "Requirements for Preoperational and Initial Startup Vibration Testing of Nuclear Power Piping Systems." The program will assess the flow-induced steady state vibration levels of selected piping systems that will experience increased flow during EPU operating conditions. The program will include branch lines and cantilevered small bore lines which industry experience has shown are vulnerable to high-cycle fatigue failures.

On the basis of its evaluation, the licensee concluded that, for all RCPB piping systems, the original piping design has sufficient design margin to accommodate the changes due to the proposed EPU. The NRC staff reviewed relevant portions of the evaluation provided by the licensee and finds the licensee's conclusion to be acceptable.

2.2.2.2.2 Balance-of-Plant Piping, Components, and Supports

The licensee's evaluations of the stresses for BOP piping and related components, connections and supports are similar to the evaluation of the RCPB piping and supports. The licensee indicated that the original Code of record as referenced in the pertinent calculations, and Code allowables, were used; and that no new assumptions were introduced. The BOP systems evaluated by the licensee include lines which are affected by the EPU, but not evaluated in Section 3.5 of the PUSAR, such as MS (outside containment), Extraction Steam (ES), Heater Vents and Drains, FW and Condensate, RWCU - Outside Containment, RHR - Outside Containment, RHR Service Water - Outside Containment, CS - Outside Containment - Pump Suction / Pump Discharge, HPCI - Outside Containment, RCIC - Outside Containment, Standby Liquid Control System (SLCS) - Outside Containment, CRD, Service Water, Reactor Building Closed Cooling Water, Turbine Building Closed Cooling Water, Spent Fuel Cooling, Standby Gas Treatment, Off Gas, and Torus Attached Piping including Emergency Core Cooling System

(ECCS) Suction Strainers. The existing design analyses of the affected BOP piping systems were reviewed against the uprated power conditions. As a result of its evaluation, the licensee concluded that there are sufficient margins in the original design analyses to accommodate the changes due to the proposed EPU and, therefore, all piping meets the requirements of USAS B31.1-1967, which is the Code of record.

The calculated maximum stresses and allowable stress limits for the critical BOP piping systems are provided in Tables 3-8a, 3-8b, and 3-8c of the PUSAR. Table 3-8a provides the maximum stresses and allowable stress limits for FW, extraction steam, FW heater vents and drains, and condensate piping systems. Table 3-8b provides the maximum stresses and allowable stress limits for torus attached piping systems. Table 3-8c provides the maximum stresses and allowable stress limits for the MS system. Fatigue usage factors are not included in the VYNPS design basis for BOP piping evaluations. The NRC staff finds that the stresses provided by the licensee for the most limiting piping systems mentioned above for the EPU conditions are within the Code-allowable limits and are, therefore, acceptable.
The licensee evaluated piping supports such as snubbers, hangers, struts, anchorages, equipment nozzles, guides, and penetrations by evaluating the piping interface loads due to the increases in pressure, temperature, and flow for affected limiting piping systems. The piping supports of the systems affected by EPU loading increases (MS, RHR, CS, HPCI, RCIC, FW, and ES) were reviewed. This review shows that there is adequate design margin between the original design stresses and Code limits of the supports to accommodate the load increase, with the exception of support RCIC-HD63C that was identified as requiring modification. The licensee indicated that the modification will be completed prior to EPU implementation.

The licensee evaluated the FIV levels of the safety-related MS and FW piping systems that are projected to increase in proportion to the increase in the fluid density and the square of the fluid velocity following the proposed EPU. The NRC staff's evaluation of the licensee's FIV program is provided in SE Section 2.2.6.

Regarding the assessment of the MS flow restrictor, the licensee stated that there is no impact on the structural integrity of the restrictor as a result of the proposed EPU. In Section 3.1 of the PUSAR, the licensee indicated that a peak RPV dome pressure of 1304 psig results from the overpressure protection event analysis for the proposed VYNPS EPU conditions, but this value remains below the ASME Code limit of 1375 psig (110% of design pressure). Also, the restrictor was designed for a maximum differential pressure due to the choke flow condition which is bounding for the uprated power condition. Therefore, the MS line flow restrictor will maintain its structural integrity following the EPU. The licensee evaluated the MSIVs by referring to the GE [[ ]] evaluation in Section 4.7 of NEDC-32523P-A, which was [[ ]] The licensee stated in PUSAR Section 3.8 that the increase in MS flow rate assists MSIV closure, which results in a slightly faster closure time. The licensee concluded that the MSIVs are acceptable for EPU operation. The NRC staff finds the licensee's conclusion to be reasonable and acceptable.

Based on the above review, the NRC staff concurs with the licensee's conclusion that the design of BOP piping, components and their supports is adequate to ensure that the BOP system will maintain its structural and pressure boundary integrity at the proposed EPU conditions.

2.2.2.2.3 Reactor Vessel and Supports

The licensee evaluated the effects of the VYNPS EPU on the RPV and internal components in accordance with its current design basis. The loads considered in the evaluation include reactor internal pressure difference (RIPD), LOCA, flow loads, acoustic loads, thermal loads, seismic loads, and dead weight. The licensee indicated that the load combinations for normal, upset, emergency, and faulted conditions were considered consistent with the current design basis analysis. In its evaluation, the licensee compared the proposed EPU conditions (pressure, temperature and flow) against those used in the design basis. For cases where the EPU conditions are bounded by the design basis analyses, no further evaluation is performed. If the EPU conditions are not bounded by the design basis, new stresses are determined by [[ ]] The resulting stresses are compared against the applicable allowable values, in accordance with the design basis. The NRC staff finds the methodology used by the licensee to be consistent with the NRC-approved methodology in Appendix I of NEDC-32424P-A, and is, therefore, acceptable.

The stresses and CUFs for the RPV components were evaluated by the licensee in accordance with the ASME Code, Section III, 1965 Edition with addenda up to and including Summer 1966, which is the Code of record at VYNPS.

The CPPU methodology demonstrates that, for most RPV components, temperature, fluid flow, RIPDs, and other mechanical loads do not increase. Consequently, there is no change in stress or fatigue usage for these components. One exception, however, is the FW nozzle which experiences an increase in FW flow and temperature, resulting in an increase in stress and fatigue. In response to the NRC staff's RAI question EMEB-B-149, the licensee, in its letter dated August 4, 2005, explained that the primary reason for the relatively large increase in stress and CUF for the FW nozzle, compared to the other RPV components reported in Table 3-3 of the PUSAR, is the increase in FW flow from 3720 gpm to 4705 gpm associated with the EPU conditions for this nozzle. This increase in flow is the basis for the development of a scaling factor used to recalculate the stresses in the RPV nozzle penetration in accordance with the method described in Appendix I of NEDC-32424P-A, which has been previously approved by the NRC staff in a letter dated February 8, 1996. The scaling factor was conservatively applied to both the pressure and thermal stresses, even though RPV pressure stresses are not increased by the CPPU process. The recalculated thermal and pressure stresses were then combined with the maximum applied mechanical loadings from the associated FW piping to arrive at a conservative total nozzle stress value. The resulting total FW nozzle stress and CUF for the CPPU conditions reported in PUSAR Table 3-3 represent the limiting stress condition for the RPV pressure boundary, and are within the allowable ASME Code limits. Due to the conservative method used to recalculate these resultant stress and fatigue usage values, the actual design margin with respect to the ASME Code limits for the EPU limiting RPV pressure boundary stress condition is conservatively underestimated.

The RPV components that are not listed in Table 3-3 of the PUSAR [[

]] This methodology has been previously approved by the NRC staff and is acceptable.

The primary plus secondary stresses and the CUF results presented in Table 3-3 of the PUSAR demonstrate that the RPV pressure boundary and pressure boundary penetrations, including the FW nozzle penetration, are less than the ASME Code, Section III stress and fatigue usage factor allowable values. The maximum stresses for critical components of the RPV internals were summarized in Table 3-7 of the PUSAR for the currently licensed power level and the proposed EPU conditions. These calculated stresses are also less than the allowable Code limits. The NRC staff's evaluation of the structural integrity of the RPV internals is discussed in SE Section 2.2.3.

Based on its review of the licensee's evaluation of the reactor vessel and internals, the NRC staff finds that the maximum stresses and fatigue usage factors are within the Code-allowable limits. The staff also concurs with the licensee's conclusion that the reactor vessel and internals will continue to maintain their structural integrity for the EPU conditions.

2.2.2.2.4 Control Rod Drive Mechanism

In PUSAR Section 3.2.2 and Table 3-7, the licensee indicated that the pressure boundary components of the control rod drive (CRD) system have been designed in accordance with the Code of record, the ASME Code, Section III,1965 Edition up to and including the Summer 1966 addenda, and that the original design basis analysis for the CRD system remains unchanged by EPU conditions. The components of the CRD system which form part of the primary pressure boundary have been designed for a bottom head pressure of 1250 psig, which is higher than the analytical limit of 1131 psig for the reactor bottom head pressure.

In response to the NRC staff's RAI question EMEB-B-16, the licensee stated, in its letter dated July 2, 2004, that the design pressure, seismic loads, and fuel lift loads remain the same as those for the CLTP level. Therefore, the EPU primary plus secondary stresses and the CUFs for the CRD housing remain the same as those calculated for the CLTP level, which are less than the ASME Code, Section III allowable values.

On the basis of its review, the NRC staff concurs with the licensee's conclusion that the CRD mechanism will continue to meet its design basis and performance requirements at the proposed EPU conditions.

2.2.2.2.5 Recirculation Pumps and Supports

In response to the NRC staff's RAI question EMEB-B-15, the licensee stated in its letter dated July 2, 2004, that at rated core flow, the required recirculation pump flow will increase by 553 gpm (1.7% of rated pump flow) for EPU conditions. The licensee reviewed the stress and fatigue calculation in the analysis of record for the current design basis of the recirculation piping and pumps, and determined that there is sufficient margin to the Code allowable limits to accommodate the 1.7% increase in pump flow rate. Consequently, the licensee concluded that the EPU conditions are within the original design capability of the system equipment including the pump, valves, piping and supports. Based on its review, the NRC staff concurs with the licensee's conclusion that the current design of the recirculation piping system (including pumps and supports) is adequate to operate at EPU conditions.

2.2.2.3 Conclusion

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of pressure-retaining components and their supports at VYNPS. For the reasons set forth above, the staff concludes that the licensee has adequately addressed the effects of the proposed EPU on these components and their supports. Based on the above, the staff further concludes that the licensee has demonstrated that pressure-retaining components and their supports will continue to meet the requirements of 10 CFR 50.55a, and draft GDC-1, 2, 9, 33, 34, 40, and 42, following implementation of the proposed EPU. Therefore, the staff finds the proposed EPU acceptable with respect to the structural integrity of the pressure-retaining components and their supports.

2.2.3 Reactor Pressure Vessel Internals and Core Supports

2.2.3.1 Regulatory Evaluation

Reactor pressure vessel internals consist of all the structural and mechanical elements inside the reactor vessel, including core support structures. The NRC staff reviewed the effects of the proposed EPU on the design input parameters and the design-basis loads and load combinations for the reactor internals for normal operation, upset, emergency, and faulted conditions. These include pressure differences and thermal effects for normal operation, transient pressure loads associated with LOCAs, and the identification of design transient occurrences. The staff's review covered (1) the analyses of FIV for safety-related and non-safety-related reactor internal components (except the steam dryer which is reviewed in Section 2.2.6 of this SE); and (2) the analytical methodologies, assumptions, ASME Code editions, and computer programs used for these analyses. The staff's review also included a comparison of the resulting stresses and CUFs against the corresponding Code-allowable limits. The NRC's acceptance criteria are based on (1) 10 CFR 50.55a and draft GDC-1, insofar as they require that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) draft GDC-2, insofar as it requires that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a LOCA; and (4) draft GDC-6, insofar as it requires that the reactor core be designed with appropriate margin to assure that acceptable fuel damage limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Specific review criteria are contained in SRP Sections 3.9.1, 3.9.2, 3.9.3, and 3.9.5; and other guidance provided in Matrix 2 of RS-001.

2.2.3.2 Technical Evaluation

The licensee evaluated the effects of the VYNPS EPU on the reactor vessel and internal components in accordance with its current design basis. The loads considered in the evaluation include RIPD, LOCA, flow loads, acoustic loads, thermal loads, seismic loads, and dead weight. The licensee indicated that the load combinations for normal, upset, emergency,

and faulted conditions were considered consistent with the current design basis analysis. In its evaluation, the licensee compared the proposed EPU conditions (pressure, temperature and flow) against those used in the design basis. For cases where the EPU conditions are bounded by the design basis analyses, no further evaluation is performed. If the EPU conditions are not bounded by the design basis, new stresses are determined by scaling up the existing design basis stresses proportionate to the proposed EPU conditions. The resulting stresses are compared against the applicable allowable values, in accordance with the design basis. The NRC staff finds the methodology used by the licensee is consistent with the NRC-approved methodology in Appendix I of NEDC-32424P-A, and is therefore acceptable.

The stresses and CUFs for the reactor vessel components were evaluated by the licensee in accordance with the ASME Code, Section III, 1965 Edition with addenda up to and including Summer 1966, which is the Code of record at VYNPS. The licensee indicated that, for VYNPS, the reactor internal components are not ASME Code components. However, as indicated in the PUSAR, the requirements in ASME Code, Section III, 1986 Edition, Subsection NB, have been used as guidelines in the design-basis documents for the EPU stress analysis of reactor internal components at VYNPS. The licensee also indicated that the evaluations supporting the thermal power increase were performed consistent with the VYNPS design basis. The NRC staff finds that this provides an acceptable basis for analysis.

The licensee provided the calculated maximum stresses and CUFs for the most limiting reactor vessel components in Table 3-3 of the PUSAR. The reactor vessel components that are not listed in Table 3-3 have maximum stresses and CUFs that are either not affected by the EPU or already bounded by those listed in the table. The maximum calculated stresses in Table 3-3 are within the Code-allowable limits, and the CUFs are less than the Code limit of unity. The maximum stresses for critical components of the reactor internals were summarized in Table 3-7 of the PUSAR for the current rated and the proposed EPU conditions. These calculated stresses are less than the allowable Code limits.

In its assessment of the potential for FIV on the reactor internals components, the licensee indicated that the steam separators and dryers in the upper elevations of the reactor are the components most affected by the increased steam flow due to the proposed EPU. The effects of the EPU on the FIV for other components in the reactor annulus and core regions are less significant, because the proposed EPU conditions do not require any increase in core flow, and very little increase in the drive flow. For components other than the steam separators and dryers, the evaluation of FIV for the reactor internal components was performed based on the vibration data recorded during startup testing at the GE prototype BWR/4 plant (Monticello) and VYNPS. The vibration levels were calculated by extrapolating the recorded vibration data to EPU conditions and compared to the plant allowable limits. The stresses at critical locations were calculated based on the extrapolated vibration peak response displacements and found to be within the GE allowable design criteria of 10 ksi (where 1 ksi = 1000 pounds per square inch). Stress values less than 10 ksi for stainless steel are within the endurance limit under which sustained operation is allowed without incurring any cumulative fatigue usage. The vibration evaluation methodology, as described in Section 3.4.2 of the PUSAR, is conservative based upon the absolute sum combination of the various modes of vibration, including the absolute sum of the maximum vibration amplitude occurring in each mode. The licensee concluded that vibration levels of all safety-related reactor internal components are within the acceptance criteria. The NRC staff finds the licensee's specified stress limit of 10 ksi for the

reactor internal components to be reasonably conservative in comparison to the ASME Code limit of 13.6 ksi for the peak vibration stress and is, therefore, acceptable.

PUSAR Section 3.4.2 states that the steam separators and steam dryer are not safety-related components; however, their failure may lead to an operational concern. The licensee evaluated the steam separators together with the shroud head, and determined the loads and stresses to be below the allowable values for normal, upset, emergency, and faulted conditions during EPU operation. The PUSAR states that FIV influence on RPV internals, including the steam separators, was evaluated based on extrapolation of vibration data recorded at similar plants and on BWR operating experience. On the basis of information provided by the licensee, the NRC staff concludes that the licensee has reasonably demonstrated that the steam separators will meet their design basis requirements and maintain their structural integrity under EPU conditions.

Based on its review of the licensee's evaluation of the reactor vessel and internals, the NRC staff finds that the maximum stresses and fatigue usage factors are within the Code-allowable limits. The staff also concurs with the licensee's conclusion that the reactor vessel and internals will continue to maintain their structural integrity for the EPU conditions.

2.2.3.3 Conclusion

The NRC staff has reviewed the licensee's evaluations related to the structural integrity of reactor vessel internals and core supports and concludes that the licensee has adequately addressed the effects of the proposed EPU on the reactor vessel internals and core supports. The NRC staff further concludes that the licensee has demonstrated that the reactor vessel internals and core supports will continue to meet the requirements of 10 CFR 50.55a, and draft GDC-1, 2, 6, 40, and 42, following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the design of the reactor vessel internals and core supports.

2.2.4 Safety-Related Valves and Pumps

2.2.4.1 Regulatory Evaluation

The NRC staff's review included certain safety-related pumps and valves typically designated as Class 1, 2, or 3 under Section III of the ASME Code and within the scope of Section XI of the ASME Code and the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (O&M Code), as applicable. The NRC staff's review focused on the effects of the proposed EPU on the required functional performance of the valves and pumps at VYNPS. The review also covered any impacts that the proposed EPU may have on the licensee's motor-operated valve (MOV) programs related to Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance;" GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves;" and GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves." The NRC staff also evaluated the licensee's consideration of lessons learned from the MOV program and the application of those lessons learned to other safety-related power-operated valves. The NRC's acceptance criteria are based on (1) draft GDC-1, insofar as it requires those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed, fabricated, erected, and tested to quality

standards commensurate with the importance of the safety functions to be performed; (2) draft GDC-38, 46, 47, 48, 59, 60, 61, 63, 64, and 65 insofar as they require that the ECCS, the containment heat removal system, the containment atmospheric cleanup systems, and the cooling water system, respectively, be designed to permit appropriate periodic testing to ensure the leak-tight integrity and performance of their active components; (3) draft GDC-57, insofar as it requires that piping systems penetrating containment be designed with the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits; and (4) 10 CFR 50.55a(f), insofar as it requires that pumps and valves subject to that section must meet the inservice testing program requirements identified in that section. Specific review criteria are contained in SRP Sections 3.9.3 and 3.9.6.

2.2.4.2 Technical Evaluation

In Attachment 4 to its application dated September 10, 2003, the licensee discussed its evaluation of the capability of safety-related valves and pumps to perform their safety functions under EPU conditions. In Attachment 1 to Supplement 5 (January 31, 2004) of its EPU request, the licensee responded to an NRC staff RAI on its evaluation of valves and pumps at VYNPS for EPU operations. The NRC staff has reviewed the licensee's evaluation of the impact of EPU conditions on safety-related valves and pumps at VYNPS. This review is summarized in the following paragraphs.

In NRC Inspection Report (IR) 50-271/97-08, dated November 28, 1997, the NRC staff closed its review of the licensee's program to demonstrate the design-basis capability of safety-related MOVs within the scope of GL 89-10 to perform their safety functions. In an SE dated December 14, 2000, the NRC staff determined that the licensee had established an acceptable program to periodically verify the design-basis capability of the safety-related MOVs at VYNPS through the actions described in its submittals. In Attachment 4 to its request for the EPU license amendment (September 10, 2003), the licensee indicated that it had evaluated the MOVs within the scope of GL 89-10 at VYNPS for the effects of the proposed EPU, including those related to pressure locking and thermal binding as addressed in GL 95-07. The licensee did not identify any changes to the design functional requirements for the GL 89-10 MOVs, but did note that minor process fluid condition changes and increased ambient room temperatures would occur for some MOVs as a result of EPU operation at VYNPS. In Attachment 1 of Supplement 5 of its EPU request, the licensee reported that it had screened the MOVs at VYNPS for impact from EPU conditions, and provided examples of that process. The licensee also indicated that the VYNPS MOVs had been evaluated for pressure locking and thermal binding under EPU conditions, and that no new MOVs were determined to be susceptible to pressure locking or thermal binding. Of the MOVs previously identified as being potentially susceptible to pressure locking or thermal binding, the licensee stated that only one MOV (RHR Drywell Spray Valve V10-26A) was calculated to experience an increase in accident condition environmental temperature (5°F) and that any setpoint adjustment would be made as necessary.

During an NRC engineering inspection conducted in August and September 2004, the NRC staff identified several weaknesses in the MOV program at VYNPS with respect to implementation of the program as described in the licensee's submittals in response to GL 96-05. As described in NRC Inspection Report 05000271/2004008, dated December 2, 2004, the NRC staff found that the licensee had not validated the adequacy of the diagnostic test instrumentation to be used from the motor control center (MCC) with respect to its ability to

detect MOV actuator torque and stem thrust degradation. The NRC staff also determined that plant procedures to be used to obtain and evaluate MCC diagnostic test data did not include specific acceptance criteria tied to MOV stem thrust or available design margin. The staff noted that the MOV program had been revised in 2002 to eliminate the periodic requirements for "at-the-valve" diagnostic testing that can measure torque and thrust to known accuracies. Although the staff did not identify any examples of degraded or inoperable valves during the inspection, the staff concluded that these weaknesses in the MOV program constituted a non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control." The staff also observed that the licensee had not maintained a current risk ranking of its MOVs at VYNPS, and that the informal MOV trending methods at VYNPS did not include DC-powered MOVs and might not identify slow acting degradation from the baseline value. In Supplement 16 (September 30, 2004) of its EPU request, the licensee committed to revise the MOV Periodic Verification Program at VYNPS to include periodic at-the-valve testing and to formalize the process for DC motor trending by December 1, 2004. In Attachment 3 to Supplement 32 (September 10, 2005) of its EPU request, the licensee reported that the commitment to revise the MOV Periodic Verification Program to include periodic at-the-valve testing as a means to verify the effectiveness of the MCC testing methodology and to formalize the process for trending DC motor performance had been completed. The NRC staff verified the implementation of the licensee's actions to resolve the findings in IR 05000271/2004008 through NRC region inspection activity as documented in IR 05000271/2005006.

In Attachment 4 to its EPU license amendment request (September 10, 2003), the licensee reported that air-operated valves (AOVs) at VYNPS were reviewed to identify any AOVs potentially affected by EPU conditions. In Attachment 1 to Supplement 5 of its EPU request, the licensee reported that AOV parameters that could be adversely affected by EPU operation (such as increases in operating differential pressure and shut-off differential pressure) had been evaluated. The licensee determined the need to evaluate a potential increase in the inlet pressure, operating and shutoff differential pressure for the high pressure FW heater drain valves and the moisture separator drain tank control valves, without anticipation of equipment modifications.

In Attachment 4 to its EPU license amendment request (September 10, 2003), the licensee reported that the increase in steam flow under EPU conditions would assist in the closure of the MSIVs at VYNPS. In Attachment 1 to Supplement 5 of its EPU request, the licensee described the self-compensating feature of the hydraulic control valve that will maintain closing time with little deviation despite the change in steam flow. The licensee indicated that other nuclear plants with this type of MSIV have implemented power uprates without reported anomalies in MSIV closing time.

In Attachment 4 to its EPU license amendment request (September 10, 2003), the licensee stated that the nuclear system pressure relief system at VYNPS prevents overpressurization of the nuclear system by means of safety relief valves and spring safety valves. The licensee determined that no increases to the setpoints for the safety relief valves or spring safety valves were necessary for EPU conditions at VYNPS because of the absence of changes in the reactor dome pressure or simmer margin. The licensee noted that the potential for increased incidents of valve leakage, or inadvertent opening of a safety relief valve, as a result of FIV during EPU operation is addressed by plant procedures. Potential adverse effects from FIV of safety relief valves and spring safety valves are also addressed as part of the piping vibration program at VYNPS discussed in Section 2.2.6 of this SE.

In Attachment 4 to its EPU license amendment request (September 10, 2003), the licensee stated that the performance of the ECCS at VYNPS will remain acceptable under EPU conditions. In Attachment 1 to Supplement 5 of its EPU request, the licensee described its evaluation of safety-related pumps in the ECCS to determine that their performance will meet the EPU requirements without any modifications or procedural changes. The licensee indicated that the pump seals of the RHR and CS pumps have been requalified for the increased suppression pool temperature under accident conditions.

2.2.4.3 Conclusion

The NRC staff has reviewed the licensee's assessments related to the functional performance of safety-related valves and pumps at VYNPS in support of the EPU license amendment request. The staff concludes that the licensee has adequately addressed the effects of the proposed EPU on safety-related pumps and valves at VYNPS. Based on this, the NRC staff concludes that the licensee has demonstrated that safety-related valves and pumps at VYNPS will continue to meet the requirements of draft GDC-1, 38, 46, 47, 48, 57, 59, 60, 61, 63, 64, and 65, and 10 CFR 50.55a(f) following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU for VYNPS to be acceptable with respect to safety-related valves and pumps.

2.2.5 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

2.2.5.1 Regulatory Evaluation

Mechanical and electrical equipment covered by this section includes equipment associated with systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal. Equipment associated with systems essential to preventing significant releases of radioactive materials to the environment are also covered by this section. The NRC staff's review focused on the effects of the proposed EPU on the qualification of the equipment to withstand seismic events and the dynamic effects associated with pipe-whip and jet impingement forces. The primary input motions due to the safe shutdown earthquake (SSE) are not affected by an EPU. The NRC's acceptance criteria are based on (1) draft GDC-1, insofar as it requires that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed; (2) draft GDC-2, insofar as it requires that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a LOCA; (4) draft GDC-9 and 33, insofar as they require that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage; and (5) draft GDC-34 insofar as it requires that the RCPB be designed to minimize the probability of rapidly propagating type failures. Specific review criteria are contained in SRP Section 3.10.

2.2.5.2 Technical Evaluation

The licensee evaluated equipment qualification for EPU conditions. The VYNPS plant-specific dynamic loads such as SRV discharge and LOCA loads (including pool swell, condensation oscillation, and chugging loads) that were used in the equipment design will remain unchanged as discussed in Section 4.1.2 of the PUSAR, since these loads are based on the range of test conditions for the design-basis analysis at VYNPS, which are bounding for EPU conditions.

Based on its review of the proposed EPU amendment, the NRC staff finds that the original seismic and dynamic qualification of safety-related mechanical and electrical equipment is not affected by the EPU conditions for the following reasons:

- The seismic loads are unaffected by the EPU;
- No new pipe break locations or pipe whip and jet impingement targets are postulated as a result of the EPU;
- Pipe whip and jet impingement loads do not increase for the EPU; and
- SRV and LOCA dynamic loads used in the original design basis analyses are bounding for the EPU.

2.2.5.3 Conclusion

The NRC staff has reviewed the licensee's evaluations of the effects of the proposed EPU on the qualification of mechanical and electrical equipment and concludes that the licensee has (1) adequately addressed the effects of the proposed EPU on this equipment; and (2) demonstrated that the equipment will continue to meet the requirements of draft GDC-1, 2, 9, 33, 34, 40, and 42, following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the qualification of the mechanical and electrical equipment.

2.2.6 Additional Review Area - Potential Adverse Flow Effects

2.2.6.1 Regulatory Evaluation

Plant operation at EPU conditions can result in adverse flow effects on the MS, FW, and condensate systems and their components (including the steam dryer in BWR plants) from increased system flow and FIV. Some plant components, such as the steam dryer, do not perform a safety function, but must retain their structural integrity to avoid the generation of loose parts that might adversely impact the capability of other plant equipment to perform their safety functions. The NRC staff reviewed the licensee's consideration of potential adverse flow effects of the proposed EPU at VYNPS, including consideration of the design input parameters and the design-basis loads and load combinations for the VYNPS steam dryer for normal operation, upset, emergency, and faulted conditions. The NRC staff's review covered the analytical methodologies, assumptions, and computer programs used in the evaluation of the VYNPS steam dryer. The NRC staff's review also included a comparison of the resulting stresses against applicable limits. The NRC staff also reviewed the licensee's evaluation of other reactor, MS, FW, and condensate system components at VYNPS for potential

susceptibility to adverse flow effects from EPU operation. The NRC's acceptance criteria are based on (1) draft GDC-1, insofar as it requires those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed, fabricated, erected, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed; (2) draft GDC-2, insofar as it requires that those systems and components which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences be designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; and (3) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a LOCA. Specific review criteria are contained in SRP Sections 3.9.1, 3.9.2, 3.9.3, and 3.9.5.

2.2.6.2 Technical Evaluation

2.2.6.2.1 Steam Dryer

As indicated in Attachment 5 to Supplement 26 (March 31, 2005) of its EPU request, the licensee originally procured the steam dryer for VYNPS as a non-safety-related, non-Seismic I, non-ASME component. In response to damage experienced by steam dryers at other nuclear power plants under EPU conditions, Entergy modified the square-hood steam dryer at VYNPS to improve its capability to withstand potential adverse flow effects that could result from operation of the plant at EPU conditions. In Supplement 8 (July 2, 2004) of its EPU request, the licensee described the modifications to the VYNPS steam dryer as follows:

- outer vertical hood plates (61-inch high) on 90° and 270° sides replaced with 1-inch thick plate;
- 3 reinforcing gussets (55.5-inch high) welded to outer vertical hood plates and lower horizontal cover plates on 90° and 270° sides;
- lower horizontal cover plates on 90° and 270° sides replaced with 5/8-inch thick plate;
- 15-inch section of upper horizontal cover plates on 90° and 270° sides at intersection of outer vertical hood plates replaced with 1-inch thick plate;
- internal bracing brackets at outer vertical hood plates removed; and
- dryer bank tie bars replaced with new design.

During a technical audit at the GE office in San Jose, CA, from August 24 to 26, 2004, NRC staff members from the Office of Nuclear Reactor Regulation (NRR) and Office of Nuclear Regulatory Research (RES) with technical assistance by contractors from the Argonne National Laboratory reviewed the VYNPS steam dryer analysis initially provided as part of the licensee's EPU request. As discussed in the audit report dated September 14, 2004, the NRC staff concluded that the licensee's analysis was inadequate to demonstrate that the steam dryer at the VYNPS will be capable of maintaining its structural integrity under EPU conditions. For example, the licensee's analysis of the steam dryer as then submitted in support of its EPU request (1) had not adequately identified and verified the excitation sources for FIV

mechanisms that resulted in significant degradation of similar steam dryers at other BWR nuclear power plants operating at EPU conditions; (2) had not provided a technically justifiable load definition for the steam dryer for EPU conditions in light of several assumptions that had not been adequately justified; (3) had not justified the applied methodology as realistic in light of assumptions to account for uncertainties that resulted in apparent significant overestimation of predicted steam dryer stresses; (4) might be non-conservative based on assumptions for reducing the stress experienced by steam dryer parts and the creation of new potential fatigue failure locations as a result of modifications to the VYNPS steam dryer; and (5) had not validated the extrapolation of pressure peaks from original power levels to EPU conditions for the steam dryer at VYNPS. In the audit report, the NRC staff indicated that the licensee could submit a revised analysis of the steam dryer in support of its request to operate VYNPS at EPU conditions.

In Supplement 26 (March 31, 2005), Supplement 27 (April 5, 2005), and Supplement 29 (June 2, 2005) of its EPU request, Entergy provided a revised analysis of the capability of the modified VYNPS steam dryer to maintain its structural integrity under EPU conditions. NRC staff members from NRR and RES have reviewed the revised VYNPS steam dryer analysis with technical assistance by contractors from the Argonne National Laboratory (including a consultant from the Pennsylvania State University), and McMaster University in Canada. On June 15 and 16, 2005, the NRC staff with its contractors conducted a technical audit of the revised analysis of the VYNPS steam dryer at the GE office in Washington, DC. On July 27, 2005, the NRC staff provided a final RAI to Entergy on the revised analysis of the VYNPS steam dryer. On August 1 and 4, 2005, the licensee submitted a response to the RAI in Supplements 30 and 31 to its EPU request. On August 15 and 16, 2005, an NRC staff member and a contractor conducted an audit at the GE Scale Model Test (SMT) facility near San Jose, CA, to obtain information on the licensee's performance of tests to validate the specific application of the acoustic circuit model (ACM) used by the licensee to determine the pressure loads on the VYNPS steam dryer during EPU operation. From August 22 to 25, 2005, the NRC staff with its contractors conducted a technical audit at the GE office in Washington, DC, of the revised analysis of the VYNPS steam dryer. In Supplement 33 (September 14, 2005) of its EPU request, the licensee provided revised RAI responses to address the NRC staff's findings from the August 22-25, 2005, audit. In Supplement 34 (September 18, 2005) of its EPU request, the licensee provided, among other information, several figures inadvertently omitted from Supplement 33. On December 5, 2005, the NRC staff conducted a follow-up audit to the June and August 2005 audits with licensee personnel at the Excel Corporation office in Rockville, MD, to verify appropriate finite element modeling of the connection of the gussets to the cover plate in the VYNPS steam dryer for the determination of stress at the connection under EPU conditions.

As described in the applicable supplements to its EPU request, the licensee evaluated the pressure loads acting on the steam dryer during operation of VYNPS through computational fluid dynamics (CFD) and acoustic circuit model (ACM) analyses. The licensee used the CFD analysis of the VYNPS steam dryer to predict hydrodynamic pressure loads that would act on the steam dryer at low frequencies under CLTP and EPU conditions. The licensee used the ACM analysis to calculate the acoustic pressure loads acting at high frequencies on the VYNPS steam dryer at CLTP based on pressure fluctuations in the MSLs measured by pressure sensors installed on the MSL venturi lines and strain gages installed on the MSLs. The licensee performed transient and static stress analyses using an ANSYS finite element model (FEM) of the VYNPS steam dryer. The licensee calculated the stresses on the VYNPS steam

dryer resulting from the CFD and ACM analyses, and combined those stresses by the square-root-of-the-sum-of-squares (SRSS) methodology with applicable weld concentration factors. The licensee then compared the peak alternating stresses for specific steam dryer locations to the fatigue limits in the ASME Code and the primary plus secondary stresses to the applicable ASME Code Service Level limits.

In its review of the VYNPS steam dryer analysis, the NRC staff evaluated the licensee's validation of its CFD and ACM analyses, and the uncertainty of those analyses and their inputs. The staff reviewed the licensee's fundamental frequency and damping assumptions for the VYNPS steam dryer. The staff evaluated the licensee's calculational methodology to convert the design pressure loads obtained from the CFD and ACM analyses to the stress at various locations on the steam dryer, the combination of the calculated CFD and ACM stresses, the stress limits used in evaluating steam dryer integrity, and the margins to those limits. The staff also reviewed the information provided by the licensee for monitoring the loads exerted on the steam dryer during plant operation and overall dryer performance.

In its CFD analysis, the licensee conducted a Large Eddy Simulation (LES) of the upper portion of the VYNPS reactor pressure vessel (including the steam dryer) and MSLs. The licensee determined pressure loads from low frequencies up into the acoustic range based on CFD analyses. Upon filtering the CFD analysis based on frequency, the licensee predicted stresses of low magnitude in the VYNPS steam dryer due to hydrodynamic loads having a frequency content of less than 30 Hz. In Attachment 5 to Supplement 33 of its EPU request, the licensee indicated that it used the full CFD predicted stress (and not the filtered CFD stress) in the evaluation of the combined stress and the limit curve factors. The licensee estimated the uncertainty of the CFD analysis as 15% based on a previous analysis of a small pipe flow model, and used measurements of low frequency pressure loads on steam dryers at four other nuclear power plants to support this uncertainty estimate. The NRC staff reviewed the CFD analysis (including the electronic data file) of the fluid dynamic loads on the VYNPS steam dryer. The NRC staff determined that significant uncertainty surrounds the CFD predictions, and that the magnitude of this uncertainty was highly underestimated by the licensee. For example, the licensee did not perform sensitivity studies of the CFD analysis applied to VYNPS to obtain an understanding of the significance of specific assumptions in the analysis. The comparison of the Vermont Yankee CFD results to the measured low frequency pressure loads at four other nuclear power plants does not establish the uncertainty value for the VYNPS CFD analysis, because CFD analyses were not performed for those other plants and all but one of those plants contained a steam dryer with an improved design to reduce hydrodynamic loads. The plant with a similar design steam dryer to VYNPS provided one pressure measurement that was in the skirt area with low flow conditions. Based on its review at that point, the staff determined that the uncertainty assumed by the licensee in its determination of the loads from the CFD analysis of the VYNPS steam dryer was significantly underestimated. To address this concern, and to confirm the licensee's predictions regarding the hydrodynamic and acoustic loads on the steam dryer, a license condition will be added to the VYNPS Facility Operating License as shown in SE Section 3.17.3. The license condition provides requirements for monitoring, evaluating, and taking prompt action in response to potential adverse flow effects as a result of operation at EPU conditions.

The licensee applied two different methods in its effort to validate the ACM used to calculate the acoustic pressure loads at high frequencies on the VYNPS steam dryer. In one method, the licensee used air tests conducted at the GE SMT facility to compare pressure loads calculated

by the ACM from steam line data to pressure measurements from a scale model steam dryer. In the second method, the licensee compared pressure sensor data collected from the instrumented steam dryer at the Quad Cities Unit 2 nuclear power plant during its power ascension to pressure loads calculated by the version of the ACM selected for application to the VYNPS steam dryer. The NRC staff determined that a number of uncertainties exist regarding the use of the SMT facility to validate the specific application of the ACM for the VYNPS steam dryer (including the relatively low flow provided by the SMT facility and the substantial deviation of the ACM predictions to SMT measurements). As a result, the staff focused on the licensee's use of the pressure sensor data obtained from the Quad Cities Unit 2 instrumented steam dryer to validate the ACM for application to VYNPS.

At VYNPS, the licensee applied a version of the ACM that was used by Exelon to assess the pressure loads on the steam dryer at Quad Cities Unit 2 at a power level of 790 megawatts electric (MWe) during EPU restart in May 2005. At Quad Cities Unit 2, Exelon revised the 790 MWe version of the ACM based on additional pressure sensor data collected from its instrumented steam dryer at 930 MWe. For VYNPS, Entergy developed an uncertainty estimate for the "790 MWe-version" of the ACM based on a comparison of the pressure loads calculated by the ACM to the measured pressure at 27 locations on the Quad Cities Unit 2 steam dryer. From its evaluation, the licensee estimated the uncertainty of the ACM as 100% of the calculated steam dryer pressure load. The NRC staff reviewed the licensee's estimation of the uncertainty of the version of the ACM used at VYNPS, and determined the 100% uncertainty value to be insufficient to provide reasonable assurance in the calculation of the pressure loads on the VYNPS steam dryer. For example, Figure EMEB-B-18-1-6 on page 16 in Attachment 1 to Supplement 33 of the licensee's EPU request indicates that the root mean square (RMS) of the pressure load calculated by the ACM, combined with the 100% uncertainty estimate, underpredicts the measured RMS pressure at many of the 27 pressure sensor locations on the Quad Cities Unit 2 steam dryer. Further, Figures EMEB-B-18-4-1 to 27 indicate that the power spectral density (PSD) from the ACM-calculated loads, combined with the 100% uncertainty estimate, underpredicts the PSD from the measured pressure data at Quad Cities Unit 2 over a wide frequency range for many of the 27 pressure sensors. As a result, the NRC staff considers the uncertainty assumed by the licensee for the version of the ACM applied at VYNPS to be significantly underestimated. To address this concern, and to confirm the licensee's predictions regarding the hydrodynamic and acoustic pressure loads on the steam dryer, a license condition will be added to the VYNPS Facility Operating License as shown in SE Section 3.17.3. The license condition provides requirements for monitoring, evaluating, and taking prompt action in response to potential adverse flow effects as a result of operation at EPU conditions.

At VYNPS, the licensee currently uses data from the MSL venturi instrument lines and one strain gage on each MSL to provide input to the ACM. The long venturi instrument lines and the lack of an array of strain gages at each MSL measurement location at VYNPS can result in significant uncertainty (over 100%) in the pressure input to the ACM. The NRC staff questioned the reliability of the ACM in calculating steam dryer pressure loads based on the large uncertainty associated with the MSL input data. To address the concerns with the uncertainty of the current MSL data used as input to the ACM, the licensee stated in the cover letter for Supplement 33 of its EPU request that it would install 32 additional strain gages on the MSL piping during the fall 2005 refueling outage (RFO) and would enhance the data acquisition system prior to EPU operation to reduce the measurement uncertainty associated with the ACM input. During the August 22-25, 2005, audit, the licensee indicated that the 32 additional strain

gages would be installed as a set of four strain gages in a quadrant array at two locations on each MSL to provide 8 independent inputs to the VYNPS ACM.

In Attachment 5 to Supplement 26 of its EPU request, the licensee described its structural analysis of the VYNPS steam dryer for CFD and ACM pressure loads at CLTP conditions. In Attachment 5 to Supplement 33 of its EPU request, the licensee discussed its updated structural analysis of the steam dryer that includes the ACM results for CLTP conditions from Supplement 26 combined with CFD pressure loads predicted for EPU conditions at VYNPS. The ACM analysis uses MSL instrumentation to project the measured pressure fluctuations as pressure loads on the steam dryer for the specific power level at which the plant is operating, and does not predict steam dryer loads for higher power levels. The ANSYS FEM for the VYNPS steam dryer analysis included the dryer support ring, dryer hoods, end plates, cover plates, upper dryer banks, cross beams, bottom support plates, tie bars, and gussets. In early 2005, the licensee identified the need to revise the FEM to model more accurately the connection of the gussets to the lower cover plate. The FEM used to evaluate steam dryer stress from CFD loads was updated at that time. The licensee performed hand calculations to verify that the stress at the gusset to cover plate connection from the ACM loads was significantly less than the applicable stress limit. As part of determining the EPU steam dryer load definition, the licensee will update the FEM model used in the ACM analysis to reflect the as-built connection of the gussets to the cover plate.

The licensee evaluated the dynamic structural response of the steam dryer to applied pressure fluctuations from acoustic loading using a time history method with modal superposition. The licensee performed a sensitivity assessment by varying the time interval between the pressure time steps by 10% (equivalent to peak broadening in the response spectrum analysis method). The licensee assigned an uncertainty to the stress amplitude of 20% due to load/response frequency uncertainty based on these shifted frequency analyses. However, the licensee did not include potential increased stress resulting from peak loading frequencies aligning with the dryer resonance frequencies in its analysis. For the fatigue stress evaluation, the licensee determined the peak stress for various locations on the VYNPS steam dryer by combining the stresses calculated from the CFD and ACM analyses by the SRSS method, and then multiplying the combined stress by applicable weld concentration and size factors. The licensee applied the acoustic and CFD uncertainties to calculate an uncertainty value for the limit curve factor used to monitor steam dryer performance. For the ASME load case assessments, the licensee increased the acoustic loading stress by a 130% uncertainty value and the CFD loading stress by a 16% uncertainty value, and combined these stresses by the SRSS method. The licensee then compared the results of these stress analyses to the applicable ASME allowable stress limits to demonstrate available structural margin in the VYNPS steam dryer. In Attachment 2 to Supplement 33, the licensee provided the results of its analysis of the dryer skirt indicating low acoustic loading stress for that region of the VYNPS steam dryer.

Based on the ASME fatigue stress limit, the licensee calculated an allowable limit curve over the frequency spectra using the CFD analysis for low frequency loads and the ACM analysis for the high frequency loads with the current MSL data input, including the consideration of uncertainties. The NRC staff reviewed the method used by the licensee to calculate the stress at various locations on the VYNPS steam dryer based on the pressure loads predicted by the CFD and ACM analyses. As a result of the uncertainties associated with the CFD and ACM analyses and MSL input data, the NRC staff indicated during the audit on August 22-25, 2005,

that it was important to demonstrate that the structural integrity of the steam dryer would not be challenged if the actual loads on the steam dryer reached the limit curve. In Attachment 5 to Supplement 33 of its EPU request, the licensee provided its assessment of the limit curve relative to the fatigue stress limit to demonstrate that, if the limit curve is not exceeded, the structural integrity of the VYNPS steam dryer will be assured. In its assessment, the licensee calculated the most limiting stress location as the [[ ]] with a stress of [[ ]] psi based on the CFD analysis at EPU conditions and a stress of [[ ]] psi based on the ACM analysis at CLTP conditions. As these stresses are associated with independent low and high frequency pressure loads, respectively, the combined peak stress for this location on the VYNPS steam dryer is calculated by the SRSS method to be [[ ]] psi. The licensee established a limit curve that would provide for an SRSS combination of CFD and ACM stress at the most limiting steam dryer location of 7393 psi. Therefore, the limit curve stress will provide considerable margin to the ASME fatigue limit stress of 13,600 psi. As discussed below, in accordance with the license condition discussed in SE Section 3.17.3, the licensee will provide its limit curve as part of the startup test procedure for VYNPS to the NRC staff prior to exceeding CLTP.

In Attachment 6 to Supplement 33 of its EPU request, the licensee describes its updated Steam Dryer Monitoring Plan (SDMP) for monitoring and evaluating the performance of the VYNPS steam dryer during power ascension testing and operation above CLTP to full EPU conditions to verify acceptable steam dryer performance. The licensee defines unacceptable steam dryer performance as a condition that could challenge steam dryer structural integrity and result in the generation of loose parts, cracks or tears in the dryer that result in excessive moisture carryover. The licensee proposed a license condition for steam dryer monitoring to require operational surveillances as well as visual inspections of the steam dryer at specific scheduled RFOs following achievement of full uprate conditions as shown in SE Section 3.17.3. The licensee stated that power ascension above CLTP would be conducted in 2.5% power steps and 5% power plateaus. The power ascension will include hold points at each 2.5% step and 5% plateau. The licensee stated that the maximum power increase would not exceed a nominal 5% power in a 24-hour period. The SDMP specifies that moisture carryover will be determined every 24 hours; MSL pressure data from strain gages will be obtained hourly when initially increasing power above a previously attained level and at least once every 2.5% power step above CLTP; and MSL pressure data from pressure transducers will be collected at least once every 2.5% power step above CLTP and within 1 hour after achieving every 2.5% power step above CLTP. The SDMP allows relaxed monitoring if the surveillance requirements are met at a power step, but requires a power reduction if a surveillance is not accomplished within the specified time intervals. In addition, the SDMP indicates that plant data which may be indicative of off-normal dryer performance will be monitored during power ascension (e.g., steam flow, feed flow, etc.).

The SDMP establishes criteria for verifying acceptable steam dryer performance at VYNPS using moisture carryover and MSL pressure data. The performance criteria are specified as Level 2 based on maintaining less than (or equal to) 80% of the ASME allowable alternating stress at 10^{11} cycles (i.e., 10,880 psi) and Level 1 based on maintaining the ASME allowable alternating stress at 10^{11} cycles (i.e., 13,600 psi). The Level 2 steam dryer performance criteria are (1) moisture carryover exceeds 0.1%; (2) moisture carryover exceeds 0.1% and increases by more than 50% over the average of the three previous measurements taken at greater than 1593 MWt; and (3) pressure data exceed the Level 2 spectra. If any of the Level 2 steam dryer performance criteria are exceeded, the SDMP specifies that (1) reactor power ascension be

promptly suspended until an engineering evaluation concludes that further power ascension is justified; and (2) before resuming reactor power ascension, the steam dryer performance data shall be reviewed as part of an engineering evaluation to assess whether further power ascension can be made without exceeding the Level 1 criteria. The Level 1 steam dryer performance criteria are (1) moisture carryover exceeds 0.35%; and (2) pressure data exceed Level 1 spectra. If either of the Level 1 steam dryer performance criteria is exceeded, the SDMP specifies that the licensee will:

(1) Promptly initiate a reactor power reduction and achieve a previously acceptable power level (i.e., reduce power to a previous step level) within 2 hours, unless an engineering evaluation concludes that continued power operation or power ascension is acceptable.

(2) Within 24 hours, re-measure moisture carryover and perform an engineering evaluation of steam dryer structural integrity. If the results of the evaluation of dryer structural integrity do not support continued plant operation, the reactor shall be placed in a hot shutdown condition within the following 24 hours. If the results of the engineering evaluation support continued power operation, implement steps (3) and (4) below.

(3) If the results of the engineering evaluation support continued power operation, reduce further power ascension step and plateau levels to nominal increases of 1.25% and 2.5% of CLTP, respectively, for any additional power ascension.

(4) Within 30 days, use the transient pressure data to calculate the steam dryer fatigue usage to demonstrate that continued power operation is acceptable.

The SDMP also specifies that, if the steam dryer performance criteria are exceeded, the following actions will be taken depending on the criteria exceeded:

(1) Either suspend reactor power ascension (Level 2 Acceptance Criteria) or reduce reactor power (Level 1 Acceptance Criteria), initiate a Condition Report, and evaluate the cause of any exceedance of the performance criteria.

(2) Prior to increasing reactor thermal power to a level higher than any previously attained, the plant conditions relevant to steam dryer integrity and associated evaluation results shall be reviewed by the on-site safety review committee, and a recommendation shall be made to the General Manager, Plant Operations prior to increasing power for each 5% power plateau.

(3) Strain gage pressure and moisture carryover data collected at each 5% power plateau will be made available to the NRC through its resident inspector.

(4) Each initial increase in reactor thermal power to the next higher 5% power plateau above 100% CLTP must be authorized by the General Manager, Plant Operations.

In addition, the SDMP states that other reactor operational parameters that may be influenced by steam dryer integrity (e.g., steam flow distribution between the individual steam lines) will be

monitored with the intent of detecting structural degradation of the steam dryer during plant operation (e.g., flow distribution between individual MSLs). Plant procedures will control the enhanced monitoring of selected plant parameters.

The SDMP states that the results of visual inspections of the steam dryer conducted during the next three RFOs shall be reported to the NRC staff within 60 days following startup from the respective RFO. The SDMP also states that its results shall be submitted to the NRC staff in a report within 60 days following completion of all EPU power ascension testing. In addition, the final full EPU power performance criteria spectra (limit curve) will be submitted to the NRC staff within 120 days.

As long-term actions, the SDMP states that the VYNPS steam dryer will be inspected during RFOs scheduled for fall 2005, spring 2007, fall 2008, and spring 2010, according to the recommendations of GE Services Information Letter (SIL) No. 644, Revision 1 (November 9, 2004). The SDMP also indicates that, following completion of EPU power ascension testing, moisture carryover measurements will continue to be made periodically, and other plant operational parameters that may be affected by steam dryer structural integrity will continue to be monitored, in accordance with GE SIL 644 and plant procedures. The SDMP notes that temporarily installed pressure monitoring sensors and strain gages may be removed from service following achievement of one operating cycle after issuance of the EPU license amendment and satisfaction of the license condition requirements for steam dryer inspections.

In Attachment 1 to Supplement 32 to its EPU request, the licensee modified its commitment to perform visual inspections of the steam dryer at VYNPS. In particular, the licensee describes its plan to perform a visual inspection during the fall 2005 RFO of the steam dryer modification, flaws left "as-is," and the repair made during the last RFO. The licensee indicates that this inspection plan satisfies recommendations A.1.c and A.1.d in GE SIL 644, Revision 1. The licensee also discusses its plan to conduct a visual inspection of all accessible, susceptible locations of the steam dryer during each of the three RFOs, beginning with RFO-26 (i.e., spring 2007) to satisfy recommendation B.2 in SIL 644, Revision 1. The licensee lists this steam dryer inspection plan as a regulatory commitment in Attachment 10 to Supplement 32.

In the cover letter for Supplement 33 of its EPU request, the licensee states that several actions will be taken with respect to providing confidence in the capability of the steam dryer at VYNPS to maintain its structural integrity under EPU conditions. In Attachment 1 to Supplement 36, Entergy specified those planned actions as part of a proposed license condition. The proposed license condition is shown in SE Section 3.17.3. The actions include:

(1) The licensee will install 32 additional strain gages on the main steam piping during the fall 2005 RFO and will enhance the data acquisition system prior to EPU operation in order to reduce the measurement uncertainty associated with the ACM.

(2) In the event that acoustic signals are identified that challenge the limit curve during EPU power ascension, the licensee will evaluate dryer loads and re-establish the limit curve based on the new strain gage data, and will perform a frequency specific assessment of ACM uncertainty at the acoustic signal frequency.

(3) After reaching 120% of CLTP, the licensee will obtain measurements from the MSL strain gages and establish the VYNPS dryer flow-induced vibration load fatigue margin, update the dryer stress report, and re-establish the SDMP limit curve with the updated ACM load definition and revised instrument uncertainty, which will be provided to the NRC staff.

(4) During power ascension, if an engineering evaluation is required in accordance with the SDMP, the licensee will perform the structural analysis to address frequency uncertainties up to ±10% and assure that peak responses that fall within this uncertainty band are addressed.

(5) The licensee will revise the SDMP to reflect long-term monitoring of plant parameters potentially indicative of a dryer failure; to reflect consistency of the VYNPS steam dryer inspection program with SIL 644, Revision 1; and to identify the NRR Project Manager for VYNPS as the point of contact for providing SDMP information during power ascension.

(6) The licensee will submit the final EPU VYNPS steam dryer load definition to the NRC upon completion of the power ascension test program.

(7) The licensee will submit the flow-induced vibration related portions of the EPU startup test procedure, including the methodology for updating the limit curve, prior to power ascension.

In Attachment 6 to Supplement 33 of its EPU request, the licensee proposed a license condition for implementation of the VYNPS SDMP. The proposed license condition was subsequently superceded by Supplement 36 of the EPU request. The proposed license condition is shown in SE Section 3.17.3.

The NRC staff has reviewed the information provided by the licensee in support of its analysis of the structural integrity of the VYNPS steam dryer under EPU conditions, and for monitoring steam dryer loads and performance during plant operation. Although significant uncertainty exists regarding the licensee's method for calculating specific stress values on the VYNPS steam dryer from its CFD and ACM analyses, the licensee's current MSL instrumentation suggests minimal excitation of the pressure frequency spectra in the MSLs at CLTP conditions. As a result, the staff finds that the licensee has demonstrated that the flow-induced stress imposed on the VYNPS steam dryer at CLTP conditions is within the fatigue stress limits provided in the ASME Code. However, the available margin to those stress limits is not readily verifiable. Therefore, the NRC staff considers the licensee's planned actions specified in Supplement 33 of its EPU request, and included in the proposed license condition in Supplement 36, to be an important part of the licensee's effort to provide confidence that the structural integrity of the steam dryer will be maintained during EPU operation. For example, the staff considers the use of the more accurate MSL strain gages to be installed for monitoring pressure fluctuations in the MSLs to be necessary in light of the large uncertainty in the current MSL instrumentation that provides input to the ACM analysis. The staff considers the selection of the new MSL instrumentation in terms of its sensitivity and signal-to-noise ratio to be important to its acceptability. The staff also considers it important to consider whether any acoustic sources might exist between the MSL strain gage locations. Further, the staff agrees with the importance of evaluating the peak frequencies within the ±10% frequency range when

the licensee re-evaluates the steam dryer loads if MSL strain gage data exceed the limit curve, or following achievement of EPU conditions, as part of establishing a new limit curve. During the licensee's evaluation of the results of the inspection of the VYNPS steam dryer to be conducted in the fall of 2005, the predictions of low stress (including in the skirt region) need to be compared to actual operating experience with the VYNPS steam dryer. The staff also considers the requirements specified by the licensee in the proposed license condition to be appropriate for establishing and implementing the SDMP at VYNPS.

In light of the large uncertainties in the CFD and ACM analyses and the fact that the ACM analysis has calculated the steam dryer pressure loads only at CLTP, the NRC staff determined that the licensee needs to closely monitor MSL strain gage data and other plant data as the reactor power is raised at VYNPS such that the ACM loads can be calculated at the increased power level to verify that the structural limits for the steam dryer are not reached. For example, the staff concluded that the new 32 MSL strain gages need to be monitored frequently during power ascension above CLTP for increasing pressure fluctuations in the steam lines. Hold points need to be established at 105%, 110%, and 115% of CLTP to collect plant data, conduct plant inspections and walkdowns, and evaluate the plant data for steam dryer performance. The time period for each hold point will need to be sufficient to complete all activities specified in the startup test procedure for the applicable hold point. Sufficient information and time will need to be provided to the NRC staff to determine whether any safety concerns exist prior to increasing power above each hold point. If any frequency peak from the MSL strain gage data exceeds the limit curve established by the licensee prior to operation above CLTP, the unit needs to be returned to a power level where the limit curve is not exceeded. The licensee would then resolve the uncertainties in the steam dryer analysis prior to further increases in reactor power. In the subsequent engineering evaluation, peak responses that fall within the ±10% frequency uncertainty band need to be considered as part of an adequate structural analysis. Further, the potential effect of the skirt in the steam dryer FEM on the stresses in the steam dryer components needs to be addressed. In addition to evaluating the MSL strain gage data, reactor pressure vessel water level instrumentation or MSL piping accelerometers need to be monitored frequently to help identify any resonance frequencies not captured by the MSL strain gage data and ACM analysis. If resonance frequencies are identified as increasing significantly above nominal levels established at CLTP conditions, power ascension needs to be stopped until an evaluation of continued steam dryer integrity is performed to demonstrate that no safety concerns exist. Within a reasonable time period following issuance of the EPU license amendment, the uncertainties in the steam dryer analysis need to be resolved to avoid long-term fatigue concerns with the steam dryer. In response to an NRC letter dated October 12, 2005, Entergy submitted a proposed license condition in Attachment 1 to Supplement 36 of its EPU application that addresses the NRC staff findings discussed above. The proposed license condition is shown in SE Section 3.17.3.

The NRC staff considers the development of an adequate EPU startup test procedure to be a significant action in confirming the safe operation of VYNPS during EPU conditions. The staff has determined that the EPU startup test procedure needs to include (a) the stress limit curve to be applied for evaluating steam dryer performance; (b) specific hold points and their duration during EPU power ascension; (c) activities to be accomplished during hold points; (d) plant parameters to be monitored; (e) inspections and walkdowns to be conducted for steam, FW, and condensate systems and components during the hold points; (f) methods to be used to trend plant parameters; (g) acceptance criteria for monitoring and trending plant parameters, and conducting the walkdowns and inspections; (h) actions to be taken if acceptance criteria

are not satisfied; and (i) verification of the completion of commitments and planned actions specified in the EPU application and all supplements to the application in support of the EPU request prior to power increase above CLTP. While the licensee indicates that plant parameters will be monitored to provide information on steam dryer performance, the staff also considers it important for additional steam dryer loading information to be obtained for qualitative evaluation from the reactor pressure vessel water level instrumentation or MSL piping accelerometers in light of the inadequacy of the ACM in calculating low frequency pressure loads on the steam dryer. While the SDMP indicates that other plant parameters (such as steam flow distribution between MSLs) will be monitored, the staff also considers it important for the frequency of such monitoring, acceptance criteria, and actions if those criteria are not satisfied, to be specified in the startup test procedure. In response to an NRC letter dated October 12, 2005, Entergy submitted a proposed license condition in Attachment 1 to Supplement 36 of its EPU application. The staff has determined that this proposed license condition addresses the NRC staff findings discussed above. The proposed license condition is shown in SE Section 3.17.3.

Prior to power ascension above CLTP and during the power ascension, the NRC staff has determined that sufficient time needs to be available during the hold points to allow the licensee to present plant information on potential adverse flow effects on the steam dryer (and other plant equipment) to the NRC staff for a determination of whether any safety concerns exist with power ascension. In Attachment 2 to Supplement 36 of its EPU request, Entergy submitted a regulatory commitment that addresses the NRC staff findings discussed above. As shown in SE Section 4.0 (Item No. 25), Entergy will provide information on plant data, evaluations, walkdowns, inspections, and procedures associated with the individual requirements of the license condition (pertaining to potential adverse flow effects) to the NRC staff prior to increasing power above 1593 MWt or each specified hold point, as applicable. If any safety concerns are identified during the NRC staff review of the provided information, Entergy will not increase power above 1593 MWt or the applicable hold point, and the specific requirements in the license condition will not be satisfied. The NRC staff considers that this commitment provides appropriate interaction between the licensee and the staff prior to and during power ascension above CLTP conditions.

2.2.6.2.2 Steam, Feedwater, and Condensate Systems and Components

In Attachment 1 to Supplement 15 (September 23, 2004) of its EPU request, the licensee stated that the VYNPS piping steady state vibration program for EPU power ascension testing follows the guidance in Part 3 of the ASME OM-S/G-2000 standard (ASME OM-3). The program assesses the FIV levels of selected piping systems that are expected to experience increased flow during EPU conditions. The licensee stated that vibration data will be taken at approximately 2.5% power increments above CLTP and will be evaluated for acceptability. For example, the MS and FW piping located in the drywell which is inaccessible during plant operation will be monitored for vibration levels using direct mounted accelerometers with acceptance criteria based on guidance in ASME OM-3. The FW regulator valves and attached FW piping located downstream of the reactor feed pumps will be monitored with a hand-held vibration meter. If vibration levels for these components increase significantly, the licensee will further evaluate the affected components.

Also in Attachment 1 to Supplement 15 of its EPU request, the licensee stated that it will employ visual monitoring during EPU power ascension testing to determine if significant

vibration is occurring in MS, FW, and condensate piping located in the turbine building heater bay. If visual observations indicate significant increased vibration, the licensee will further monitor this piping with a hand-held device. The licensee will also monitor system components determined to have FIV vulnerabilities based on plant-specific experience, industry operating experience, identification of FIV through plant inspections and walkdowns, and additional evaluation of components potentially susceptible to FIV at increased system flow. The licensee did not identify any components requiring FIV monitoring based on its own plant-specific experience. However, based on industry experience, the licensee will monitor the MS safety/relief valves using accelerometers on the MS piping; MS low point drain lines using accelerometers; and FW heater level control valves by means of inspections and walkdowns.

The licensee performed baseline inspections and walkdowns of the condensate, FW, and MS systems at VYNPS to identify systems and components with elevated vibration during CLTP operations. The licensee will compare the results of inspections and walkdowns performed during EPU power ascension testing, along with available vibration measurement data, to the baseline results. The licensee will enter components with significant increases in vibration into the VYNPS corrective action program and will evaluate those components for acceptability and additional action. The licensee stated that it will evaluate additional system components that might be susceptible to FIV at EPU conditions. The licensee indicated that it had identified condensate, FW, MS piping for cantilevered piping configurations as potentially susceptible to FIV, and that those components will be monitored if found susceptible. In its list of commitments attached to Supplement 15, the licensee stated that, during EPU power ascension testing, it will implement FIV and steam dryer monitoring, including associated evaluation as necessary during EPU power ascension testing as described in Entergy letter BVY 04-100 (Supplement 15 to its EPU request).

In Attachment 2 to Supplement 32 of its EPU request, the licensee determined, since the submittal of Supplement 15, that isokinetic sample probes are used in the MS, FW, and condensate systems at VYNPS. Those sample probes are subject to the effects of FIV. The licensee evaluated the susceptibility of the sample probes to high cycle fatigue failure. The licensee stated that the four susceptible probes (SP-26, 27, 30, and 31) in the FW and condensate systems needed to be modified to address the failure vulnerability. The licensee specified its plan to modify the four susceptible isokinetic sample probes in the FW and condensate systems during the fall 2005 RFO as a regulatory commitment in Attachment 10 to Supplement 32. This commitment was satisfied as documented in Reference 74.

The NRC staff has reviewed the information submitted by the licensee on the monitoring of MS, FW, and condensate systems and components during EPU power ascension testing. The licensee indicated that significant vibration monitoring and walkdown/inspection activity will be conducted during the power ascension above CLTP. However, the power ascension test plan described in Supplement 15 does not specify the frequency of the vibration data collection, or the walkdowns and inspections with respect to the power ascension hold points discussed in the SDMP. For example, it is not apparent whether the vibration monitoring and walkdown/inspection activities can be accomplished within the hold points specified in the SDMP. While some acceptance criteria for vibration monitoring are provided, actions to be taken with respect to the power ascension in the event of failed acceptance criteria for the vibration monitoring and walkdown/inspection activities are not clearly indicated. Therefore, the NRC staff considers it important for the licensee to provide the relevant sections of its EPU

startup test procedure to the NRC prior to plant operation above CLTP. This requirement is included in the license condition shown in SE Section 3.17.3.

2.2.6.3 Conclusion

The NRC staff has reviewed the licensee's consideration of potential adverse flow effects on the MS, FW, and condensate systems and their components (including the steam dryer) for operation of VYNPS at EPU conditions. The staff concludes that the licensee has provided reasonable assurance that the flow-induced effects on the steam dryer and other plant equipment are within the structural limits at CLTP conditions. The staff further concludes that the licensee has demonstrated that the MS, FW, and condensate systems and their components (including the steam dryer) will continue to meet the requirements of draft GDC-1, 2, 40, and 42 following implementation of the proposed EPU at VYNPS, subject to the license condition discussed above. Therefore, the staff finds the proposed license amendment to operate VYNPS at EPU conditions to be acceptable with respect to potential adverse flow effects.

As noted in the technical evaluation, a license condition will be added to the VYNPS Facility Operating License as shown in SE Section 3.17.3. The license condition provides requirements for monitoring, evaluating, and taking prompt action in response to potential adverse flow effects as a result of operation at EPU conditions. The intent of the license condition is to (1) confirm the licensee's predictions regarding the hydrodynamic loads on the steam dryer; (2) confirm the licensee's predictions regarding the acoustic pressure loads on the steam dryer; and (3) confirm the safe operation of VYNPS during power ascension above CLTP.

2.3 Electrical Engineering

2.3.1 Environmental Qualification of Electrical Equipment

Regulatory Evaluation

Environmental qualification (EQ) of electrical equipment involves demonstrating that the equipment is capable of performing its safety function under significant environmental stresses which could result from design-basis accidents (DBAs). The NRC staff's review focused on the effects of the proposed EPU on the environmental conditions that the electrical equipment will be exposed to during normal operation, anticipated operational occurrences, and accidents. The NRC staff's review was conducted to ensure that the electrical equipment will continue to be capable of performing its safety functions following implementation of the proposed EPU. The NRC's acceptance criteria for EQ of electrical equipment are based on 10 CFR 50.49, which sets forth requirements for the qualification of electrical equipment important to safety that is located in a harsh environment. Specific review criteria are contained in SRP Section 3.11.

Technical Evaluation

As described in Section 10.3 of the PUSAR, the VYNPS safety-related electrical equipment was reviewed by the licensee to assure the existing qualification for normal and accident conditions expected in the areas where the devices are located remain adequate at EPU conditions.

The EQ for safety-related electrical equipment located inside the containment is based on main steam line break (MSLB) and/or LOCA conditions and their resultant temperature, pressure, humidity and radiation consequences, and includes the environments expected to exist during normal plant operation. Normal temperatures are expected to increase slightly for EPU, but remain bounded by the normal temperatures used in the EQ analyses. The accident conditions for temperature and pressure, used in the current EQ analyses, bound the EPU accident conditions.

Accident temperature, pressure, and humidity environments used for the qualification of equipment outside containment result from a MSLB or other high energy line breaks (HELBs) whichever is limiting. The peak HELB temperatures at EPU conditions, in some cases, exceed the values used for EQ at CLTP conditions. The EPU temperature peaks that are not bounded by the CLTP conditions were evaluated by the licensee. Affected components were either requalified to EPU conditions by crediting new qualification tests or analysis, or by relocating the components to milder environments. The accident temperature resulting from a LOCA/MSLB inside containment increased the temperature in some reactor building areas due to additional heat load from the increase in wetwell temperatures. However, the increase in long-term post-accident temperatures was evaluated and determined not to adversely affect the qualification of safety-related electrical equipment.

In response to an NRC staff RAI, the licensee submitted additional information by letter dated May 19, 2004 (Reference 8), to provide the results of additional radiation dose analyses that were required to demonstrate electrical equipment qualification at EPU conditions. The results of the analysis show that the VYNPS safety-related electrical equipment is qualified for operation under EPU conditions.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the EQ of electrical equipment and concludes that the licensee has adequately addressed the effects of the proposed EPU on the environmental conditions for and the qualification of electrical equipment. The NRC staff further concludes that the electrical equipment will continue to meet the relevant requirements of 10 CFR 50.49 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the EQ of electrical equipment.

2.3.2 Offsite Power System

Regulatory Evaluation

The offsite power system includes two or more physically independent circuits capable of operating independently of the onsite standby power sources. The NRC staff's review covered the descriptive information, analyses, and referenced documents for the offsite power system; and the stability studies for the electrical transmission grid. The NRC staff's review focused on whether the loss of the nuclear unit, the largest operating unit on the grid, or the most critical transmission line will result in the loss of offsite power (LOOP) to the plant following implementation of the proposed EPU. The NRC's acceptance criteria for offsite power systems are based on draft GDC-39. Specific review criteria are contained in SRP Sections 8.1 and 8.2, Appendix A to SRP Section 8.2, and Branch Technical Positions (BTPs) PSB-1 and ICSB-11.

Technical Evaluation

2.3.2.1 Grid Stability

In an RAI, the NRC staff asked the licensee to describe the actions that would be taken to address the depletion of the nuclear unit mega-volt amperes reactive (MVAR) capability with the EPU on a grid-wide basis. By letter dated January 31, 2004 (Reference 6), the licensee responded that the study done by GE Power Systems Energy Consulting indicated that the plant requires to implement a number of modifications to address the impact of the EPU on the transmission system including the additional MVAR capacity to maintain voltage support on a grid-wide basis. The modifications related to grid stability are listed in SE Section 1.4.

As discussed in the licensee's letter dated September 10, 2005 (Reference 33), the licensee originally planned to implement the EPU in two steps, 15% for the first step, and 5% for the second step. The two step process was necessary because some of the EPU-related plant modifications were scheduled to be completed in the next refueling outage after approval of the EPU request. However, since all the modifications necessary to support full EPU operation were completed during the fall 2005 refueling outage (RFO 25), the EPU will be implemented in one step.

As discussed in PUSAR Section 7.1, the existing VYNPS generator (i.e., at CLTP) is rated 626 MVA, which results in a rated electrical output (gross) of 563 megawatts electric (MWe) at a power factor (pf) of 0.9. As discussed in the licensee's letter dated January 31, 2004 (Reference 6), the existing generator MVAR capability at rated output is approximately 330 MVAR, however, the VYNPS experiences increased turbine-generator vibration at MVAR loading greater than 150 MVAR. The vibration is related to uneven heating of the generator rotor at increased field current. To support the EPU, the main generator has been rewound with a rating of 684 MVA at 0.969 pf and the existing rotor was re-insulated. Under EPU conditions, the nominal generator output would increase to 667 MWe. At this output, the generator MVAR capability will remain at the pre-EPU capability of 150 MVAR. The licensee has installed a 60 MVAR capacitor bank at the VYNPS 115 kilovolt (kV) switchyard to maintain proper system voltage requirements. A grid impact study was provided by the licensee's letter dated August 25, 2004 (Reference 13). The NRC staff reviewed the licensee's submittals and, based on the modifications performed to support the EPU, concluded that the EPU will have no adverse impact on grid stability.

2.3.2.2 Main Generator

The current main generator is rated at 626 MVA which results in a rated electrical output of 563 MWe at a 0.9 pf. In order to achieve the higher electrical output for EPU, the generator was rewound. The NRC staff questioned the licensee regarding the modifications to the main generator. By letter dated January 31, 2004 (Reference 6), the licensee responded that the main generator is being upgraded/rewound from a rating of 626 MVA to a rating of 684 MVA by replacement of water-cooled stator bars. The existing stator bars are original plant equipment and have experienced some corrosion, leakage, and in some instances, deterioration of insulation. The new stator bars have an improved design with new material to minimize the chance of leakage. By letter dated August 25, 2004 (Reference 13), the licensee stated that the proposed EPU was represented by a generator rating of 684 MVA, a final power output rating of 667 MWe, and a gross output rating of 150 MVAR at rated power output. Revised

generator reactive capability curves at EPU conditions maintain the generator stator core and field winding within their design limits, (i.e., no modification to the stator core and field winding is required for EPU). The generator hydrogen cooling system pressure is unchanged at EPU. However, the hydrogen cooling system heat exchangers have been replaced by heat exchangers of higher capacity due to increased heat removal requirements at EPU conditions. Additionally, the bushing current transformers (CTs) have been replaced for the EPU. No generator protective relay changes are necessary, however, some protective relay setpoints will be modified for the rewound generator rating.

The NRC staff reviewed the licensee's submittals and concluded that the main generator would be acceptable for EPU based on the modifications described above.

2.3.2.3 Main Transformer

The main transformer is rated at 675 MVA. The main power transformer has recently been replaced and was sized to support the EPU. The associated switchyard components (rated for maximum transformer output) are adequate for transformer output. The loading on the main transformer is 650 MVA (main generator output of 684 MVA minus the 34 MVA house load fed through the unit auxiliary transformers), which is below the main transformer rating of 675 MVA.

The NRC staff reviewed the licensee's submittal and concluded that the main power transformer and the associated switchyard components are adequate for the uprated generator output and, therefore, operating the main transformer at the uprated power condition is acceptable.

2.3.2.4 Isophase Buses

The isophase bus duct connects the main generator to the primary windings of the main transformer and the unit auxiliary transformer and is rated at 17.9 kilo-amps (KA). The rating at the EPU conditions will be 19 KA. The NRC staff questioned the licensee regarding how the capacity of the isophase bus duct would be increased for the EPU. By letter dated January 31, 2004 (Reference 6), the licensee responded that the isophase bus duct is being upgraded from a rating of 17.9 KA to a rating of 19 KA by replacement of the bus duct cooler and by internal modifications to the bus duct cooling air distribution system.

The NRC staff reviewed the licensee's submittals and concluded that the operation of the isophase bus duct would be acceptable for the EPU after upgrading it from a rating 17.9 KA to a rating of 19 KA.

2.3.2.5 Unit Auxiliary Transformer (UAT)

The UAT is rated at 39.2 MVA. The EPU output is 34.4 MVA based on the worst-case loading.

The NRC staff reviewed the licensee's submittals and concluded that the increase in house loads resulting from the EPU is below the maximum UAT design rating and, therefore, operating the UAT at the uprated power condition is acceptable.

2.3.2.6 Startup Transformers

The two startup transformers are each rated at 28 MVA. Under EPU conditions, the loading on one transformer is 17.9 MVA and the loading on the other is 24.8 MVA.

The NRC staff reviewed the licensee's submittals and concluded that the startup transformers are not impacted by the EPU, and, therefore, operating the startup transformers at the uprated power condition is acceptable.

2.3.2.7 Station Loads

The licensee reviewed the station loads under normal, transient, and emergency operating scenarios for EPU conditions. In all cases, loads were computed based on equipment nameplate ratings or brake horsepower and were found acceptable for the EPU conditions. However, the licensee's application did not provide an evaluation for the operation of condensate and reactor feedwater pump motors at higher summer temperatures at the EPU conditions. In response to an NRC staff RAI, the licensee provided a detailed analyses for the condensate and reactor feedwater pump motors by letter dated May 19, 2004 (Reference 8). The two Westinghouse condensate pump motors and one GE condensate pump motor are adequate for operation at EPU conditions. Both the Westinghouse and the GE analyses bound the predicted pump flow run out. The Westinghouse reactor feedwater pump motors are adequate for operation at the EPU conditions. The feedwater pump motors remain adequate for all pump operating conditions including flow run-out.

The NRC staff reviewed the licensee's submittals and concluded that the station loads are not impacted by the EPU, and, therefore, operating the VYNPS with station loads at the uprated power condition is acceptable.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the offsite power system and concludes that the offsite power system will continue to meet the requirements of draft GDC-39 following implementation of the proposed EPU. Results of these evaluations show that following implementation of the proposed modifications to the main generator, isophase bus duct and an addition of a 60 MVAR capacitor bank, the design will be acceptable for EPU conditions. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the offsite power system.

2.3.3 AC Onsite Power System

Regulatory Evaluation

The alternating current (ac) onsite power system includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to safety-related equipment. The NRC staff's review covered the descriptive information, analyses, and referenced documents for the ac onsite power system. The NRC's acceptance criteria for the ac onsite power system are based on draft GDC-24 and 39, insofar as they require the system to have the capacity and capability to perform its intended functions during

anticipated operational occurrences and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.1.

Technical Evaluation

The emergency diesel generators (EDGs) supply the source of power following a LOOP or degraded voltage conditions. The EDGs automatically supply ac power to the Class 1E buses in order to provide motive and control power to equipment required for safe shutdown of the plant and mitigation and control of accidents. The amount of power required to perform safety-related functions (pump and valve loads) will not increase with the EPU and the current power system remains adequate. Therefore, the performance of the EDGs and the 4 kV emergency system is not affected by the EPU.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ac onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system's functional design. The NRC staff further concludes that the ac onsite power system will continue to meet the requirements of draft GDC-24 and 39 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ac onsite power system.

2.3.4 DC Onsite Power System

Regulatory Evaluation

The direct current (dc) onsite power system includes the dc power sources and their distribution and auxiliary supporting systems that are provided to supply motive or control power to safety-related equipment. The NRC staff's review covered the information, analyses, and referenced documents for the dc onsite power system. The NRC's acceptance criteria for the dc onsite power system are based on draft GDC-24 and 39, insofar as they require the system to have the capacity and capability to perform its intended functions during anticipated operational occurrences and accident conditions. Specific review criteria are contained in SRP Sections 8.1 and 8.3.2

Technical Evaluation

The licensee reviewed the dc power distribution system and determined that there were no identified load changes that affect the dc power system, therefore, the battery duty cycle, voltages-to-end devices, and available fault currents are within the design rating.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the dc onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system's functional design. The NRC staff further concludes that the dc onsite power system will continue to meet the requirements of draft GDC-24 and 39 following implementation of the proposed EPU. Adequate physical and electrical separation exists and the system has the capacity and capability to supply power to all

safety loads and other required equipment. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the dc onsite power system.

2.3.5 Station Blackout

Regulatory Evaluation

Station blackout (SBO) refers to a complete loss of ac electric power to the essential and nonessential switchgear buses in a nuclear power plant. SBO involves a LOOP concurrent with a turbine trip and failure of the onsite emergency ac power system. SBO does not include the loss of available ac power to buses fed by station batteries through inverters or the loss of power from alternate ac (AAC) sources. The NRC staff's review focused on the impact of the proposed EPU on the plant's ability to cope with and recover from an SBO event for the period of time established in the plant's licensing basis. The NRC's acceptance criteria for SBO are based on 10 CFR 50.63. Specific review criteria are contained in SRP Section 8.1 and Appendix B to SRP Section 8.2; and other guidance provided in Matrix 3 of RS-001.

Technical Evaluation

SBO Coping Analysis

As described in UFSAR Section 8.5.5.1, VYNPS uses an AAC source approach for coping with an SBO using the methodology of RG 1.155, "Station Blackout," dated August 1988. VYNPS relies on the Vernon Hydroelectric Station (VHS) to provide power to an emergency bus until offsite or onsite AC power is available. The VYNPS licensing basis with respect to coping duration is 8 hours.

As described in SE Section 1.6, an engineering inspection was conducted by the NRC at VYNPS from August 9 through September 3, 2004. As documented in the NRC's inspection report dated December 2, 2004 (Reference 55), the inspection team identified a non-cited violation of 10 CFR 50.63, "Loss of all alternating current power," because the licensee had not completed a coping analysis for the period of time the AAC source (the VHS) would be unavailable and had not demonstrated by test the time required to make the alternate source available for an SBO involving a grid collapse. This finding applies to current plant operation as well as under EPU operating conditions.

The engineering inspection team found that in the event of a regional grid collapse, the VHS would trip offline and have to be restarted. For SBO scenarios where the licensee cannot demonstrate by test that the AAC source would be available in 10 minutes, 10 CFR 50.63 requires the licensee to complete a coping analysis for the period of time it would take for power to be restored. Prior to the inspection, the licensee had credited the VHS as being available within 10 minutes. As such, the licensee had not performed a coping analysis. As a result of issues raised by the NRC during the inspection, concerning the communications and actions required to restart the VHS, the licensee created a preliminary timeline which estimated the time to restore power following a grid collapse could be between 20 minutes and 2 hours. Since it was determined that the VHS could not be made available in 10 minutes, the licensee performed a coping analysis. The coping analysis, which the licensee performed assuming EPU conditions, was submitted to the NRC by letter dated March 24, 2005 (Reference 26).

The licensee's coping analysis used the guidelines provided in RG 1.155 and Nuclear Management and Resource Council, Inc. (NUMARC) 87-00, Revision 1, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors." The licensee's analysis is based on a 2-hour coping requirement (i.e., the period of time the AAC is assumed unavailable). The coping analysis determined that:

- adequate condensate inventory will be available for decay heat removal;
- the plant 125 VDC station batteries have adequate capacity to supply all SBO DC loads for 2 hours;
- the SBO equipment operability will be maintained at elevated room temperatures caused by loss of ventilation;
- containment isolation capability will be maintained, as required, to ensure containment integrity;
- the plant compressed air system is not essential to cope with the SBO; and
- the resultant torus temperature satisfies the net positive suction head (NPSH) requirements of the residual heat removal (RHR) and core spray (CS) pumps without the need for crediting containment accident pressure.

Based on the results of the analysis the license concluded that: (1) the plant is able to safely cope with a total loss of AC power for a minimum of 2 hours from the onset of the SBO to the restoration of offsite AC power to a 4160-volt emergency bus; and (2) a 2-hour coping time is sufficient to envelope the time required to start and align the AAC source.

The NRC staff's review of each of the areas in the coping analysis (i.e., condensate inventory, battery capacity, loss of ventilation, containment isolation, compressed air, and torus temperature) is provided in separate parts of this SE section below.

Impact of EPU on SBO Coping Capability

For the EPU, the duration for which VYNPS must cope until the AAC source is available is 2 hours. The total coping time of 8 hours remains unchanged for EPU conditions. No changes are needed to the systems and equipment required to respond to an SBO event. The licensee stated that the use of two RHR service water (RHRSW) pumps with one RHR heat exchanger has been shown by calculation to provide sufficient suppression pool cooling to ensure adequate NPSH is available to ECCS pumps without crediting containment accident pressure under EPU conditions.

VHS and Vernon Tie Line

As discussed in UFSAR Section 2.4.1, the VHS is located on the Connecticut River, about 3,500 feet downstream of the VYNPS. As discussed in Attachment 4 of the licensee's letter dated August 1, 2005 (Reference 31), TransCanada, is the owner/operator of the VHS. TransCanada currently has four personnel that work out of the VHS during the normal day shift and report to that location. These personnel may be used to support activities at the other

TransCanada units. These workers are trained to return the unit to service and make it available to be connected to the system. They have an on-call supervisor assigned who covers the Vernon, Bellows Falls and Wilder hydroelectric facilities and carries a pager. For off-hour events, TransCanada's control station, which is continuously manned and located in Wilder, Vermont, would contact the on-call supervisor who would call in the necessary support personnel to restart the VHS.

The VHS is designated as a "black-start" facility under arrangements with the regional grid operator. This designation requires that the facility be capable of being black-started within 90 minutes after the operator is notified. The VHS is connected to a 69 kV offsite transmission system, which is not directly connected to VYNPS' normal offsite power source. The majority of the lines emanating from the VHS are routed in completely different directions from the lines supplying offsite power to VYNPS. The Vernon Tie is a highly reliable tie line that connects the VHS to either of the two VYNPS 4160 V emergency buses and is capable of supplying power to required loads under postulated SBO conditions. The Vernon Tie is physically and electrically independent of other sources of power to the plant's emergency buses, including the normal offsite power circuits. The load-carrying capability of the Vernon Tie is approximately equal to one EDG. The cable from VYNPS to the Vernon switchyard is buried and about 4000 feet in length. Energization of a VYNPS 4160-volt emergency bus from the Vernon Tie requires the closure of two circuit breakers from the VYNPS control room. Once energized, it takes very little time to align loads from the control room. Loss of the Vernon Tie is annunciated and its voltage is monitored in the VYNPS control room. Surveillance testing of the Vernon Tie demonstrated the ability to energize an emergency bus and supply required SBO loads in less than 10 minutes.

TransCanada conducts and documents the black-start of the VHS annually. The combination of the periodic testing of the AAC source together with the energization test of the emergency bus that is conducted every operating cycle encompass the condition of the SBO event, and therefore meets the requirement of 10 CFR 50.63.

Restoration of the bulk power system is of high importance following a regional blackout, and an emergency condition at VYNPS would receive top priority by the Rhode Island, Eastern Massachusetts, Vermont Energy Control (REMVEC) and National Grid Operators who control regional grid operations. This is evident in Independent System Operator - New England (ISO-NE) operating procedure OP-6, "System Restoration," which states that a high priority must be given to the restoration of offsite AC power to nuclear units and that they are the most critical during the restoration of power after a blackout.

If the Vernon Tie is de-energized an alarm is annunciated in the VYNPS control room. Per operational transient procedure OT-3122, if an SBO occurs and the Vernon Tie is de-energized due to a regional grid blackout, the VYNPS control room operator will immediately contact the REMVEC control center to black-start the VHS and re-energize the Vernon Tie in accordance with REMVEC operating procedures. Once the Vernon Tie is energized, a source of power is available to either of two VYNPS emergency buses, and equipment can be powered and operated in accordance with procedure OT-3122. The licensee provided a realistically conservative timeline for restoration of the AAC power source during a postulated regional blackout scenario. The licensee stated that operation of the VHS is monitored 24 hours per day, 7 days per week, by the owner/operator's staff located at the Wilder control center. Loss of the VHS would be immediately indicated at the Wilder station.

During the review, the NRC staff raised a concern that the licensee had not indicated in the March 24, 2005, submittal that it was planning to perform an integrated test, as required by 10 CFR 50.63(c)(2), with all parties involved to show that they can meet the 2-hour basis for starting and aligning the AAC power source, should it have to be re-started during a regional blackout. The staff considers such a test to be critical to showing that appropriate procedures and protocols are in place to coordinate between the multiple entities that would be involved. In response to the above concern, the licensee, on August 1, 2005 (Reference 31), stated that Entergy's letter of March 24, 2005, described the regular, periodic testing currently conducted to demonstrate the ability of the AAC source to power required electrical loads under a postulated SBO event. The testing consists of two components:

- Actual black-start of the AAC source by TransCanada in accordance with ISO New England operating procedure OP-11, "Black Start Capability Testing Requirements." This testing is conducted and documented annually.

- Surveillance testing of the Vernon Tie in accordance with Entergy VYNPS procedure OP-4142, "Vernon Tie and Delayed Access Power Source Backfeed Surveillance." Performance of this test demonstrates actual ability to power required SBO loads. This testing is conducted and documented during each refueling outage.

These tests perform all actions required to restart the VHS and, upon delivery of power to the Vernon Tie, the re-energization of a VYNPS 4 kV bus. The only step not performed is the opening and closing of certain breakers in the interfacing switchyard as this would cause an unnecessary blackout to the general public. These breaker manipulations are performed in a continuously manned regional operations center and controlled by system procedures. Completion of these external activities will take less than 5 minutes of the 2-hour coping duration and are considered simple actions not requiring periodic validation.

Entergy recently held discussions with TransCanada, the owner/operator of the VHS, and the regional grid control center regarding procedural requirements and communication protocols for a postulated SBO event. These communications have resulted in system restoration procedure improvements and have served to promote a better understanding of the expectations relative to Entergy's reliance on the VHS during an SBO.

Entergy has established administrative controls to assure performance of a once per operating cycle tabletop review of the procedures that complete the actions to repower a VYNPS 4KV bus from the VHS. Pursuant to discussions with co-host REMVEC, a system-wide annual tabletop review took place in October 2005. During this meeting, Entergy led a tabletop review of all actions required to support the restoration of 4 kV AC to VYNPS. This review discussed the interfaces with the operator of VHS and the regional grid operator to verify that roles and responsibilities and timelines are understood and that there have been no changes that would impact the assumption in the VYNPS SBO coping strategy. Entergy also provided the participants with additional insights regarding offsite power issues for nuclear power stations including plant response to and consequences of an SBO.

In a letter dated September 7, 2005 (Reference 62, RAI EEIB-A-8), the NRC staff requested that the licensee provide additional information regarding how the periodic tabletop review will verify the activities associated with notifying and staffing the VHS within 90 minutes as shown in the licensee's timeline for AAC source startup and alignment (Reference 26, Attachment 1,

Table 1, Step 3). The licensee responded to this concern in a letter dated September 10, 2005 (Reference 33, Attachment 7). The licensee stated that Step 3 in Table 1 of Attachment 1 to Supplement 25 provided a realistically conservative estimate of the time required to staff and prepare the VHS for black-start under worst-case SBO conditions. The periodic tabletop review of this step in the power restoration sequence will provide added assurance that the VHS can be staffed and prepared to commence generation within 90 minutes of notification as specified in ISO-NE procedure OP-11. The licensee stated that the ISO-NE system restoration exercise tabletop review, or a separate TransCanada/Entergy discussion of this 90-minute assumption, will include discussions with the VHS operator to confirm that the assumptions and completion times of restoration activities continue to remain valid. The activities and support elements to be reviewed include confirming that:

- VHS black-start restoration procedures support the 90-minute objective and are consistent with interfacing procedures of other participants involved in restoring AC power to VYNPS during an SBO event.
- The assumption that the 90-minute timeframe includes off-hours response under adverse weather conditions (e.g., snow storms).
- VHS units with black-start capability have actually been black-start tested in the past year and are in a condition to be black-started.
- Key operating aids used to support black start, such as telephone and radio communications, have been tested in the past year.
- VHS on-call personnel are sufficient in number and proximity to VHS to support timeline assumptions.
- VHS on-call personnel are subject to fitness-for-duty requirements
- VHS on-call personnel are qualified for black-start operations.
- Future plans (if any) to modify procedures, staffing requirements or the black-start units will continue to support the 90-minute objective.

Suggestions will be made when appropriate to increase time margins where situations warrant. The tabletop reviews will be interactive discussions intended to verify that the 90-minute objective can be met with reasonable assurance. The above discussion adequately addressed the NRC staff's concern.

During the review, the NRC staff raised the concern that the operators of the VHS are not Entergy personnel or Entergy contractors or vendors, and the station is not manned 24 hours per day, 7 days per week. The staff requested, in its letter dated July 27, 2005 (Reference 61, RAI EEIB-A-2), the licensee to address the following:

- Are specific operators designated "on-call" to respond to the VHS, as needed, during periods when the station is unmanned?
- Are the operators subject to any fitness-for-duty requirements?

- Are the operators, responsible for responding to the station when it is unmanned, required to remain within a certain distance from the VHS?

- From the onset of a regional grid collapse, during a period for which the VHS is unmanned, discuss all assumptions regarding the time required for the operator to reach the station (e.g., adverse weather conditions, distance traveled), and the time required for the operator to perform the necessary actions to black-start the station.

In its letter dated August 1, 2005, the licensee stated that Entergy has discussed these questions with TransCanada, the owner/operator of the VHS. In support of its commitment with ISO-NE to provide black-start capability, TransCanada currently has four personnel that work out of the VHS during the normal day shift and report to that location. These personnel may be used to support activities at the other TransCanada units. These workers are trained to return the unit to service and make it available to be connected to the system. They have an on-call supervisor assigned who covers the Vernon, Bellows Falls and Wilder hydroelectric facilities and carries a pager. For off-hour events, TransCanada's control station, which is continuously manned and located in Wilder, Vermont, would contact the on-call supervisor who would call in the necessary support personnel to restart the VHS. Any alarm indication that the VHS has tripped off-line is treated as a critical alarm and would prompt the call-in immediately upon receiving the alarm. Based on their experience, which includes off-hours events in which the VHS needed to be re-started, TransCanada indicated that they had restarted the unit within the required ISO-NE response timeframe. They also indicated that they had not experienced situations where personnel were unavailable to support restart of the unit.

TransCanada indicated that they did have company policies that include a fitness-for-duty program. Although random drug testing is not performed on all personnel, supervisory observations that identified a potential for alcohol or drug abuse would lead to drug testing. It is TransCanada's expectation that the on-duty supervisor be fit to perform this duty when on-call.

A key assumption in the coping evaluation time line is that the personnel in the Wilder control station would be aware of a regional blackout almost immediately. During off hours, which maximizes response time, the control center would contact the on-call supervisor who would contact and dispatch personnel to restart the VHS. Given current agreements, testing practices and past experience, the timeline assumes that this will be completed within 90 minutes.

For a regional blackout, ISO-NE would direct the system restoration and order the transmission owner to close the switchyard breakers supplying VYNPS. These breakers can be operated remotely by the transmission owner. VYNPS' operators would then close the breakers supplying power to the emergency bus. The VYNPS breakers are operated from the VYNPS main control room, and these actions can occur very quickly. The coping study uses a 2-hour duration which bounds the actions discussed above.

To account for additional unforeseen circumstances (e.g., adverse weather beyond assumed travel time) the coping study is done with conservative inputs that provide additional margin. For example, should the SBO event occur during a winter snow storm that could delay VHS startup, the conservatisms in heat sink temperature (which assumes peak summer allowable temperature) would allow for additional coping time.

Entergy believes that use of a 2-hour coping time together with conservatisms inherent in the coping analysis, in addition to conservatively estimated response times, provide reasonable assurance that the VHS will be available to support mitigation of an SBO event. During the review, the NRC staff raised a concern that during a snow storm, startup of the VHS could be delayed such that the 2-hour coping time may not be bounding (Reference 62, RAI EEIB-A-7). Additionally, the staff asked the licensee to provide details regarding the ISO-NE response timeframe. The licensee responded to this concern in a letter dated September 10, 2005 (Reference 33, Attachment 7). The licensee's response stated that:

> The coping study assumes worst-case conditions corresponding to the design basis river water temperature of 85°F. These conservatisms are bounding as they result in a minimum coping time of two hours. The statement made in the response to RAI EEIB-A-2 in Supplement 30, Attachment 4, was an engineering judgment that a lower (e.g., winter) river water temperature would enable the plant to cope for a duration longer than two hours as suppression pool temperature is the limiting constraint. However, the assertion that this capability exists in no way implies that a coping time in excess of two hours will ever be required. The coping time of two hours is based on worst-case conditions and is bounding. The two hour coping time is adequate for all times of the year, as well as all postulated weather conditions. The VYNPS SBO coping analysis report, which is applicable for EPU conditions, was provided by Entergy letter of March 24, 2005.
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> An integral portion of the ISO-NE system restoration procedure is the requirement that generating units having black start capability strive to achieve the fastest start time possible. ISO-NE black start units, such as the VHS units, are expected to be manned and prepared to commence generation within ninety (90) minutes of receiving instructions to initiate black start operations. In addition, ISO-NE procedure OP-6 requires that during system restoration a high priority must be given to the restoration of off-site AC power sources to nuclear generators. It is stated in procedure OP-6: "[T]he most critical power requirement after a [system] blackout is the assurance of reliable shutdowns of nuclear generators.... The expeditious restoration of alternative off-site AC power sources to nuclear units is imperative to promote the continued reliability of shutdown operations." Based on the designation of the TransCanada VHS units as black start units by ISO-NE, the procedural requirements for achieving black start, and the operating history of the VHS units, there is reasonable assurance that a VHS unit will be available within the SBO coping timeframe.

The NRC staff has determined that the above discussion adequately resolves the staff's concern regarding the 2-hour coping time.

Additionally, the NRC staff requested the licensee to discuss the agreement between Entergy and other entities to bring the VHS online from black-start conditions in order to provide electric power to VYNPS during an SBO event (i.e., whether there are formal written agreements supported by written procedures). In its response dated August 1, 2005 (Reference 31), the licensee stated that VYNPS has an AAC Source Agreement dated July 31, 2002, with Green Mountain Power (GMP), which is the retail electricity provider in the area, to make available at the point of interconnection between the Vernon switchyard and the Vernon Tie up to 3 MW of energy from the VHS during an emergency affecting VYNPS. This agreement requires GMP to take all reasonable steps to keep the Vernon Tie energized at all times. GMP has in turn

entered into an agreement, dated July 31, 2002, with USGenNE (the operator of the VHS at the time) to supply this power to VYNPS in an emergency, and a Service Agreement for Network Integration Transmission Service Between New England Power Company and Green Mountain Power Corporation, effective July 31, 2002, which commits New England Power (the owner of the transmission facilities at the VHS) to keep the Vernon Tie energized during normal utility operations and to make reasonable efforts to keep the line energized during emergency situations, subject to ISO, New England Power Pool (NEPOOL) and REMVEC requirements. TransCanada has affirmed that they are committed under tariff to provide black-start capability of the VHS to ISO-NE.

Both the NEPOOL and REMVEC procedures state that "the most critical power requirement after a blackout is the assurance of reliable shutdowns of nuclear generators, and that expeditious restoration of alternative off-site power sources to nuclear units is imperative to promote the continued reliability of shutdown operations."

As a backup to local indication available to grid operators of a regional blackout, VYNPS procedure OT-3122, "Loss of Normal Power" Appendix A "Station Blackout Procedure" directs operators to immediately contact the regional grid control center to initiate a black start of the VHS if the Vernon Tie is unavailable due to a regional grid blackout. The regional grid control center actions are directed by Operating Procedure OP-6, "System Restoration." This directs certain hydro-station operators (including the VHS) to initiate black start procedures, and upon notification that the units are started, provide instructions to align power to VYNPS and to communicate when these actions are complete to the VYNPS control room. The owner of the VHS has a procedure for the actual black start. The above response adequately addressed the NRC staff's concern.

Condensate Inventory and Reactor Coolant System Inventory

The licensee has performed a plant-specific analysis that shows that injecting 75,000 gallons of condensate storage tank (CST) inventory is sufficient to remove decay heat, provide leakage makeup, and depressurize the reactor to 100 psia. A leakage rate of 61 gpm was used which is comprised of recirculation pump leakage of 18 gpm per pump (36 gpm total) and the TS RCS leakage limit of 25 gpm. The leakage rate is maintained at 61 gpm during depressurization.

The licensee used the NUMARC 87-00, Section 7.2.1, methodology for determining condensate inventory for the 2-hour coping duration, and it was determined to be 47,161 gallons. The licensee has further applied the NUMARC 87-00, Section 7.2.1, methodology for determining condensate inventory out to the time period where the reactor is depressurized to 100 psia where low pressure CS pumps can provide injection with the torus as a water source. This time period is approximately 5 hours, and the total condensate inventory requirement is 86,439 gallons. To ensure that at least 100,000 gallons of usable CST inventory is available for 5 hours injection during an SBO, the minimum administrative limit for CST level identified in plant procedure OP 0150 will be increased. This administrative limit accounts for required instrument uncertainty.

The NRC staff concludes that the ability to maintain adequate RCS inventory to ensure that the core is cooled has been assessed for the required coping duration by plant-specific analysis and the NUMARC methodology. The expected rates of reactor coolant inventory loss under SBO conditions will not result in core uncovery in an SBO event of the required 2-hour coping

duration. The AAC power source will be available within 2 hours to the necessary makeup systems. Therefore, makeup systems available under SBO conditions are adequate to maintain core cooling for the entire SBO duration at EPU. Additionally, the torus remains available to supplement the CST for low pressure systems once the reactor is depressurized.

Battery Capacity

The licensee stated that worst-case battery scenario occurs when dc buses DC-1 and DC-2 are cross-tied and an SBO occurs just after the largest DC MOV (V23-19) is started for testing. A separate scenario has been added to calculation VYC-2154 to show that the battery has capacity to handle the SBO required loads for the full 2 hours. The calculations for battery capacity used the lowest electrolyte temperature (60°F) anticipated, design margin of 1.1, and an aging factor of 1.25 as recommended by Institute of Electrical and Electronics Engineers, Inc. (IEEE) Standard 485 and as recommended by NUMARC 87-00. The battery terminal voltage profile corresponding to the duty cycle was calculated to verify that minimum voltages reached during the duty cycle were higher than the minimum voltages required for operation of the dc loads. The licensee concluded that the 125 Vdc station batteries have adequate capacity to supply all SBO loads for 2 hours with no manual load stripping.

In a letter dated July 27, 2005 (Reference 61, RAI EEIB-A-4), the NRC staff requested that the licensee verify that sufficient DC power is available, under worst-case conditions during the 2-hour coping period, to close the 4160-volt breakers associated with the AAC power source. In its response dated August 1, 2005 (Reference 31), the licensee stated that a review of the worst-case scenario has been performed by adding the 6 ampere (amp) load of the Vernon Tie breaker closing for a full minute, for conservatism, to the end of the 2-hour duty cycle. The review indicated that there is no impact on the battery capacity or end voltage. Based on the licensee's response, 6 amps are needed to close one breaker. However, two breakers are involved for the AAC power source. Additionally, the spring charging current after the breakers are closed will be much higher. In a letter dated September 7, 2005 (Reference 62, RAI EEIB-A-6), the NRC staff asked the licensee why the spring-charging current was not considered in the battery capacity and voltage calculations, and whether there are any other loads not currently considered in the battery calculations. In its response dated September 10, 2005 (Reference 33), the licensee stated that two 4160-volt breakers are involved in aligning power from the AAC power source, and the spring-charging motor current should also be included. An evaluation was performed using a 20-amp load applied for a full minute at the end of the 2-hour duty cycle, instead of the original 6 amps. The breaker closing current for each 4160-volt breaker is 6.0 amps. The breaker spring charging motor draws 10 amps, but this draw is not concurrent with the closing current. Therefore, the additional 20-amp load is conservative. The evaluation confirms that the additional load has no effect on end voltage and does not change the required battery capacity. Additionally, the licensee stated that all other battery loads that occur during the 2-hour coping period are currently considered in the coping analysis calculation.

Based on the above, the NRC staff concludes that Class 1E batteries have adequate capacity to meet the SBO loads for 2 hours. The battery chargers will be available within 2 hours. Therefore, adequate battery capacity will be available to meet the SBO loads at EPU conditions.

Loss of Ventilation

By letters dated March 24 and August 1, 2005 (References 26 and 31), the licensee provided additional information including analyses addressing SBO and 10 CFR Part 50 Appendix R fire events. In accordance with the licensee's March 24, 2005 letter, for EPU conditions, the duration for which VYNPS must cope until the AAC source is available is 2 hours. The licensee evaluated the effect of a loss of ventilation on equipment operability for the 2-hour coping period. For an SBO, the ventilation areas of interest are the RCIC room, HPCI room, main steam tunnel, control room, switchgear room, and intake structure. A summary of the licensee's evaluation follows.

The licensee stated that the RCIC and HPCI rooms were evaluated for heat-up during an SBO by calculation VYC-0886. Since the heat loads in these rooms do not change for the EPU, the analyses remain valid. The steady-state temperature in both the RCIC and HPCI rooms is less than 150°F. The licensee stated that RCIC and HPCI operability at steady state has been evaluated in accordance with NUMARC 87-00, Appendix F, methodologies and the equipment is operable for the required duration.

Calculation VYC-1347 provides a plant-specific heat-up analysis for the main steam tunnel following a loss of ventilation for the current power level. It concludes that automatic isolation of RCIC or HPCI due to room heat-up would occur approximately 18 hours after a loss of ventilation occurs. Therefore, automatic isolation will not occur during an SBO event. Calculation VYC-2279 is an analysis of the impact of the EPU on main steam tunnel ambient temperature. The impact is less than 1°F. The increase in steam tunnel heat load due to higher feedwater temperature is approximately 2.4%. The licensee stated that this higher heat load will result in a slightly quicker isolation, which is still hours after the required SBO coping duration.

Calculation VYC-1502 provides a plant-specific heat-up calculation of the control room following an Appendix R fire which causes a loss of ventilation. The analysis shows that the control room could reach approximately 120°F at 4 hours given the Appendix R control room heat load. The Appendix R heat load bounds the SBO heat load. The licensee stated that this analysis is based on removing acoustic ceiling tiles. Removing acoustic ceiling tiles is addressed in operating procedure OP-2192, Revision 31. Other actions (addressed in OP-2192) such as opening panel doors, opening the control room doors and providing temporary ventilation can also be used to reduce control room temperature. Normal control room HVAC is available when the AAC source becomes available at 2 hours.

The licensee stated that the heat load in the switchgear room is unaffected by the EPU. The licensee also stated that heat-up of the intake structure on loss of ventilation with two service water pumps available is unaffected by EPU since the heat loads in the intake structure are unaffected by EPU.

Based on review of the information provided in the licensee's letters dated March 24 and August 1, 2005, the NRC staff concludes that the licensee has adequately addressed a loss of ventilation during an SBO event and that equipment operability will be maintained at EPU conditions.

Containment Isolation

The licensee's coping analysis provided in its letter dated March 24, 2005 (Reference 26) included an evaluation of containment isolation capability during an SBO event. The evaluation was performed using the guidance in Section 7.2.5 of NUMARC 87-00.

The evaluation states that since air-operated valves fail in their required position on loss of air or power, an SBO event will result in air-operated containment isolation valves closing. Therefore, air-operated valves do not need further evaluation.

Since motor-operated valves (MOVs) fail as-is on a loss of power, MOVs that are designated as containment isolation valves need to be evaluated. The licensee reviewed the MOVs associated with the primary containment penetrations listed in UFSAR Table 5.2.2. Using the NUMARC guidance, the licensee determined that each of the affected MOVs was acceptable based on meeting one of the following criteria:

- MOV is smaller than the 3" diameter specified in the NUMARC guidance;
- MOV is in a non-radioactive line that is not expected to be breached;
- MOV has a counterpart DC-powered valve that provides isolation of the line;
- MOV does not receive an isolation signal and is not expected to provide containment isolation; or
- MOV is normally closed.

Based on review of the licensee's coping analysis, the NRC staff concludes that the containment isolation capability following an SBO event is acceptable at EPU conditions.

Compressed Air/Gas Systems

In a letter dated March 24, 2005 (Reference 26), the licensee stated that the coping analysis is based on using the HPCI system as the high pressure makeup source. HPCI system operation is independent of the compressed air system. VYNPS only relies on those air operated valves which fail to their required position on loss of air or power during an SBO event.

Additionally, the licensee stated that the safety relief valves (SRVs) which are used to depressurize the reactor are provided with nitrogen accumulators. Additionally, a backup nitrogen supply system was installed to support manual operation of the SRVs for 72 hours. The nitrogen supply provides SRV operation capability well in excess of that needed for an SBO event. The backup system automatically (via a check valve) provides makeup to the SRV nitrogen accumulators. Following restoration of AC power, an instrument air compressor will be powered by the AAC power source.

As described in a VYNPS letter dated September 10, 2005 (Reference 33, Attachment 8, response to RAI SPLB-A-31), the control room operators will use the SRVs to remove decay heat and depressurize the reactor so that shutdown cooling can be initiated when AC power is restored following the assumed 2-hour coping duration. The reactor is cooled to cold shutdown

conditions in less than 24 hours. SRV operation is no longer needed once RHR shutdown cooling begins and the reactor reaches cold shutdown. Restoration of instrument air provides assurance that compressed air would be available for long-term recovery actions.

Based on the information that was provided, the NRC staff finds that the SBO coping time of 2 hours is acceptable at EPU conditions with respect to compressed air and gas systems.

Torus Temperature

In its letter dated July 2, 2004 (Reference 9), the licensee analyzed the suppression pool temperature following an SBO assuming one RHR service water pump is available to cool the suppression pool water. This necessitated credit for containment accident pressure in determining available NPSH for the RHR pumps.

The licensee's letter dated March 24, 2005 (Reference 26), reports the results of a revised analysis which credits two RHR service water pumps. As a result of this assumption, credit for containment accident pressure is no longer required for adequate NPSH for an SBO event.

The licensee's revised analysis predicts the peak suppression pool temperature following an SBO to be 182.2°F. The previous peak temperature, assuming only one RHR service water pump is available was 187.9°F.

The results of the NRC staff review of the licensee's methods of calculating available NPSH are provided in SE Section 2.6.5.

The NRC staff finds the licensee's SBO analysis to be acceptable with respect to torus temperature and available NPSH for EPU conditions.

Procedures and Training

The licensee's letter dated August 1, 2005 (Reference 31, response to RAI EEIB-A-5), provided information regarding changes required to plant procedures and operator training related to SBO coping. The licensee stated that Operating Procedure OP-2124, "Residual Heat Removal System" has been revised for training to include direction on how and when to place the second RHRSW pump per RHR heat exchanger in service when placing the torus cooling mode of RHR in service. Additionally, Operational Transient Procedure OT-3122, "Loss of Normal Power" has been revised to direct operators to immediately contact the regional grid control center to initiate a black-start of the VHS if the Vernon Tie is unavailable due to a regional grid blackout. In addition, the procedure has been revised for training to commence a cooldown within 1 hour of the SBO event, and when power is restored, to place two RHRSW pumps in service per OP- 2124. Also, a note was added about exceeding the drywell air temperature of 280°F for a short period of time without exceeding the 280°F drywell shell temperature.

In Reference 31, the licensee provided a commitment for providing training on the changes to procedures OP-2124 and OP-3122. The licensee also committed to revise various operating, surveillance and administrative procedures to incorporate a higher condensate storage tank inventory limit as either a precaution or an administrative limit.

On the basis of its review, the NRC staff finds that the licensee had adequately addressed procedures and training related to SBO coping.

Licensee Commitments

The licensee provided two regulatory commitments related to SBO. These commitments are shown as items 21 and 22 in the table in SE Section 4.0.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the plant's ability to cope with, and recover from, an SBO event for the period of time established in the plant's licensing basis. The NRC engineering inspection team found that should an SBO occur at the plant as a result of a regional grid disturbance, AAC power would not be restored from the VHS within the 10-minute timeframe. VYNPS is no longer a 10-minute AAC plant. The AAC power source will be available at VYNPS within 2 hours. The plant can cope for 2 hours without an AAC power source and the remaining 6 hours with an AAC power source. The NRC staff concludes that the licensee has adequately evaluated the effects of the proposed EPU on SBO and has demonstrated that the plant will meet the requirements of 10 CFR 50.63. Therefore, the NRC staff finds the proposed EPU is acceptable with respect to SBO.

2.4 Instrumentation and Controls

2.4.1 Reactor Protection, Safety Features Actuation, and Control Systems

Regulatory Evaluation

Instrumentation and control systems are provided (1) to control plant processes having a significant impact on plant safety, (2) to initiate the reactivity control system (including control rods), (3) to initiate the engineered safety features (ESF) systems and essential auxiliary supporting systems, and (4) for use to achieve and maintain a safe shutdown condition of the plant. Diverse instrumentation and control systems and equipment are provided for the express purpose of protecting against potential common-mode failures of instrumentation and control protection systems. The NRC staff conducted a review of the reactor trip system, engineered safety feature actuation system (ESFAS), safe shutdown systems, control systems, and diverse instrumentation and control systems for the proposed EPU to ensure that the systems and any changes necessary for the proposed EPU are adequately designed such that the systems continue to meet their safety functions. The NRC staff's review was also conducted to ensure that failures of the systems do not affect safety functions. The NRC's acceptance criteria related to the quality of design of protection and control systems are based on 10 CFR 50.55a(a)(1), 10 CFR 50.55a(h), and draft GDC -1, 11, 12, 14, 15, 19, 20, 22, 23, 25, 26, 40, and 42. Specific review criteria are contained in SRP Sections 7.0, 7.2, 7.3, 7.4, 7.7, and 7.8.

Technical Evaluation

As discussed in PUSAR Section 5, the licensee evaluated instrumentation in the nuclear steam supply system (NSSS) and balance-of-plant (BOP) systems to determine its suitability for the

revised operating ranges of the affected process parameters at EPU conditions. Where necessary, the licensee revised the calibration and/or the setpoint and uncertainty calculations for the affected instruments. As discussed in Reference 6, there are no changes to instrument control philosophy as a result of EPU except for the new recirculation runback logic. That change is evaluated in SE Section 2.5.4.4. The proposed EPU does not change the safety functions or design basis of the VYNPS instrumentation with respect to separation, redundancy, or diversity.

The licensee's evaluation of the suitability of the existing instruments for EPU operation resulted in the following changes:

Parameter	Change
Main Steam Line (MSL) High Flow	Respan transmitters to cover new 140% steam flow value.
MSL High Flow	Replace 4 of the transmitters used to provide the 40% setpoint with more accurate transmitters. The setpoint remains at 40% of CLTP.
MSL High Flow	Setpoint changes for new setpoint for 140% isolation at new steam flows.
MSL High Flow	Install new indicators on the master trip units.
Neutron Monitoring	Average Power Range Monitor (APRM) flow-biased scram analytical limits (ALs) and rod block limits will be changed for the EPU.
Neutron Monitoring	APRM will be re-calibrated to reflect EPU operation.
Neutron Monitoring	Rod Block Monitors (RBM) will be re-calibrated to reflect EPU operation.
MSL Radiation Monitor	Normal setpoint changes based on new 100% MSL radiation level.
Feedwater Control (FWC) System, Feed Flow	Respan transmitters for EPU flows.
FWC System, Feed Flow	New indicator/recorder ranges for EPU flows.
FWC System, Steam Flow	Respan transmitters for EPU flows.
FWC System, Steam Flow	New indicator/recorder ranges for EPU flows.
Rod Worth Minimizer	Change the setpoint to maintain the setpoint at the same absolute value of steam flow because of the range changes of the associated instruments.
Recirculation Pump Net Positive Suction Head (NPSH) trip	Change setpoint to maintain the setpoint at the same absolute value of steam flow because of the range changes of the associated instruments.

Turbine First Stage Pressure	Setpoint change for the scram bypass.
Turbine Control System	Operating setpoint change to address increased steam line differential pressure.
Condensate Flow	Respan transmitters for EPU flows.
Condensate Flow	Computer point respan.
Condensate Heater Pressure Low	Setpoint change.
Condensate Flow to Oxygen Injection System	Instrument recalibration.
Steam Line Leak Alarm Module	Recalibration of transmitter and alarm module.
Condensate Pump Discharge Pressure	Indicator rebanding for new normal pressure.
Feedwater Pump Suction Pressure	Instrument recalibration.
Feed Pump Low Suction Pressure trip	Setpoint change for low-pressure pump trip.
Feed Pump Low Suction Pressure	Add a second pressure switch to each pump to provide signal for recirculation runback on loss of condensate pump.
Recirculation Motor Generator Control	New runback to reduce reactor power on loss of feedwater or condensate pump.

Since the instrumentation and control functions related to the above changes will be confirmed during post-modification testing, power ascension testing, instrument calibration, and TS surveillance testing, as applicable, the NRC staff has reasonable assurance that the instrumentation and related systems will continue to perform their intended safety functions at EPU conditions.

Conclusion

The NRC staff has reviewed the licensee's application related to the effects of the proposed EPU on the functional design of the reactor trip system, ESFAS, safe shutdown system, and control systems. The NRC staff concludes that the licensee has adequately addressed the effects of the proposed EPU on these systems and that the changes that are necessary to achieve the proposed EPU are consistent with the plant's design basis. The NRC staff further concludes that the systems will continue to meet the requirements of 10 CFR 50.55a(a)(1), 10 CFR 50.55(a)(h), and draft GDC-1, 11, 12, 14, 15, 19, 20, 22, 23, 25, 26, 40, and 42. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to instrumentation and controls.

2.5 Plant Systems

2.5.1 Internal Hazards

2.5.1.1 Flooding

2.5.1.1.1 Flood Protection

For proposed EPUs, the NRC staff reviews flood protection measures to ensure that SSCs important to safety are adequately protected from the consequences of internal flooding that result from postulated failures of tanks and vessels. Because the staff's review focuses on increases of fluid volumes in tanks and vessels that result due to a proposed EPU, and the licensee indicated that no such changes are being made at VYNPS, an evaluation of this particular area by the staff is not required (Reference 9, Attachment 1, response to RAI SPLB-A-2).

2.5.1.1.2 Equipment and Floor Drains

The function of the equipment and floor drainage system (EFDS) is to assure that waste liquids, valve and pump leak-offs, and tank drains are directed to the proper area for processing or disposal while preventing a backflow of water that might result from maximum flood levels to areas of the plant containing equipment that is important to safety. Because the sources and quantities of liquids that enter the equipment and floor drains are not appreciably affected by the proposed EPU and postulated flood levels will not increase, an evaluation of the EFDS is not required.

2.5.1.1.3 Circulating Water System

The circulating water system (CWS) provides a continuous supply of cooling water to the main condenser to remove excess heat from the turbine cycle and auxiliary systems. The NRC staff's review of the CWS focuses on the impact that the proposed EPU will have on existing flooding analyses due to any increases that may be necessary in fluid volumes and installation of larger capacity CWS pumps or piping. Because the impact of the proposed EPU on the licensee's flooding analysis is considered in SE Sections 2.5.1.1.1 and 2.5.1.3, a separate evaluation for the CWS in this section is not required.

2.5.1.2 Missile Protection

2.5.1.2.1 Internally Generated Missiles

Regulatory Evaluation

The NRC staff's review concerns the protection of SSCs important to safety from missiles that could result from in-plant component overspeed conditions and high-pressure system ruptures. Potential missile sources include pressurized systems and components, and high-speed rotating machinery. The purpose of the staff's review is to confirm that: (1) SSCs important to safety are protected from internally generated missiles, and (2) the failure of SSCs not important to safety due to missiles will not pose a challenge to SSCs that are important to safety. The staff's review focuses on system modifications and increases in system pressures

that are necessary for the proposed EPU, and component overspeed considerations that may affect the impact that missiles could have on SSCs important to safety. The criteria that are most applicable to the staff's review of the protection of SSCs important to safety from the effects of internally generated missiles for VYNPS are based on draft GDC-40, "Missile Protection (Category A)," and other licensing-basis considerations that are applicable. The staff's review related to internally generated missiles is performed in accordance with the guidance provided in Section 2.1 of RS-001, Matrix 5, and acceptability for EPU operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in Chapters 5, 6, 10, 12, and Appendix B of the VYNPS UFSAR, except where proposed changes are found to be acceptable based upon the specified review criteria.

Technical Evaluation

The licensee evaluated the impact of the proposed EPU on SSCs important to safety due to internally generated missiles that may result from failures in high energy systems and overspeed of rotating equipment (Reference 9, Attachment 1, response to RAI SPLB-A-1). The licensee's evaluation included the effects of missiles due to new or modified equipment for the new pressures, flow rates, and fluid velocities that will exist at the proposed EPU conditions (Reference 29, Attachment 1, response to RAI SPLB-A-23). The new high-pressure turbine will result in higher extraction steam pressures and flow rates. However, there are no SSCs important to safety located in the vicinity of these extraction steam lines. Also, the licensee determined that the estimated main turbine peak overspeed condition that can result due to the proposed EPU conditions is less than what is expected for operation at the current licensed power level with the original high and low pressure turbine rotors (discussed in SE Section 2.5.1.2.2, below) and therefore, the impact of missiles from the main turbine on equipment important to safety will not be affected by the proposed EPU.

The design of the four new high pressure feedwater heaters incorporated enhancements in design margin including increased shell side pressure ratings. The feedwater piping system was evaluated for changes in operating parameters (i.e., pressure and flow rates) that would result from EPU conditions. The design pressure will not be increasing as a result of EPU. The increase in flow rates was assessed for potential impact on flow-induced fluid transient loads in the feedwater piping system, and was found acceptable since the system does not contain any fast closing valves. Hence, no new missile concerns for the feedwater system will result due to the implementation of the EPU.

A detailed evaluation of the main steam system was performed by the licensee to assess the higher system flow rate and its impact on turbine stop valve closure event transient loads. No new postulated pipe break locations were identified, hence, no new missile concerns for the main steam system are present as a result of the EPU implementation. For the remaining piping systems which will experience flow rate increases, it was concluded that no new missile concerns will result due to EPU implementation.

The licensee determined that the proposed EPU will not result in any increases in system pressures or changes in existing system or equipment configurations from what was previously considered. Consequently, the EPU will not affect the impact of internally generated missiles (outside containment) on SSCs important to safety.

Based on a review of the information that was submitted, the NRC staff is satisfied that the licensee has adequately evaluated and addressed the potential impact of the proposed EPU on existing considerations and features that are credited for protecting equipment important to safety from the effects of internally generated missiles. The licensee has determined that the EPU will not cause the effects of internally generated missiles (outside containment) on SSCs important to safety to be more severe than previously assumed and therefore, the staff agrees that SSCs important to safety will continue to be adequately protected from internally generated missiles following EPU implementation.

Conclusion

The NRC staff has reviewed the licensee's assessment of changes in system pressures, configurations, and equipment rotational speeds necessary to support the proposed EPU and finds that SSCs important to safety will continue to be protected from the effects of internally generated missiles in accordance with licensing-basis assumptions. Therefore, the proposed EPU is considered to be acceptable with respect to the protection of SSCs important to safety from internally generated missiles.

2.5.1.2.2 Turbine Generator

Regulatory Evaluation

The large steam turbines of the main turbine-generator set have the potential for producing high-energy missiles. The NRC staff's review of the turbine generator focuses on the impact of the proposed EPU on the overspeed protection features of the main turbine to ensure that adequate turbine overspeed protection will be maintained. The criteria that are most applicable to the staff's review of the turbine generator are based on draft GDC-40, "Missile Protection (Category A)," in that engineered safety features are expected to be protected from the effects of turbine missiles; and other licensing-basis considerations that are applicable. The staff's review of the turbine generator is performed in accordance with the guidance provided in Section 2.1 of RS-001, Matrix 5, and acceptability for EPU operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in Sections 7.11 and 11.2 of the VYNPS UFSAR, except where proposed changes are found to be acceptable based upon the specified review criteria.

Technical Evaluation

The high pressure (HP) turbine at VYNPS has been redesigned with a new rotor, diaphragms, and buckets to increase its flow capacity for the proposed EPU operations. Also, prior to the HP turbine conversion to support EPU, the low pressure (LP) turbine rotors at VYNPS were converted to the monoblock design from the original built-up construction. This conversion to monoblock increased the rotor inertia which slows the acceleration rate of the machine should a load rejection event occur.

As discussed in Reference 9 (Attachment 1, response to RAI SPLB-A-3, Page 137 of 189 and response to RAI SPLB-A-6) and Reference 24 (Attachment 1, response to RAI SPLB-A-12), the main turbine is provided with two overspeed protection trip settings: one is the minimum mechanical trip setting for normal overspeed (NOS), and the other is the maximum trip setting for emergency overspeed (EOS). The original design setting of the minimum mechanical trip

for NOS was 110.5% - 111.5% of the rated speed. With the modifications to the HP turbine for the EPU conditions and with the LP turbine monoblock rotors, the peak speed following a full load rejection is estimated to be 109.6%, which is below the current minimum mechanical setting. At NOS, it is assumed that all protective steam valves and control systems have responded as intended to minimize the resulting peak speed when a load rejection occurs. At EOS, following a full load rejection, it is assumed that the first line-of-defense valves and speed control systems completely fail. For this condition, the unit would rapidly accelerate to the mechanical trip speed range, which would activate the trip function and close the main and intermediate stop valves. According to GE, the limit for EOS is 120% of the rated speed and the estimated EOS value for the VYNPS EPU is 119.2% of the rated speed, which is below the limit of 120% limit. Therefore, because the acceptance criteria for NOS and EOS will continue to be satisfied following EPU implementation, the licensee has concluded that no changes are required for the NOS and EOS turbine trip setpoints, which the NRC staff finds acceptable.

Based on a review of the information that was submitted, the NRC staff is satisfied that the licensee has adequately evaluated and addressed the potential impact of the proposed EPU on the capability to prevent turbine overspeed. The licensee has determined that the existing turbine trip setpoints for NOS and EOS conditions will continue to prevent turbine overspeed consistent with the turbine design criteria and therefore, the staff agrees that the EPU will not increase the likelihood that turbine missiles will be generated due to an increased likelihood of turbine overspeed.

Conclusion

The NRC staff has reviewed the licensee's assessment of changes being made to the high pressure turbine, steam mass flow rate, and other operational characteristics necessary to support the proposed EPU and finds that existing design features will continue to protect the main turbine from overspeed conditions following postulated transient and accident conditions in accordance with licensing-basis assumptions. Therefore, the proposed EPU is considered to be acceptable with respect to the main turbine.

2.5.1.3 Pipe Failures

Because (1) the reactor dome and system pressures that were used in the existing high energy line break analyses continue to bound EPU conditions, (2) no new high energy line break locations are postulated, and (3) flooding due to postulated pipe breaks continues to be bounded by postulated failures in lower energy piping systems, the protection of SSCs important to safety from the effects of postulated pipe failures is not affected by the proposed EPU (Reference 9, Attachment 1, response to RAI SPLB-A-5 and Reference 24, Attachment 1, response to RAI SPLB-A-13). Therefore, NRC evaluation of this area is not required. Note that the effects of pipe break on environmental qualification is considered in SE Sections 2.3.1.

2.5.1.4 Fire Protection

Regulatory Evaluation

The purpose of the fire protection program (FPP) is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases to the environment. The NRC staff's review focused on the effects of the increased decay heat on the plant's safe shutdown analysis to ensure that SSCs required for the safe shutdown of the plant are protected from the effects of the fire and will continue to be able to achieve and maintain safe shutdown following a fire. The NRC's acceptance criteria for the FPP are based on (1) 10 CFR 50.48 and associated Appendix R to 10 CFR Part 50, insofar as they require the development of an FPP to ensure, among other things, the capability to safely shut down the plant; and (2) draft GDC-3, insofar as it requires that the reactor facility be designed (a) to minimize the probability of events, such as fire and explosions, and (b) to minimize the potential effects of such events to safety. Specific review criteria are contained in SRP Section 9.5.1, as supplemented by the guidance provided in Attachment 2 to Matrix 5 of Section 2.1 of RS-001.

Technical Evaluation

RS-001, Attachment 1 to Matrix 5, "Supplemental Fire Protection Review Criteria," states that "... power uprates typically result in increases in decay heat generation following plant trips. These increases in decay heat usually do not affect the elements of a fire protection program related to (1) administrative controls, (2) fire suppression and detection systems, (3) fire barriers, (4) fire protection responsibilities of plant personnel, and (5) procedures and resources necessary for the repair of systems required to achieve and maintain cold shutdown. In addition, an increase in decay heat will usually not result in an increase in the potential for a radiological release resulting from a fire...[W]here licensees rely on less than full capability systems for fire events ..., the licensee should provide specific analyses for fire events that demonstrate that (1) fuel integrity is maintained by demonstrating that the fuel design limits are not exceeded and (2) there are no adverse consequences on the reactor pressure vessel integrity or the attached piping. Plants that rely on alternative/dedicated or backup shutdown capability for post-fire safe shutdown should analyze the impact of the power uprate on the alternative/dedicated or backup shutdown capability...The licensee should identify the impact of the power uprate on the plant's post-fire safe shutdown procedures." Section 6.7, "Fire Protection," of Attachment 4 to the licensee's application (Reference 1) satisfactorily addresses these Fire Protection requirements of RS-001. In addition, the licensee's application states that "a plant-specific evaluation was performed to demonstrate safe shutdown capability in compliance with the requirements of 10 CFR 50 Appendix R assuming CPPU conditions. The results of the Appendix R evaluation for CPPU provided in Table 6-5 demonstrate that fuel cladding integrity and containment integrity are maintained and that sufficient time is available for the operator to perform the necessary actions." The information provided in this table, as supplemented by information in Reference 6, Attachment 2, satisfactorily demonstrates the licensee's compliance.

Conclusion

The NRC staff has reviewed the licensee's fire-related safe shutdown assessment and concludes that the licensee has adequately accounted for the effects of the increased decay heat on the ability of the required systems to achieve and maintain safe shutdown conditions. The NRC staff further concludes that the FPP will continue to meet the requirements of 10 CFR 50.48, Appendix R to 10 CFR Part 50, and draft GDC-3 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to fire protection.

2.5.2 Fission Product Control

2.5.2.1 Fission Product Control Systems and Structures

The purpose of the NRC staff's review of fission product control systems and structures is to confirm that modeling assumptions, analytical methodologies, limits associated with key parameters, and the assumed capability of ventilation systems to limit fission product releases are valid for design-basis loss-of-coolant accidents. Consequently, the staff's review focuses primarily on any adverse effects that the EPU might have in these areas. Because SE Sections 2.7 and 2.9 encompass these areas of review, a separate evaluation in this section is not required.

2.5.2.2 Main Condenser Evacuation System

The main condenser evacuation system (MCES) is not impacted by the proposed EPU because the condenser air removal requirements are not affected. The MCES is sized based upon the volume of the condenser and desired evacuation time, neither of which is impacted by the proposed EPU. Consequently, the existing effluent holdup time and monitoring capability are not affected by the proposed EPU and therefore, NRC review of the MCES is not required.

2.5.2.3 Turbine Gland Sealing System

The turbine gland sealing system (TGSS) is designed to provide sealing steam for the main turbine shafts to prevent the leakage of air into the turbine casing and the escape of steam into the turbine building, thereby preventing the uncontrolled release of radioactive material from steam in the turbine to the environment. Because no modifications are being made to the TGSS and non-condensable gases will continue to be monitored for radiation, the function of the TGSS will not be impacted by the proposed EPU and therefore an evaluation of the TGSS is not required (Reference 9, Attachment 1, response to RAI SPLB-A-4).

2.5.2.4 Main Steam Isolation Valve Leakage Control System

Because VYNPS does not have a main steam isolation valve leakage control system, this review section is not applicable.

2.5.3 Component Cooling and Decay Heat Removal

2.5.3.1 Spent Fuel Pool Cooling and Cleanup System

Regulatory Evaluation

The spent fuel pool (SFP) provides wet storage of spent fuel assemblies. The SFP cooling and cleanup system (SFPCCS) consists of a fuel pool cooling and demineralizer system (FPCDS) and the residual heat removal (RHR) system augmented fuel pool cooling (FPC) mode. The FPCDS consists of two subsystems: a non-safety normal fuel pool cooling subsystem (NFPCS) and a safety-related standby fuel pool cooling subsystem (SFPCS). The safety function of the SFPCS is to cool the spent fuel assemblies and keep the spent fuel assemblies sufficiently covered with water during all storage conditions. The NRC's review focuses on the impact that proposed EPUs have on the capability of the SFPCS to provide adequate cooling of the spent fuel. The criteria that are most applicable to the staff's review of the SFPCCS are based primarily on draft GDC-67, "Fuel and Waste Storage Decay Heat (Category B)," insofar that reliable decay heat removal systems are necessary to prevent damage to stored spent fuel; and other licensing-basis considerations that are applicable. The staff's review of the SFPCCS is performed in accordance with the guidance provided in Section 2.1 of RS-001, Matrix 5, and acceptability for EPU operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in Section 10.5 of the VYNPS UFSAR, except where proposed changes are found to be acceptable based upon the specified review criteria.

Technical Evaluation

As described in VYNPS UFSAR (Reference 50), Section 10.5.3 and 10.5.5, the current licensing basis for the VYNPS fuel pool cooling system is to maintain the SFP bulk water temperature below the limit of 150°F for a normal batch off-load (nominal one-third core discharge) and for abnormal (full core offload) conditions. Also, an administrative limit of 125°F has been established as the maximum fuel pool temperature during normal cooling and filtering.

In order to assure adequate SFP cooling for EPU conditions, the licensee performed heat load analyses for various fuel off-load scenarios. In its submittal (Reference 1, Attachment 6, Section 6.3.1 and Table 6-3), the licensee presented five configurations demonstrating that certain (not necessarily safety-related) systems have adequate SFP cooling capability for the fuel off-loads identified under each of those five configurations. However, the NRC staff raised a concern that these configurations did not address the capability of the safety-related SFPCS for the following limiting cases of core offload as described in UFSAR Section 10.5.5, Page 10.5-9, third paragraph:

- Limiting Normal Batch (one-third core): One train (i.e., one heat exchanger and one pump) of SFPCS in service, and

- Limiting Full Core Offload: Both trains (i.e., two heat exchangers and two pumps) of SFPCS in service.

In response to the concern raised by the NRC staff regarding the capability of the safety-related SFPCS to perform its function, the licensee provided additional information in a letter dated July 30, 2004 (Reference 11, Attachment 1, response to RAI SPLB-A-7), as supplemented by letters dated February 24, 2005 (Reference 24, Attachment 1, response to RAI SPLB-A-11), and April 22, 2005 (Reference 29, Attachment 1, response to RAI SPLB-A-14). As discussed in the response to RAI SPLB-A-11, the licensee performed fuel pool heat-up calculations for the above two limiting cases at EPU conditions, assuming no credit for the NFPCS. The licensee's calculations also assumed that the initial fuel pool temperature is at the UFSAR administrative limit (i.e., 125°F) and remains at that temperature until the fuel pool gates are installed (6 days for batch offload and 10 days for full core offload cases). The licensee stated that this assumption is conservative because the VYNPS administrative procedures preclude installation of the fuel pool gates if the fuel pool temperature is above the administrative limit. The results of the licensee's analysis indicated that, 1) peak bulk pool temperature for partial core off-load (consisting of a 136 bundle batch) will be 140.6°F at 7.5 days after shutdown (i.e., 1.5 days after the SFP gates are installed), and 2) peak bulk pool temperature for full core off-load will be 145.7°F at 11 days after shutdown (i.e., 1 day after the SFP gates are installed). Thus, with the existing administrative controls in place, the bulk pool temperature will be maintained below 150°F for both scenarios. Also, according to the response to RAI SPLB-A-14, the licensee determined that the estimated SFP heat load following a batch off-load (i.e., 136 bundles, while completely filling the pool with 3,353 spent fuel assemblies from the last normal discharge), 6 days after plant shutdown at EPU conditions is 10.46 MBtu/hr. Similarly, the licensee determined that the SFP heat load for the full core off-load scenario (i.e., 368 bundles, while completely filling the pool with 3,353 spent fuel assemblies from the last normal discharge), 10 days after plant shutdown at EPU conditions, is 21.78 MBtu/hr. This compares with the current design capacity of 11 MBtu/hr for one train of the SFPCS for the batch off-load, and 22 MBtu/hr for two trains of the SFPCS for the full core off-load (VYNPS UFSAR, Table 10.5.3). Thus, the estimated SFP heat loads for the batch and full core off-loads will remain within the design capacity of the SFPCS following EPU implementation. The NRC staff reviewed the information that was submitted and found the current SFPCS acceptable to cool the spent fuel for the limiting off-load conditions during EPU operation.

Further, as discussed in Reference 11, response to RAI SPLB-A-7, and Table SPLB-A-7-1, additional information regarding the inputs and assumptions that were used in the SFP heat removal analyses for EPU were provided. The licensee indicated that the analyses are based on the most limiting service water/ultimate heat sink temperature, cooling system flow rates, and heat exchanger performance (i.e., the fouling and tube plugging factors). Also, the licensee stated that in the unlikely event of a complete loss of SFP cooling, it would take at least 6 hours for the SFP to begin to boil in the worst-case scenario, after completing the limiting full core offload (Reference 31, Attachment 8, response to RAI SPLB-A-25). The licensee determined that the worst case boil-off rate would be about 90 gpm, and that this rate is within the existing 250 gpm Seismic Category 1 emergency makeup capability that is available.

Based on a review of the information that was submitted, the NRC staff is satisfied that the licensee has adequately evaluated and addressed the potential impact of the proposed EPU on the capability of the SFPCS to adequately cool the spent fuel. The licensee has determined

that the existing design capacity of the SFPCS will continue to exceed the SFP heat load that results from EPU operation and the time to boil following a loss of SFP cooling for the full core offload case will continue to afford plant operators sufficient time to take corrective actions. Therefore, the staff agrees that the design-basis capability of the SFPCS will be maintained following the proposed EPU.

Conclusion

The NRC staff has reviewed the licensee's assessment related to the SFPCCS and finds that the SFPCCS will continue to provide sufficient SFP cooling and that the SFP makeup capability will continue to be adequate in accordance with licensing-basis considerations. Therefore, the proposed EPU is considered to be acceptable with respect to the SFPCCS and associated SFP makeup capability.

2.5.3.2 Station Service Water System

Regulatory Evaluation

The VYNPS service water system (SWS) is a safety-related system that supplies cooling water from the Connecticut River to essential and non-essential components of both primary and secondary plant systems. The residual heat removal service water system (RHRSWS) is one of the heat loads serviced by the SWS and it is included within the scope of this evaluation. VYNPS also has an alternate cooling system (ACS) that provides an alternate means of cooling in the unlikely event that the service water pumps become inoperable (see SE Section 2.5.3.3 for the NRC staff's evaluation of the ACS). The safety objective of the SWS is to provide cooling water to systems and components that are credited for accident mitigation. The NRC staff's review focused on the impact that the proposed EPU will have on the capability of the SWS to perform its safety functions. The criteria most applicable to the staff's review are based primarily on draft GDC-41, "Engineered Safety Features Performance Capability (Category A)," insofar that the SWS is relied upon by engineered safety features for performing their safety functions; draft GDC-44, "Emergency Core Cooling System Capability (Category A)," insofar that the SWS is relied upon by emergency core cooling systems for performing their safety functions; draft GDC-52, "Containment Heat Removal Systems (Category A)," insofar that the SWS is relied upon by containment heat removal systems for performing their safety functions; draft GDC-67, "Fuel and Waste Storage Decay Heat (Category B)," insofar that the SWS is relied upon by fuel and waste storage decay heat removal systems for performing their functions; and other licensing-basis considerations that are applicable. The staff's review of the SWS was performed in accordance with the guidance provided in Section 2.1 of RS-001, Matrix 5, and acceptability for EPU operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in Section 10.6 of the VYNPS UFSAR, except where proposed changes are found to be acceptable based upon the specified review criteria.

Technical Evaluation

The licensee has evaluated the impact of the proposed EPU on the capability of the SWS (including the RHRSWS) to perform its safety functions (Reference 11, Attachment 1, response to RAI SPLB-A-8). Except for the SBO and Appendix R events, the licensee's analyses for EPU operation use the same SWS flow rates that are credited for the current licensed power level and therefore, no system modifications are required. As discussed in Reference 26 and in SE Section 2.6.5, the licensee originally credited one RHRSW pump for the SBO and Appendix R analyses. For the EPU, the licensee has revised these analyses to credit two RHRSW pumps.

Essential components that are serviced by the SWS include the RHR heat exchangers, SFPCS heat exchangers, EDG coolers, ECCS room coolers, and the RHRSWS pump motor coolers. The licensee indicated that key heat exchanger performance parameters that were used in the EPU analyses are consistent with the GL 89-13 (heat exchanger performance testing) program results. Entergy found that the suppression pool temperature and containment pressure will be higher when in the most limiting RHR suppression pool cooling and containment spray cooling modes; additional time will be required to cool down the reactor when in the shutdown cooling mode due to the higher reactor decay heat; and the ECCS corner room temperatures could increase by several degrees following a LOCA. However, because these effects do not cause any design limits of SSCs to be exceeded and because licensing-basis considerations will continue to be satisfied, the licensee concluded that the current SWS performance capability and flow balance are sufficient for EPU conditions. Note that GL 96-06 considerations do not apply to the SWS and are therefore not discussed in this section (see SE Section 2.5.3.3).

Regarding the SW temperature, the NRC staff raised a concern in reference to UFSAR Section 10.6.5, where it describes a higher temperature limit of 88°F under certain conditions, when the maximum design basis limit is 85°F. In response to RAI SPLB-A-15 (Reference 29, Attachment 1), the licensee stated that the higher SW temperature (i.e., 88°F) discussed in UFSAR Section 10.6.5 addresses a unique summer operating condition during hybrid mode of circulating water system operation which will not be applicable for EPU conditions. The revised design-basis analyses for EPU conditions assume a SW temperature limit of 85°F. The licensee stated that the UFSAR will be updated in conjunction with issuance of the EPU license amendment in this regard.

Based on a review of the information that was submitted, the NRC staff is satisfied that the licensee has adequately evaluated and addressed the impact of the proposed EPU on the capability of the SWS (including the RHRSWS) to perform its safety functions. Because design limits of SSCs will not be exceeded and licensing-basis considerations will continue to be satisfied, the staff agrees that the capabilities of the SWS will not be impacted by the proposed EPU. Furthermore, existing GL 89-13 programmatic controls will continue to assure that heat exchanger performance is maintained consistent with licensing-basis considerations following implementation of the proposed EPU.

Conclusion

The NRC staff has reviewed the licensee's assessment of the impact that the proposed EPU will have on the SWS (including the RHRSW system) and finds that the SWS (including the RHRSWS) will continue to be capable of performing its equipment cooling and decay heat removal functions in accordance with licensing-basis considerations. Therefore, the proposed EPU is considered to be acceptable with respect to the SWS.

2.5.3.3 Reactor Auxiliary Cooling Water Systems

Regulatory Evaluation

Reactor auxiliary cooling water systems (RACWS) that are included within the scope of this evaluation include the reactor building closed cooling water system (RBCCWS) and the ACS. The RBCCWS is a non-safety-related system that is used to cool non-essential plant equipment and therefore, its cooling function is not important to safety. However, the RBCCWS is relied upon to maintain pressure boundary integrity so that containment bypass via the RBCCWS piping that penetrates containment does not occur, and some of the RBCCWS piping is used as part of the flow path for the ACS. The NRC staff's review of the RBCCWS confirms that the licensee's resolution of the GL 96-06 waterhammer and two-phase flow issues remains valid for EPU operation such that system integrity will not be challenged by waterhammer and two-phase flow conditions.

The cooling function of the ACS is relied upon in the event that the SWS becomes unavailable due to a failure of the Vernon Dam or due to a fire or flooding in the intake structure. The NRC staff's review of the ACS focused on the impact that the proposed EPU will have on the capability of the ACS to perform its safety function. The criteria most applicable to the staff's review are based primarily on draft GDC-67, "Fuel and Waste Storage Decay Heat (Category B)," insofar that the ACS is relied upon by fuel and waste storage decay heat removal systems for performing their functions; and other licensing-basis considerations that are applicable. The staff's review of the RACWS is performed in accordance with the guidance provided in Section 2.1 of RS-001, Matrix 5, and acceptability for EPU operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in Sections 10.8 and 10.9 of the VYNPS UFSAR, except where proposed changes are found to be acceptable based upon the specified review criteria.

Technical Evaluation

The licensee evaluated the capability of the ACS to perform its safety function following the proposed EPU, including all required heat loads assuming the worst-case EPU conditions. The major heat loads include reactor decay heat, spent fuel decay heat, the EDGs, and various pumps and coolers. The source of cooling water for the ACS is the west cooling tower deep basin. The analysis assumes an initial deep basin temperature of 105°F. In conjunction with the thermal heat load analysis, the licensee also performed inventory evaporative loss analysis to confirm that at least 7 days of cooling capability will be provided by the water that is available in the deep basin. The quantity of evaporative loss is calculated based on a cooling tower

mass and energy balance using psychrometric properties of the air-water mixture entering and leaving the tower. The summer design meteorological conditions of 90°F dry bulb and 50% relative humidity are conservatively used throughout the 7-day period. Based on the above analyses, the licensee determined that adequate cooling tower (deep basin) inventory is assured by the EPU-related plant modification that directs the ACS (RHRSW) pump motor cooler recirculation flow back to the cooling tower basin and that the ACS pump NPSH and capacity are adequate for EPU conditions. The ACS deep basin return temperature will remain below 130°F to protect the cooling tower fill, and the system cooling capacity is adequate for the required heat loads (Reference 11, Attachment 1, response to RAI SPLB-A-9).

Additionally, in response to a concern raised by the NRC staff, the licensee confirmed that the design basis meteorological conditions assumed for ACS operation under EPU conditions are the same as those used for the original design (Reference 29, Attachment 1, response to RAI SPLB-A-16). The original design used a 1% design wet bulb temperature of 75°F concurrent with an average maximum dry bulb temperature of 90°F and a relative humidity of 50%. This set of conditions conservatively envelopes the composite average of the 1967 American Society of Heating, Refrigerating and Air-Conditioning Engineers, Inc. (ASHRAE) data for various weather stations surrounding Vernon, Vermont. Based on the most recent ASHRAE data published in 1977 and again in 1997, the 1% wet bulb temperatures for these years varied from a high of 74°F to a low of 72°F, which demonstrates that the original design value of 75°F remains conservative. No credit was taken for wind effects in either the original or revised ACS analysis.

The licensee evaluated the impact of the proposed EPU on the resolution of the GL 96-06 waterhammer and two-phase flow issues, which relates to the RBCCWS. The licensee determined that the RBCCWS heat load will increase by less than 0.6% for the proposed EPU and that the previous analyses that were completed for resolving the GL 96-06 waterhammer and two-phase flow issues will continue to be bounding for EPU conditions. Therefore, the proposed EPU will not impact the licensee's resolution of GL 96-06 with respect to waterhammer and two-phase flow.

Based on a review of the information that was submitted, the NRC staff is satisfied that the licensee has adequately evaluated and addressed the impact of the proposed EPU on the resolution of GL 96-06 with respect to waterhammer and two-phase flow, and on the capability of the ACS to perform its safety function. Because the calculated heat loads for post-EPU operation are bounded by the current design parameters and the modification to recover RHRSW pump motor bearing cooling water will preserve inventory margins to pre-EPU levels, the staff agrees that the capabilities of the ACS will not be impacted by the proposed EPU. Also, because the existing GL 96-06 waterhammer and two-phase flow analyses bound EPU conditions, the EPU will not impact the resolution of GL 96-06 in this regard.

Conclusion

The NRC staff has reviewed the licensee's assessment of the proposed EPU on the RACWS and finds that the RACWS will continue to be capable of performing its equipment cooling and decay heat removal functions in accordance with licensing-basis considerations; and that the

resolution of GL 96-06 will not be impacted with respect to waterhammer and two-phase flow. Therefore, the proposed EPU is considered to be acceptable with respect to the RACWS.

2.5.3.4 Ultimate Heat Sink

Regulatory Evaluation

The ultimate heat sink (UHS) is the water source that is relied upon for cooling SSCs during all modes of reactor operation, during plant transients, and following postulated accidents. The Connecticut River and the on-site cooling towers perform the UHS function at VYNPS. The NRC staff's review of the UHS focused on the impact that the proposed EPU will have on the capability of the UHS to perform its safety functions. The staff also reviewed the UHS design-basis temperature limit determination to confirm that post-licensing data trends (e.g., air and water temperatures, humidity, wind speed, water volume) do not establish more severe conditions than previously assumed. The criteria most applicable to the staff's review are based primarily on draft GDC-41, "Engineered Safety Features Performance Capability (Category A)," insofar that the UHS is relied upon by engineered safety features for performing their safety functions; draft GDC-44, "Emergency Core Cooling System Capability (Category A)," insofar that the UHS is relied upon by emergency core cooling systems for performing their safety functions; draft GDC-52, "Containment Heat Removal Systems (Category A)," insofar that the UHS is relied upon by containment heat removal systems for performing their safety functions; draft GDC-67, "Fuel and Waste Storage Decay Heat (Category B)," insofar that the UHS is relied upon by fuel and waste storage decay heat removal systems for performing their safety functions; and other licensing-basis considerations that are applicable. The staff's review of the SWS is performed in accordance with the guidance provided in Section 2.1 of RS-001, Matrix 5, and acceptability for EPU operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in Sections 10.6 and 10.8 of the VYNPS UFSAR, except where proposed changes are found to be acceptable based upon the specified review criteria.

Technical Evaluation

The Connecticut River is relied upon as the UHS for accident mitigation purposes, as well as for removing heat from SSCs important to safety whenever the SWS is available. The Connecticut River is considered to be an infinite heat sink and will continue to function as previously assumed and therefore, it will not be affected by the proposed EPU.

When the SWS is not available due to a failure of the Vernon Dam or due to a fire or flooding in the intake structure, the north-end cell of the west cooling tower and its deep basin are credited for satisfying the cooling needs of the plant for at least 7 days.

Upon initiation of the ACS, normal cooling tower operation ceases and the alternate cooling cell (i.e., cell CT2-1 of the west cooling tower CT-2) is lined up for the ACS mode of operation. As described in SE Section 2.5.3.3, the licensee has determined that the cooling capability of the ACS for EPU operation is bounded by the original design basis assumptions, and that the cooling tower inventory will continue to be sufficient for satisfying the cooling needs of the plant

for at least 7 days following the proposed EPU. Also, in response to NRC staff's concerns, the licensee provided a description of the ACS analysis methodology that was used to assess the cooling tower heat removal capability for EPU conditions (Reference 11, Attachment 1, response to RAI SPLB-A-9; and Reference 29, Attachment 1, response to RAI SPLB-A-16). Regarding the water capacity of the cooling tower and the RHRSW pump NPSH requirements for post-EPU ACS mode operation, in its response to RAI SPLB-A-26 (Reference 31, Attachment 8), the licensee further confirmed that calculations were performed for post-EPU ACS mode design-basis heat loads and meteorological conditions, including the water losses due to evaporation, drift and external factors (e.g., pipe drainage during ACS setup, silt buildup and collapse of non-seismic portions of cooling structures). The licensee also reconfirmed that the cooling tower has a minimum capacity of 1,451,700 gallons. Based on the results of these calculations, the licensee determined that at the end of 7 days of post-EPU ACS operation, the cooling tower basin will still have at least 116,000 gallons which is a margin of about 8%. The licensee also confirmed that the cooling tower basin will have sufficient water inventory for satisfying NPSH requirements for the RHRSW pumps over the entire 7-day period.

In order to preserve the cooling tower basin inventory and assure at least 7 days worth of cooling capability for EPU conditions, the licensee added three-way ball valves in the RHRSW pump motor bearing oil cooling water return piping to allow the cooling water to be redirected from the reactor building storm drains (normal alignment) to the suction piping of the RHRSW pumps (ACS cooling alignment). When switching to ACS cooling, an additional step will be added to the operating procedure to reposition the three-way ball valves to the ACS cooling alignment. Operator action is assumed to be completed within 2 hours.

Based on a review of the information that was provided, the NRC staff is satisfied that the licensee has adequately evaluated and addressed the impact of the proposed EPU on the capability of the UHS to perform its safety functions. The Connecticut River will not be affected by the proposed EPU and will continue to function as currently assumed; and the licensee has adequately demonstrated that the north cell of the west cooling tower and its deep basin will continue to be capable of cooling SSCs important to safety for at least 7 days if the SWS should become unavailable during EPU operation.

In December 2005, the Atomic Safety and Licensing Board admitted a late-filed contention, challenging the adequacy of the licensee's April 2005 ACS cooling tower seismic evaluation. The NRC staff has reviewed the contention and supporting documentation, the challenged seismic evaluation, and the licensee's substantive responses to the issues raised in the contention. Based on its review, the staff has concluded that the licensee's seismic evaluation adequately accounts for the cooling tower modifications which were installed to support the EPU. Further, there is reasonable assurance that the cooling tower modifications, and operations under EPU conditions, will not adversely affect the ability of the ACS to continue to perform its intended safety function following a design basis seismic event.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the capability of the UHS to perform its safety functions and finds that the UHS will remain capable of dissipating reactor decay heat and cooling SSCs important to safety in accordance with licensing-basis considerations. Therefore, the proposed EPU is considered to be acceptable with respect to the UHS.

2.5.4 Balance-of-Plant Systems

2.5.4.1 Main Steam

The main steam supply system (MSSS) transports steam from the reactor to the power conversion system and to various auxiliary steam loads. While the MSSS does not perform a safety function, marginal design aspects of the MSSS can result in adverse safety consequences. The NRC staff's review of the MSSS for a proposed EPU focuses primarily on system design limitations to assure that reactor safety will be preserved. The effects of increased steam flow and changes in steam quality on corrosion are evaluated in SE Section 2.1; the capability of the MSSS to withstand the loads that result from the rapid closure of the main steam isolation valves (MSIVs) and turbine stop valves, the capability of the MSIVs to isolate steam flow within the time period required, design considerations associated with the rapid closure of the MSIVs, and evaluation of piping stresses due to plant transients and accident conditions are evaluated in SE Section 2.2; transient testing is addressed in SE Section 2.12; and protection of SSCs important to safety from the effects of high energy line breaks and missiles is evaluated in SE Section 2.5.1. Because the impact of the proposed EPU on the MSSS is encompassed by these other areas of review, a separate evaluation for the MSSS in this section is not required.

2.5.4.2 Main Condenser

Regulatory Evaluation

The main condenser system (MCS) is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the turbine steam bypass system (TSBS), and is typically credited for providing sufficient condensate retention time to allow short-lived radioactive isotopes to decay. For BWRs without an MSIV leakage control system, the MCS may also act as a holdup volume for the plate-out of fission products leaking through the MSIVs following core damage, but the NRC staff does not expect this function to be affected by proposed EPUs. The NRC staff's review focuses primarily on the impact of the proposed EPU on the capability to maintain sufficient condensate retention time to allow short-lived radioactive isotopes to decay, thereby maintaining radiation doses as low as reasonably achievable. The impact of the proposed EPU on the MCS flooding analysis is included within the scope of SE Section 2.5.1.1.1, and is not evaluated in this section. The criteria most applicable to the staff's review of the MCS are based on draft GDC-70, "Control of Releases of Radioactivity to the Environment (Category B)," insofar as it specifies that the plant design include means to control the release of radioactive effluents; and other licensing-basis considerations that are applicable.

The staff's review of the MCS is performed in accordance with the guidance provided in Section 2.1 of RS-001, Matrix 5, and acceptability for EPU operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in Section 11.3 of the VYNPS UFSAR, except where proposed changes are found to be acceptable based upon the specified review criteria.

Technical Evaluation

The MCS includes two 50% capacity condensers with hotwells that are designed to provide a minimum condensate retention time of 2 minutes to allow for the decay of short-lived radioactive isotopes. The licensee has determined that the 2-minute retention time will continue to be maintained for EPU operation. Since VYNPS does not have a MSIV leakage control system, the licensee's evaluation also pertains to heater drain and extraction steam holdup times in the condenser hotwells.

Based on a review of the information that was submitted, the NRC staff is satisfied that the licensee has adequately evaluated and addressed the impact of the proposed EPU on the capability of the MCS to maintain sufficient condensate retention time to allow short-lived radioactive isotopes to decay consistent with the existing plant licensing basis.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the MCS and finds that condensate retention times for decay of short-lived radioactive isotopes will be maintained in accordance with the plant licensing basis. Therefore, the proposed EPU is considered to be acceptable with respect to the MCS.

2.5.4.3 Turbine Steam Bypass System

The turbine steam bypass system (TSBS) is a non-safety-related system designed to discharge a percentage of rated main steam flow directly to the main condenser, bypassing the turbine and in some cases enabling the plant to take step-load reductions without causing a reactor trip. The system is also used during startup and shutdown to control reactor pressure. The NRC staff's review of the TSBS focused on the impact that the proposed EPU will have on the existing capability to protect SSCs important to safety from TSBS piping and component failures (Reference 9, Attachment 1, response to RAI SPLB-A-6). Because TSBS piping and component failures are included within the scope of SE Sections 2.5.1.2 and 2.5.1.3, a separate evaluation of the TSBS in this section is not required.

2.5.4.4 Condensate and Feedwater System

Regulatory Evaluation

The condensate and feedwater system (CFS) provides feedwater at a particular temperature, pressure, and flow rate to the reactor. While the CFS does not perform a safety function, marginal system design and operational capability could result in loss of feedwater transients

and increased challenges to safety systems. The NRC staff's review of the CFS focuses primarily on system design limitations and reductions in operational flexibility that will result due to EPU operation. Because the effects of increased CFS flow on corrosion rates is evaluated in SE Section 2.1, the capability of the CFS piping and supports to withstand postulated loads is evaluated in SE Section 2.2, and protection of SSCs important to safety from the effects of high energy line breaks and missiles is evaluated in SE Section 2.5.1, these areas are not included within the scope of this section. The acceptance criteria that are most applicable to the staff's review of the CFS are based on existing plant licensing-basis considerations, especially with respect to maintaining CFS reliability and minimizing challenges to reactor safety systems during EPU operation. The staff's review of the CFS is performed in accordance with the guidance provided in Section 2.1 of RS-001, Matrix 5, and acceptability for EPU operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in Section 11.8 of the VYNPS UFSAR, except where proposed changes are found to be acceptable based upon the specified review criteria.

Technical Evaluation

The CFS does not perform a safety-related function per se; however, its performance can have a major effect on plant availability and capability to operate reliably at the EPU conditions, and failures in the CFS can result in loss of feedwater events and present challenges to reactor safety systems. The CFS is designed to provide sufficient feedwater at an elevated pressure and temperature to maintain the reactor vessel level within a predetermined range during all modes of power operation. The licensee has evaluated the capability of the CFS to perform its intended functions during EPU operation and has determined that in order for the CFS to support the proposed EPU, the following changes are necessary:

- All three reactor feedwater pumps (RFPs) must be operating in order to satisfy EPU feedwater demands whereas for the current rated power level, only two RFPs are required to be operating. Consequently, the existing spare RFP must be placed in service for EPU operation and it will no longer be available for standby operation in the event that one of the operating RFPs experiences a failure. Three RFPs will provide about 16% additional flow margin for EPU operation.

- All three condensate pumps must be operating in order to satisfy EPU feedwater demands. While the licensee's practice is to operate all three condensate pumps at the current licensed power level, two pumps are marginally capable of performing this function. At EPU conditions, the condensate pumps will exceed their name-plate rating but will remain within their design capability as confirmed by the pump motor vendors.

- A reactor recirculation system run-back modification will be installed to avoid the plant trip that would otherwise occur due to the loss of a condensate pump or an RFP. The loss of an RFP or condensate pump at the current licensed power level will not cause a reactor trip due to the extra capacity that is available.

- In order to provide additional operating margin so as to prevent a trip of all RFPs due to the loss of a condensate pump, the suction pressure trip set point was lowered (but not to the

point where NPSH requirements for the RFPs will not be satisfied); and a sequential trip of the RFPs was installed so that all RFPs will not trip immediately following the loss of a condensate pump, thereby allowing time for the transient to pass without causing a loss of all reactor feedwater. The loss of a condensate pump at the current licensed power level will not cause a trip of the RFPs due to the flow margin that is available.

- The high pressure feedwater heaters were replaced to support the higher extraction steam pressures that are necessary for EPU operation.

- A filtered bypass around the condensate demineralizer system was installed to allow for the removal of one condensate demineralizer elements during the periodic back-washing and pre-coating process as well as for general maintenance or element replacement.

In light of the reduced flow margin that will be available following the proposed EPU, the NRC staff raised concerns regarding the reliability of the CFS and the increased challenges to reactor safety systems that could result during EPU operation. In response to the staff's concerns, in a letter dated April 22, 2005 (Reference 29, Attachment 1, response to RAI SPLB-A-17), the licensee provided the following additional information:

- The RFP reliability is maintained through monitoring, preventative and on-line maintenance. If needed, an RFP can be removed from service during planned power reductions. The automatic runback of the recirculation pumps is designed to prevent an inadvertent reactor trip on loss of an RFP, thus preserving overall reliability of the plant. During normal EPU operation, the three RFPs will operate at lower capacity (with less stress on pumps and motors) than two RFPs operating at the current licensed power level.

- During operation at EPU conditions, each RFP will deliver 5,831 gpm, a decrease of approximately 16% when compared to the flow rate of 6,965 gpm per RFP that is currently required. This reduction in individual RFP flow increases the available margin from normal operating flows to runout for the individual pump. On the other hand, the three condensate pumps will be required to provide the increased flow associated with EPU operation thereby reducing the available flow margin for the condensate pumps. However, the condensate pump flow margin for EPU conditions will be approximately 7% to pump runout. The licensee concluded that this will continue to exceed the industry recommended criteria of 5% margin.

- The lower individual RFP flow rate that will be required for three pump operation will result in reduced power consumption by the RFP motors; whereas, the condensate pump motor requirements will increase from 1,410 hp to approximately 1,500 hp for EPU operation. The licensee indicated that the motor manufacturer has determined that sufficient design margin for continuous operation of the condensate pump motors at EPU conditions would still be available.

- The minimum calculated RFP suction pressure following the trip of a condensate pump at EPU conditions will be approximately 124 psig. During the spring 2004 refueling outage (RFO 24), the RFP low suction pressure trip setpoint was reduced from 150 psig to 98 psig

decreasing with staggered time delays of 30, 40, and 45 seconds for tripping each additional RFP. This prevents a total loss of main feedwater to the reactor under transients such as a trip of a condensate pump or an RFP at EPU conditions, which is important to safety. Also, there is an additional (i.e., low-low suction pressure) trip setpoint of 90 psig +/- 2 psi with a one second time delay for non-sequential trips of all three RFPs, which provides protection to the RFPs for loss of NPSH events (Reference 33, Attachment 8, response to RAI SPLB-A-30). The licensee concluded that there is sufficient margin between the minimum transient RFP suction pressure and the current RFP suction pressure trip setpoint to support RFP operation for EPU conditions. The licensee reiterated in its followup response to RAI SPLB-A-27 (Reference 31, Attachment 8) that based on the analysis that was performed, the RFP suction pressure following a condensate pump trip is expected to drop to about 124 psig, which is sufficient to avoid a consequential RFP trip.

- By letter dated November 2, 2005 (Reference 42), the licensee reported a calculation error in its analysis regarding the RFP suction pressure following a CP trip due to a non-conservative input assumption. This non-conservative input assumption resulted in a reduction in the calculated RFP suction pressure following a CP trip below the previously calculated value of 124 psig. The licensee determined that this reduction in the RFP suction pressure could potentially result in loss of feedwater. Therefore, the licensee committed to implement a plant modification that will automatically trip the "B" RFP upon a trip of any CP when operating at EPU conditions. This modification will result in a calculated RFP low suction pressure of approximately 162 psig upon loss of a CP, thus increasing the margin to loss of all RFPs due to low suction pressure. The net result of this modification is an increase in the margin to the low suction pressure RFP trip setting, thus preserving continued operation of the CFS and avoiding challenges to reactor safety systems. The licensee provided a regulatory commitment to implement this modification during the fall 2005 refueling outage. The staff reviewed the details described in the above letter and its attachments with respect to the proposed modification and its regulatory commitment. The staff found the modification and commitment acceptable.

Also, in response to questions that were raised by the NRC staff, the licensee described post-modification testing that was performed for the reactor recirculation pump runback and the RFP low suction pressure trip modifications (Reference 29, Attachment 1, response to RAI SPLB-A-18). For the recirculation pump runback modification, testing included complete logic verification from the initiation signal to movement of the actuating device, instrument calibration, simulation of recirculation motor-generator set operation, and confirmation that the expected response to simulated plant inputs occurred. For the RFP low suction pressure trip modification, a similar approach was used. Instrumentation was calibrated, the breakers for the RFPs were placed in the test position, and the required breaker trips were observed in response to simulated plant inputs.

In addition, the NRC staff, citing operating experience at the Dresden nuclear station, requested that the licensee explain how feedwater level control system operation for EPU conditions will assure that the margins for reactor vessel level overshoot will be maintained. In Reference 29 (response to RAI SPLB-A-22), the licensee stated that an evaluation was conducted to determine if the effects from a Dresden Unit 3 feedwater level event (discussed in Licensee

Event Report 2004-002-00) would be applicable to VYNPS. The licensee concluded that "[b]ased on (a) the margins provided by the differences in design of the HPCI steam supply line configuration and the feedwater control system; (b) the procedural response to a reactor trip; and (c) the startup testing previously performed and to be performed during the EPU Power Ascension Test Program, the increase in feedwater flow by approximately 20% at VYNPS should not cause a feedwater overshoot situation as experienced at Dresden Unit 3." The staff reviewed the details of licensee's response and found the response acceptable with respect to reactor vessel overshoot considerations.

The increased demands that will be placed on the CFS as a consequence of EPU operation in conjunction with the modifications that are being made will result in substantial changes in the CFS transient response from what has been experienced previously while operating at the current licensed power level. While the NRC staff agrees that the CFS modifications are appropriate and necessary in order to minimize any increase in the frequency of loss of feedwater events (which is a design-basis initiating event) and to minimize challenges to reactor safety systems during EPU operation, the staff requested that the licensee propose transient testing in order to confirm that the CFS response to loss of condensate pump and RFP events is consistent with the analytical results. Because the licensee has no CFS transient data that are representative of EPU full power operating conditions, uncertainties in CFS modeling and analysis could be substantial. If the margins in the CFS modeling and analysis are not sufficient to bound these uncertainties, the loss of a condensate pump may cause all RFPs to trip, contrary to what is expected based solely on analytical results. The NRC staff concluded that transient testing is necessary in order to demonstrate that the loss of a condensate pump will not result in a total loss of reactor feedwater or otherwise present a challenge to reactor safety systems.

The licensee provided supplemental information regarding CFS testing in letters dated August 5 and September 10, 2005 (Reference 31, Attachment 8, response to RAI SPLB-A-28 and Reference 33, Attachment 8, response to RAI SPLB-A-30, respectively). The licensee maintains that the CFS analysis alone is sufficient and that transient testing of the CFS is not warranted. However, the NRC staff notes that design features similar to those being installed at VYNPS to assure reliable CFS operation following the proposed EPU and to otherwise minimize challenges to reactor safety systems were tested by the original plant startup/transient test programs to confirm that the analytical results were in fact correct and that the expected plant response was achieved. The licensee's justification for not performing transient testing of the CFS as requested by the NRC staff is inadequate in that confirmation of the analytical results in accordance with review criteria provided in SRP 14.2.1 has not been demonstrated. Consequently, the NRC staff proposed a license condition to require the completion of CFS transient testing in order to confirm the adequacy of analytical results as a prerequisite for EPU operation.

Based on a review of the information that was submitted, the NRC staff is satisfied that the licensee has adequately evaluated and addressed the impact of the proposed EPU on the capability and reliability of the CFS to provide reactor feedwater for EPU operation. The NRC staff agrees that the modifications that are being made to the design and operation of the CFS are appropriate and necessary in order to maintain the reliability of the CFS and minimize

challenges to reactor safety systems during EPU operation. However, in order to confirm that analytical results are accurate and to assure acceptable transient behavior of the CFS during EPU operation, the following license condition will be added to Facility Operating License (FOL) DPR-28 (i.e., VYNPS FOL) as discussed in SE Section 3.17:

L. Transient Testing

1. During the extended power uprate (EPU) power ascension test program and prior to exceeding 168 hours of plant operation at the nominal full EPU reactor power level, with feedwater and condensate flow rates stabilized at approximately the EPU full power level, Entergy Nuclear Operations, Inc. shall confirm through performance of transient testing that the loss of one condensate pump will not result in a complete loss of reactor feedwater.

2. Within 30 days at nominal full-power operation following successful performance of the test in (1) above, through performance of additional transient testing and/or analysis of the results of the testing conducted in (1) above, confirm that the loss of one reactor feedwater pump will not result in a reactor trip.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the CFS and finds that the CFS will remain capable of providing an adequate supply of reactor feedwater during EPU operation. A license condition will be added to FOL DPR-28 to require transient testing of the CFS as a prerequisite to EPU operation in order to assure that CFS transient performance is consistent with analytical results. Given these considerations, the NRC staff is satisfied that adequate assurance will be established, prior to commencing full power operation at the EPU reactor power level, that the CFS will continue to be reliable and will not pose increased challenges to reactor safety systems during EPU operation. Therefore, the staff has concluded that, subject to this additional testing, with the modifications being made, the CFS will continue to satisfy licensing-basis considerations and the proposed EPU is acceptable with respect to the CFS.

2.5.5 Waste Management Systems

2.5.5.1 Gaseous Waste Management Systems

Regulatory Evaluation

Gaseous waste management systems (GWMSs) involve the gaseous radwaste system, which deals with the control of radioactive gases collected in the off-gas system or the waste gas storage and decay tanks. In addition, it involves the management of the condenser air removal system; the gland seal exhaust and the mechanical vacuum pump exhaust; and building ventilation system exhausts. The NRC staff's review focused on the effects that the proposed EPU may have on (1) the design criteria of the gaseous waste management systems, (2) methods of treatment, (3) expected releases, (4) principal parameters used in calculating the

releases of radioactive materials in gaseous effluents, and (5) design features for precluding the possibility of an explosion if the potential for explosive mixtures exists. Note that the standby gas treatment system is evaluated in SE Section 2.7.2 and is not included within the scope of this evaluation. The criteria that are most applicable to the staff's review of the GWMS for proposed EPUs are based on (1) 10 CFR 20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified values; (2) 10 CFR Part 50, Appendix I, Sections II.B, II.C, and II.D, which set numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" (ALARA) criterion; (3) 10 CFR Part 100, insofar that offsite dose limits must not be exceeded; (4) draft GDC-3, "Fire Protection (Category A)," insofar as it specifies that the reactor facility shall be designed to minimize the probability of fire and explosions; (5) draft GDC-70, "Control of Releases of Radioactivity to the Environment (Category B)," insofar as it specifies that the plant design include means to control the release of radioactive effluents; and (6) other licensing-basis considerations that apply. The staff's review of the GWMS is performed in accordance with the guidance provided in Section 2.1 of RS-001, Matrix 5, and acceptability for EPU operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in Section 9.4 of the VYNPS UFSAR, except where proposed changes are found to be acceptable based upon the specified review criteria.

Technical Evaluation

The licensee evaluated the impact of the proposed EPU on the capability of the GWMS to perform its functions and determined that sufficient capacity exists without modification to process the increase in gaseous waste that will result from EPU operation. The radiological release rate is administratively controlled to remain within existing site release rate limits, and is a function of fuel cladding performance, main condenser air inleakage, charcoal adsorber inlet dew point, and charcoal adsorber temperature. The licensee found that none of these parameters are significantly affected by the proposed EPU, and concluded that the EPU will only affect the flow rate of radiolytic hydrogen and oxygen to the offgas system. Consequently, only the catalytic recombiner temperature and offgas condenser heat load are affected. Because the VYNPS offgas system component design for heat load provides a 70% margin relative to the current radiolytic gas flow rate, the licensee concluded that the gaseous radwaste system will continue to satisfy the plant licensing basis.

Based on a review of the information that was submitted, the NRC staff is satisfied that the licensee has adequately evaluated and addressed the impact of the proposed EPU on the capability of the GWMS to perform its functions. Because the increase in offsite dose will be relatively small and remain well below that reported in the VYNPS Final Environmental Statement (FES), and it complies with 10 CFR 20.1302, 10 CFR Part 50, Appendix I, Sections II.B, II.C, and II.D and 10 CFR Part 100 requirements, the staff agrees that the capabilities of the GWMS will continue to satisfy the plant licensing basis during EPU operation.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the capability of the GWMS to perform its functions and finds that the GWMS will continue to control the release of radioactive materials and preclude the possibility of waste gas explosions in accordance with licensing-basis considerations. Therefore, the proposed EPU is considered to be acceptable with respect to the GWMS.

2.5.5.2 Liquid Waste Management Systems

Regulatory Evaluation

The liquid waste management system (LWMS) consists of process equipment and instrumentation necessary to collect, process, monitor, store, recycle, and/or dispose of liquid radioactive waste. Major components include floor and equipment drains, transfer pumps, and various waste system tanks. The NRC staff's review of the LWMS focuses on the effects that the proposed EPU may have on previous analyses and considerations related to the processing and management of liquid radioactive wastes, such as expected releases and principal considerations used in estimating the increase in volume of the liquid radioactive waste that will be released. The criteria that are most applicable to the staff's review of the LWMS are based on (1) 10 CFR 20.1302, insofar as it provides for demonstrating that annual average concentrations of radioactive materials released at the boundary of the unrestricted area do not exceed specified limits; (2) 10 CFR Part 50, Appendix I, Sections II.A and II.D, which set numerical guides for dose design objectives and limiting conditions for operation to meet the "as low as reasonably achievable" criteria; (3) draft GDC-70, "Control of Releases of Radioactivity to the Environment (Category B)," insofar as it specifies that the plant design include means to control the release of radioactive effluents; and (4) other licensing-basis considerations that apply. The staff's review of the LWMS is performed in accordance with the guidance provided in Section 2.1 of RS-001, Matrix 5, and acceptability for EPU operation is judged based upon conformance with existing licensing-basis considerations as discussed primarily in Section 9.2 of the VYNPS UFSAR, except where proposed changes are found to be acceptable based upon the specified review criteria.

Technical Evaluation

The LWMS is designed to collect, process, recycle and dispose of radioactive liquid waste in accordance with the requirements outlined in 10 CFR Part 20 and in 10 CFR Part 50, Appendix I, and in accordance with the criteria specified by draft GDC-70. The information that was provided indicated that the proposed EPU will not change the operation or design of the equipment used in the LWMS, the radiological and environmental monitoring of the waste streams will not be affected, and no new or different radiological release paths will be introduced as a result of the proposed EPU. However, the licensee determined that the proposed EPU will cause the amount of liquid radioactive waste to increase due to more frequent backwashing of condensate demineralizers (the largest source of additional liquid radioactive waste), and more frequent backwashing of the reactor water cleanup filter-demineralizers. In response to a question that was asked by the NRC staff (Reference 29,

Attachment 1, response to RAI SPLB-A-24), the licensee indicated that the proposed EPU will cause the volume of liquid processed waste to increase by about 1.15%. Since the design and operation of the LWMS will not change, and the volume of fluid flowing into the liquid radwaste system will not increase significantly as a result of EPU, the licensee concluded that the capacity of the LWMS will continue to be adequate.

Based on a review of the information that was submitted, the NRC staff is satisfied that the licensee has adequately evaluated and addressed the impact of the proposed EPU on the capability of the LWMS to perform its functions. Because the increase in additional radioactive waste being generated due to EPU operation is expected to be minimal and well within the capacity of the liquid radioactive waste processing system, any increase in offsite dose projections as a consequence is expected to be inconsequential and remain well below established plant release limits.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the capability of the LWMS to perform its functions and finds that the LWMS will continue to control the release of liquid radioactive materials in accordance with licensing-basis considerations. Therefore, the proposed EPU is considered to be acceptable with respect to the LWMS.

2.5.5.3 Solid Waste Management Systems

Solid radioactive waste consists of wet and dry waste. Wet waste consists mostly of low specific activity spent secondary and primary resins and filters, and oil and sludge from various contaminated systems. The NRC staff's review related primarily to wet waste dewatering and liquid collection processes, and focused on the impact that the proposed EPU will have on the release of radioactive materials to the environment via gaseous and liquid effluents. Because SE Sections 2.5.5.1 and 2.5.5.2 encompass these considerations, a separate evaluation of solid waste management systems in this section is not required.

2.5.6 Additional Review Areas

2.5.6.1 Emergency Diesel Engine Fuel Oil Storage and Transfer System

Nuclear power plants are required to have redundant onsite emergency power supplies (e.g., emergency diesel generators (EDGs)), of sufficient capacity to perform their safety functions assuming a single failure. The NRC staff's review focused on the effects that the proposed EPU may have on the fuel oil storage requirements for the EDGs. As clarified in response to a question that was raised by the NRC staff (Reference 29, Attachment 1, response to RAI SPLB-A-19), the licensee indicated that the electrical rating and the fuel oil consumption rate of the EDGs are not affected by the proposed EPU. Consequently, the existing fuel oil storage requirements are also not affected. Therefore, an evaluation of the EDG fuel oil storage requirements for the proposed EPU is not required.

2.5.6.2 Light Load Handling System (Related to Refueling)

The light load handling system (LLHS) includes components and equipment used for handling new fuel at the receiving station and for loading spent fuel into shipping casks. Because the licensee is not introducing any new fuel designs in conjunction with the proposed EPU, the fuel handling analysis for the current licensed power level is not affected by the EPU. Therefore, an evaluation of the LLHS for the proposed EPU is not required.

2.6 Containment Review Considerations

2.6.1 Primary Containment Functional Design

Regulatory Evaluation

The containment encloses the reactor system and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident inside containment. The NRC staff's review for the primary containment functional design covered (1) the temperature and pressure conditions in the drywell and wetwell due to a spectrum of postulated LOCAs, (2) suppression pool dynamic effects during a LOCA or following the actuation of one or more reactor coolant system (RCS) safety/relief valves, (3) the capability of the containment to withstand the effects of steam bypassing the suppression pool, (4) the suppression pool temperature limit during RCS safety/relief valve operation, and (5) the analytical models used for containment analysis. The NRC's acceptance criteria for the primary containment functional design are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for engineered safety features (ESFs) against the dynamic effects that might result from plant equipment failures, as well as the effects of a LOCA; (2) draft GDC-10, insofar as it requires that reactor containment be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other ESFs as may be necessary, to retain for as long as the situation requires the functional capability; (3) draft GDC-49, insofar as it requires that the containment and its associated heat removal systems be designed so that the containment structure can accommodate, without exceeding the design leakage rate, the pressures and temperatures resulting from the largest credible energy release following a LOCA, including considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems; (4) draft GDC-12, insofar as it requires that instrumentation and controls be provided as required to monitor and maintain variables within prescribed operating ranges; and (5) draft GDC-17, insofar as it requires that means be provided to monitor the reactor containment atmosphere for radioactivity that may be released from normal operations and from postulated accidents. Specific review criteria are contained in SRP Section 6.2.1.1.C.[1]

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1 Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Section 6.2.1.1.C, Pressure Suppression Type BWR Containments, Revision 6, US NRC, August 1984.

Technical Evaluation

The primary containment, as described in Section 5 of the VYNPS UFSAR, is a Mark I design consisting of (1) a drywell which encloses the reactor vessel, the RCS and other branch connections to the RCS, (2) a toroidal shaped pressure suppression chamber (the suppression chamber, wetwell, or torus) partially filled with a large volume of water (the suppression pool), (3) a vent system connecting the drywell atmosphere to the water space of the torus, (4) containment isolation valves, (5) containment cooling systems, and (6) other equipment.

The proposal to operate at EPU conditions requires that safety analyses for those DBAs whose results depend on power level be recalculated at the higher power level. The containment design basis is primarily established based on the LOCA and the actuation of the reactor vessel safety relief valves and their discharge into the suppression pool.

Short-Term LOCA Analysis

The short-term LOCA analysis is performed to show that the peak drywell pressure and temperature remain below the drywell design pressure of 56 pounds per square inch gauge (psig) and the drywell design temperature of 281°F. The licensee made predictions of the short-term LOCA containment response using analytical methods approved for EPUs. The power uprate methods approved by the NRC permit the use of either the M3CPT computer code or the LAMB computer code to calculate the mass and energy release from the postulated pipe break into the drywell.[2] The licensee has used the LAMB code[3] with the Moody slip critical flow model.[4] The Moody slip critical flow model is conservative compared to more realistic prediction methods such as the homogeneous equilibrium model (HEM).[5] The HEM was used for break flow calculations as part of the Mark I Long-term Program to address containment hydrodynamic loads.[6]

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2 Licensing Topical Report: Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate, GE Nuclear Energy, NEDC-32424P, February 1995.

3 GE Nuclear Energy, "General Electric Model for LOCA Analysis in Accordance with 10 CFR 50 Appendix K," NEDE-20566-P-A, September 1986.

4 Moody, F. J., "Maximum Flow Rate of a Single Component, Two Phase Mixture," Transaction of the ASME, Volume 87, Series C, 1966.

5 R. T. Lahey, Jr. and F. J. Moody, The Thermal-Hydraulics of a Boiling Water Reactor, The American Nuclear Society, 1984.

6 Safety Evaluation Report Mark I Containment Long-Term Program Resolution of Generic Technical Activity A-7 NUREG 0661, US NRC, July 1980.

The licensee provided a list of the assumptions used for the short-term peak pressure and temperature calculations.[7] These assumptions are conservative for predicting these quantities. For example, the maximum values of suppression pool water volume and downcomer submergence are assumed and initial conditions are assumed which maximize the amount of noncondensible gas in the drywell. These assumptions maximize the peak short-term drywell pressure.

The results of these analyses and the acceptance criteria are provided in Table 4-1 of the PUSAR[8] which is reproduced in the following table.

RESULTS OF VYNPS SHORT-TERM LOCA ANALYSIS

Parameter	CLTP from UFSAR	CLTP with CPPU Method	CPPU	Design Limit
Peak Drywell Pressure (psig)	38.2	41.6	41.8	56
Peak Drywell Air Space Temperature (°F)*	284	287.7	287.8	281**

* This is the peak drywell air space temperature due to a design basis double-ended guillotine break of a recirculation suction line.

** The 281°F value is the structure design temperature.

The table compares the peak pressure and temperature predicted using the same calculation methods at the current power level and the CPPU power level (columns 3 and 4) so that the effect of power level is highlighted. There is little difference between the peak pressure and the peak temperature at the CLTP level and the CPPU power level. Since the power uprate maintains the reactor vessel pressure constant and the peak short-term pressure and temperature values are reached in a few seconds, the values depend mostly on the break critical flow model (the Moody model) and the flow resistance modeling of the vents and downcomers between the drywell and the torus. Since these are the same in this case for the CLTP level and the CPPU power level, little difference is expected.

Section 4.1.1.3 of the PUSAR states that the difference in the values of drywell pressure and drywell temperature at the current power level calculated with the current analysis methods and with the proposed CPPU analysis methods is due to the use of the Moody slip flow correlation

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7 Entergy letter (BVY 04-081) to NRC dated August 12, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 11, Extended Power Uprate - Response to Request for Additional Information" (Table SPSB-C-32-1).

8 Entergy letter (BVY 03-80) to NRC dated September 10, 2003, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Extended Power Uprate" (Attachment 4).

in the CPPU methods and the HEM in the UFSAR methods. The Moody correlation is more conservative than the HEM.

Based on the above discussion, the trends of the licensee's predictions of the containment parameters in the above table are as expected.

The results of these calculations show that the peak drywell pressure remains below the design limit. The drywell air space temperature exceeds 281°F structural design value. The licensee states that the drywell air space temperature exceeds the structural design temperature for less than 10 seconds which is insufficient time for the drywell structure to reach the 281°F design limit. The NRC staff concurs.

The NRC staff has performed an independent calculation of the mass and energy releases for the drywell peak pressure case. These calculations were done using the RELAP5/MOD 3 code and a model of the VYNPS core and vessel provided by the licensee in Reference 49. The staff modified this model for EPU conditions and GE-14 fuel. This is not the mass and energy release model used by the licensee for the licensing basis calculations shown in the above table and therefore the calculation is independent of the licensee's methods. As stated previously, the licensee used the General Electric LAMB code for the short-term mass and energy releases and the M3CPT code for the containment response.

The NRC staff calculated the mass and energy release for a double-ended guillotine break in the recirculation line. The calculation was performed to verify the mass and energy released to the containment following a large-break LOCA. A description of the RELAP5/MOD3 model is provided in a letter from the licensee dated September 15, 2004.[9]

Figure 1 presents the RELAP5/MOD3 model of the VYNPS NSSS. As stated above, the RELAP5/MOD3 deck obtained from the licensee was modified to reflect the higher power level and the GE-14 fuel design.

Figure 2 presents the mass flow rate from the break in the suction line. This flow rate consists of the break flow exiting both sides of the double-ended guillotine break in the recirculation line. To model the double-ended break, the break junctions are located at the exit to volume 304 and the inlet to volume 306 in Figure 1. The Henry-Fauske critical flow model in the RELAP5/MOD3 code was activated to predict the break flow. Inspection of Figures 2 and 3 shows that the RELAP5 code confirms the conservative nature of the VYNPS calculation at EPU conditions.

After the initial NRC staff analysis was completed, it was found that heat from wall heat structures was not included in the staff results. A reanalysis by the NRC staff found that this made a difference of less than 1% and the previous results remain valid.

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9 Entergy letter (BVY 04-098) to NRC dated September 15, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 14, Extended Power Uprate - Response to Request for Additional Information."

P_a is the pressure at which containment leakage rate testing is performed. It is defined in 10 CFR Part 50, Appendix J as the calculated peak containment internal pressure related to the design basis LOCA. The VYNPS TSs specify a value of P_a equal to 44 psig. Since the calculated peak containment internal pressure for this EPU is less than 44 psig, no change to the TSs is needed.

Based on the use of acceptable calculation methods and conservative assumptions, results less than the design containment pressure and temperature, and the NRC staff's independent calculations, the NRC staff concludes that the VYNPS short-term containment response at EPU conditions is acceptable.

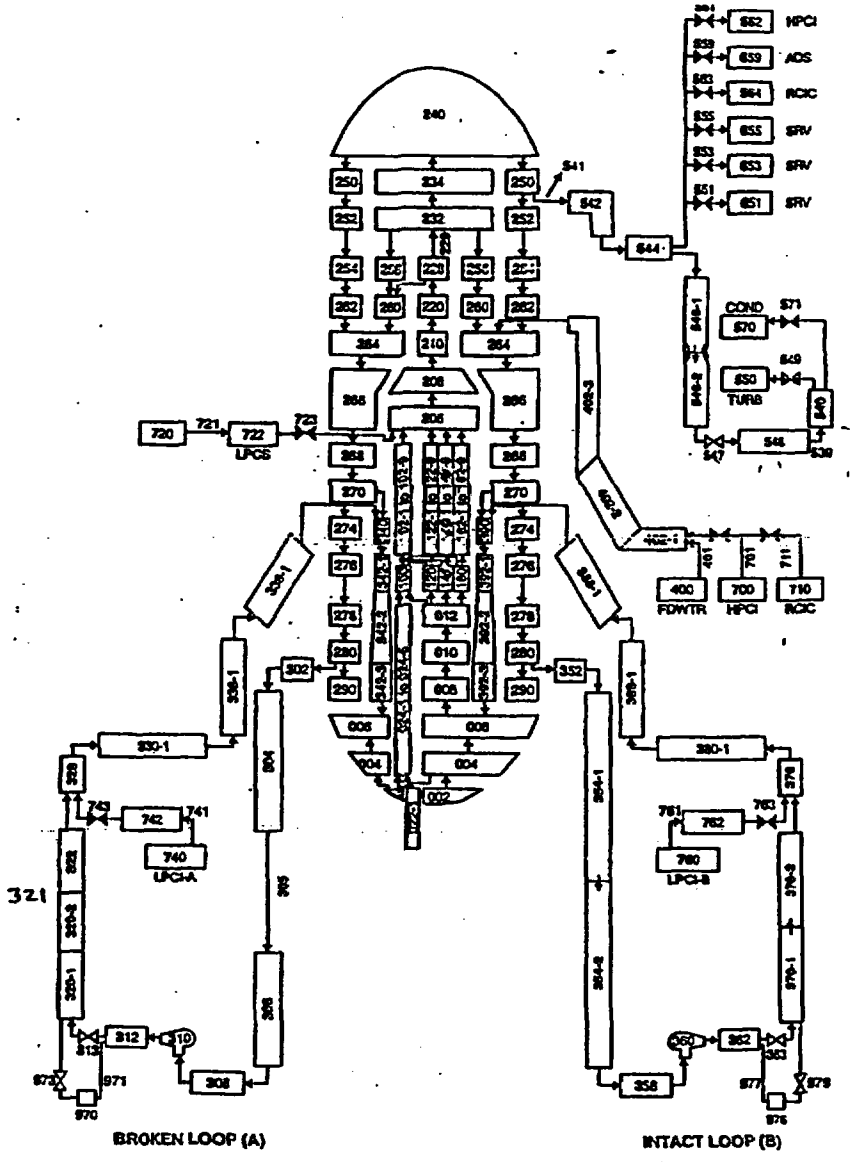


Fig. 1: Vermont Yankee RELAP5 Nodalization

Fig. 2: Break Mass Flow Rate vs Time

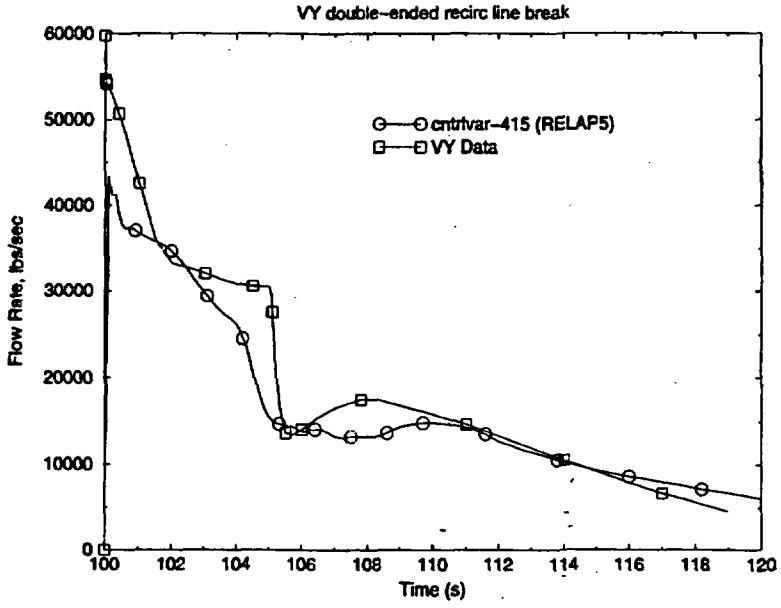
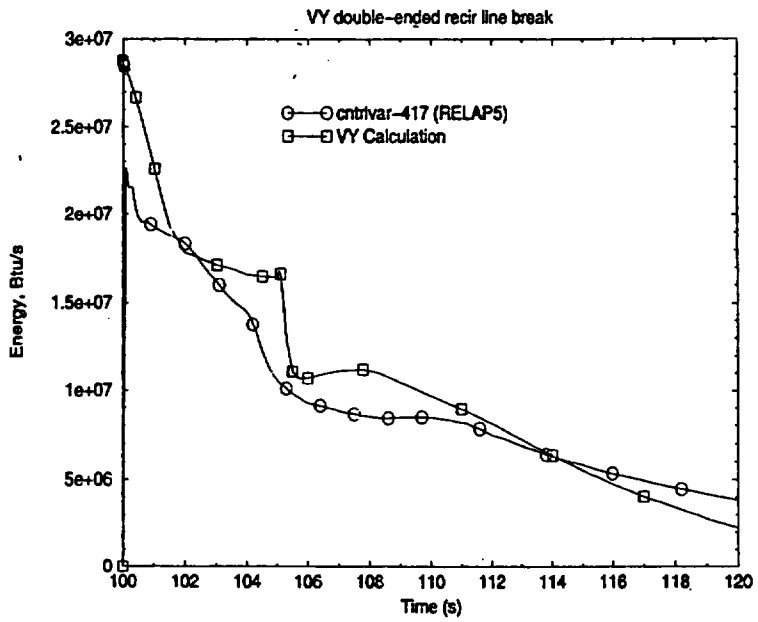


Fig. 3: Total Break Energy vs. Time



Long-Term LOCA Analysis

The long-term LOCA analysis demonstrates that the peak suppression pool temperature, wetwell pressure and wetwell air space temperature remain below their respective design limits. The results of these analyses and the acceptance criteria are provided in Table 4-1 of the PUSAR which is reproduced below.

RESULTS OF VYNPS LONG-TERM LOCA ANALYSIS

Parameter	CLTP from UFSAR	CLTP with CPPU Method	CPPU	Design Limit
Peak Bulk Pool Temperature (°F)	182.6	182.4	194.7	281
Long-term Peak Wetwell Pressure (psig)	N/A	11.1	13.9	56
Peak Wetwell Air Space Temperature (°F)*	N/A	182.4	194.7	281**

* This is the peak drywell air space temperature due to a design basis double-ended guillotine break of a recirculation suction line.

** 281°F is the structure design temperature.

The wetwell pressure peaks early in the event, and then peaks again around the time at which the wetwell temperature peaks. The value of the second peak is presented in the table.

There is a greater difference between the CLTP level and the CPPU power level values of the long-term parameters: the peak bulk pool temperature, the long-term peak wetwell pressure and the peak wetwell air space temperature. For these parameters, the difference in the initial power levels and the decay heat values play an important role.

The SHEX computer code[10] is used for the analysis of the peak suppression pool temperature, long-term peak wetwell pressure and peak wetwell air temperature. Calculations using this computer code have been accepted by the NRC staff for previous power uprate applications.

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10 MC3PT: "The General Electric Mark III Pressure Suppression Containment Analytical Model," NEDO-20533, General Electric Company, June 1974 and Supplement 1, September 1975.

SHEX: "The General Electric Pressure Suppression Containment Analytical Model," NEDO-10320, General Electric Company, April 1971; Supplement 1, May 1971; Supplement 2, January 1973.

The licensee used the American National Standards Institute/American Nuclear Society (ANSI/ANS) 5.1-1979 decay heat model with a 2σ uncertainty added.[11] The NRC has previously approved this model in many licensing applications. The licensee previously[12] combined the decay heat uncertainty with the 2% uncertainty in initial reactor power used in accordance with RG 1.49. For the EPU, the licensee applied these two uncertainties separately, which is more conservative.[13] The licensee incorporated the guidance of GE Service Information Letter 636 Revision 1[14] which recommends accounting for additional actinides and activation products and further increases the predicted decay heat.

For the long-term calculations, the minimum suppression pool water volume allowed by the TSs is used because this results in a higher pool temperature.

The peak suppression pool temperature is 194.7°F. This temperature is less than the torus design temperature of 281°F. Piping attached to the torus must be at a temperature of less than or equal to 195°F (Section 3.5.2 of the PUSAR). Therefore, this criterion is also satisfied.

The long-term wetwell air space temperature follows the suppression pool temperature and is therefore less than the design limit of 281°F. The secondary (long-term) wetwell air space pressure peak is 13.9 psig which is well below the torus design limit of 56 psig.

The most limiting drywell air space temperature is a result of small steam line breaks. The licensee examined a range of break sizes. The peak drywell air space temperature for these breaks is 337.1°F which occurs prior to containment spray. The drywell shell temperature is 271.6°F which is below the design limit of 281°F. The drywell air space temperatures are used to assess the environmental qualification of equipment. (NOTE: The peak drywell air temperatures given in the previous tables were for the design basis accident break which is the double-ended guillotine break of a recirculation suction line.)

Since the licensee used acceptable calculation methods and conservative assumptions and the calculated values are below the design limits, the NRC staff concludes that the long-term containment calculations for EPU conditions are acceptable.

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11 Shrock, V. E., A Revised ANS Standard for Decay Heat From Fission Products, *Nuclear Technology*, Volume 46, Page 323, 1979; and ANSI/ANS 5.1-1979: Decay Heat Power in Light Water Reactors, Hinsdale, IL, American Nuclear Society, 1979.

12 Entergy letter (BVY 99-45) to NRC dated March 31, 1999, "Request to Correct Safety Evaluation Report for License Amendment No. 163."

13 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-4).

14 Service Information Letter No. 636, "Additional Terms Included in Reactor Decay Heat Calculations," General Electric Nuclear Energy, May 24, 2001.

Hydrodynamic Loads

Part of the containment design basis is the acceptable response of the containment to hydrodynamic loads associated with the discharge of reactor steam into the suppression pool following a LOCA or following actuation of the safety relief valves. Analytical and empirical methods,[15] approved by the NRC staff in NUREG-0661[16] were used by the licensee to address these issues for VYNPS and to develop a plant unique structural evaluation.[17] The staff found the VYNPS resolution of these issues to be acceptable.[18]

The licensee, as part of the power uprate evaluation, needs to ensure that these analyses remain bounding. This is done for the LOCA by means of short-term calculations of the pressure and temperature response to a double-ended break of a reactor coolant system recirculation line. The key parameters are the drywell and wetwell pressure, vent flow rates and the suppression pool temperature.

Section 4.1.2.1 of the PUSAR states that:

> the short-term DBA-LOCA containment responses for CPPU are within the range of test conditions used to define the pool swell and CO [condensation oscillation] loads for VYNPS. The containment responses with CPPU, in which chugging would occur, are within the conditions used to define chugging loads. The vent thrust loads with CPPU are calculated to be less than plant-specific values defined for VYNPS.

The licensee stated that the difference in the vent thrust loads calculated at the CLTP level and at the EPU power level is due to the difference in the methods used to calculate the break flow rate and enthalpy.[19]

The licensee's evaluation of containment hydrodynamic loads as a result of a LOCA are in accordance with the CPPU topical report[20] and show acceptable results and are therefore acceptable.

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15 General Electric Company, Mark I Containment Program Load Definition Report," General Electric Topical Report NEDO-21888, Revision 0, December 1978.

16 Safety Evaluation Report, Mark I Containment Long-term Program, US NRC, NUREG 0661, July 1980.

17 Plant Unique Analysis Report of the Torus Suppression Chamber for Vermont Yankee Nuclear Power Station," Technical Report TR-5319-1, Teledyne Engineering Services, Revision 1, April 1983.

18 Letter from US NRC to J. B. Sinclair, Vermont Yankee Nuclear Power Corporation, July 2, 1984.

19 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-16).

20 GE Nuclear Energy, Licensing Topical Report NEDC-33004P-A, "Constant Pressure Power Uprate," Revision 4, dated July 31, 2003.

The dynamic loads on the suppression pool due to the discharge of steam from the safety relief valves are part of the containment design basis. The safety relief valve loads are evaluated for two cases: initial actuation and re-actuation. Because the proposed EPU does not result in an increase in the safety relief valve opening pressure, the safety relief valve loads remain unchanged. The loads due to re-actuation of the safety relief valves depend on the vacuum breaker setpoint and valve discharge line geometry. Neither of these parameters is changed by the EPU. Therefore, the existing load definition for safety relief valve initial actuation and re-actuation remains applicable.

The licensee is proposing to eliminate the limit on local suppression pool temperature due to safety relief valve discharge. The need for these limits was transmitted to licensees of BWRs by NRC GL 82-27.[21] These temperature limits were defined in NUREG 0783.[22]

The Boiling Water Reactor Owners Group (BWROG) proposed eliminating these temperature limits in two GE Topical Reports.[23] The NRC approved elimination of these temperature limits in a letter to the BWROG dated August 29, 1994.[24] The NRC safety evaluation report stated that two plant-specific items must be addressed. The first is to show that the installed quencher design is consistent with the quencher design which is used to form the basis for the approval. The second item is to show that there is no concern with steam ingestion by the ECCS pumps during accident mitigation. The NRC staff considers the first criterion to be met as a result of an acceptable staff audit of the licensee's implementation of the Mark I Containment Long-Term Program.[25] The second condition is met since the licensee has performed an analysis which demonstrates that there is "no adverse effect on the ECCS suction strainer due to steam

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21 Transmittal of NUREG-0763, "Guidelines for Confirmatory In-Plant Tests of Safety Relief Valve Discharges for BWR Plants," and NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments." Generic Letter 82-27, November 15, 1982.

22 T. M. Su, Suppression Pool Temperature Limits for BWR Containments, NUREG-0783, US NRC, November 1981.

23 NEDO-30832, "Elimination of Limit on Suppression Pool Temperature for SRV Discharge with Quenchers," General Electric Company, December 1984.

NEDO-31695 "BWR Suppression Pool Temperature Technical Specification Limits," General Electric Company, August 29, 1994.

24 NRC Letter to BWROG, "Transmittal of Safety Evaluation of General Electric Co. Topical Reports; NEDO-30832 entitled "Elimination of Limit on Local Suppression Pool Temperature for SRV Discharge With Quenchers," and NEDO-31695 entitled "BWR Suppression Pool Temperature Technical Specification Limits," dated August 29, 1994.

25 Letter from US NRC to J. B. Sinclair, Licensing Engineer, Vermont Yankee Nuclear Power Corporation, July 2, 1984.

ingestion."[26] This analysis supported the installation of large passive ECCS suction strainers in the VYNPS torus in accordance with NRC Bulletin 96-03.[27]

Since the licensee has satisfied the two criteria discussed above, the NRC staff finds the licensee's elimination of the local pool temperature limits to be acceptable.

Steam Bypass

The intent of a pressure suppression containment design is that steam generated in the drywell by a LOCA is directed through the downcomers to the suppression pool where the steam is condensed. There are leakage paths from the drywell which would direct some fraction of this steam to the suppression chamber air space, bypassing the suppression pool. This bypass steam flow must be limited to ensure that the torus design pressure is not exceeded. VYNPS TS 4.7.6.b.3 requires a leakage rate test to ensure that the bypass leakage rate limit is not exceeded.

The licensee stated that the primary factors affecting the peak containment pressure during steam bypass events are not adversely affected by the EPU. The NRC staff has reviewed the basis for the licensee's statements and finds the basis acceptable.[28]

The licensee has not proposed any changes to instrumentation and controls provided to monitor and maintain variables within prescribed operating ranges. The licensee has also not proposed any changes to instrumentation provided to monitor the reactor containment atmosphere for radioactivity that may be released from normal operations and from postulated accidents.

Conclusion

The NRC staff has reviewed the licensee's assessment of the containment temperature and pressure transient and concludes that the licensee has adequately accounted for the increase of mass and energy resulting from the proposed EPU. The NRC staff further concludes that containment systems will continue to provide sufficient pressure and temperature mitigation capability to ensure that containment integrity is maintained. The NRC staff also concludes that containment systems and instrumentation will continue to be adequate for monitoring containment parameters and release of radioactivity during normal and accident conditions and the containment and associated systems will continue to meet the requirements of draft

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26 PUSAR Section 4.1.1.1(b).

27 Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors, NRC Bulletin 96-03, May 6, 1996.

28 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-24).

GDC-10, 12, 17, 40, 42, and 49 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to primary containment functional design.

2.6.2 Subcompartment Analyses

Regulatory Evaluation

A subcompartment is defined as any fully or partially enclosed volume within the primary containment that houses high-energy piping and would limit the flow of fluid to the main containment volume in the event of a postulated pipe rupture within the volume. The NRC staff's review for subcompartment analyses covered the determination of the design differential pressure values for containment subcompartments. The NRC staff's review focused on the effects of the increase in mass and energy release into the containment due to operation at EPU conditions, and the resulting increase in pressurization. The NRC's acceptance criteria for subcompartment analyses are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of a LOCA; and (2) draft GDC-49, insofar as it requires that the containment structure, including access openings and penetrations, and any necessary containment heat removal systems be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a LOCA. Specific review criteria are contained in SRP Section 6.2.1.2.

Technical Evaluation

An annular structure of reinforced concrete enclosed in steel plate inside the drywell, called a sacrificial shield, provides thermal and radiation shielding. Section 12.3.5.2 of the VYNPS UFSAR describes the sacrificial shield as well as an analysis of the capability of the sacrificial shield to withstand the differential pressure which would develop across the wall as a result of a high pressure pipe break between the reactor vessel and the shield wall. This differential pressure is a function of the break size and the annular vent area to the rest of the drywell.

The UFSAR states that the only safe-end-to-nozzle welds, safe ends, or piping located in the annulus are small diameter lines whose rupture would result in relatively small pressure differences. Larger diameter lines are located within the shield wall penetration. This minimizes the pressure transient that would follow a rupture. The licensee conservatively assumes the failure of the largest pipe which is a 28-inch recirculation line. The UFSAR states that 100% of the energy released from the postulated break is assumed to discharge into the annulus and to distribute uniformly.

Section 4.1.2.3 of the PUSAR states that the break flow from the 28-inch recirculation line at EPU conditions increases due to the increased subcooling of the water initially in the recirculation loops. The pressure differential due to this increased subcooling is given by the licensee as less than 3 psid. The current licensing basis calculation gives the pressure difference as 110 psid. The design capability of the shield wall is 134 psid. The peak pressure

difference remains below the design capability of the sacrificial shields with the 3 psid increase. The UFSAR states that this design pressure does not credit the strength of the concrete filler between the steel plates or the 27-inch vertical beams to which the steel plates are welded.

Conclusion

The NRC staff has reviewed the subcompartment assessment performed by the licensee and the change in predicted pressurization resulting from the increased mass and energy release. The NRC staff concludes that containment structure, system, and components important to safety will continue to be protected from the dynamic effects resulting from pipe breaks and that the subcompartments will continue to have sufficient margins to prevent fracture of the structure due to pressure difference across the walls following implementation of the proposed EPU. Based on this, the NRC staff concludes that the plant will continue to meet draft GDC-40, 42, and 49 for the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to subcompartment analyses.

2.6.3 Mass and Energy Release

2.6.3.1 Mass and Energy Release Analysis for Postulated Loss-of-Coolant

Regulatory Evaluation

The release of high-energy fluid into containment from pipe breaks could challenge the structural integrity of the containment, including subcompartments and systems within the containment. The NRC staff's review covered the energy sources that are available for release to the containment and the mass and energy release rate calculations for the initial blowdown phase of the accident. The NRC's acceptance criteria for mass and energy release analyses for postulated LOCAs are based on (1) draft GDC-49, insofar as it requires that the containment structure be designed to accommodate, without exceeding the design leakage rate, the pressures and temperatures resulting from the largest credible energy release following a LOCA; and (2) 10 CFR Part 50, Appendix K, insofar as it identifies sources of energy during a LOCA. Specific review criteria are contained in SRP Section 6.2.1.3.

Technical Evaluation

The mass and energy release following a high energy line break in containment is discussed under Primary Containment Functional Design (SE Section 2.6.1).

10 CFR Part 50 Appendix K requires that the following sources of energy be considered in performing LOCA analyses: initial stored energy in the fuel, fission heat, decay of actinides, fission product decay, metal-water reaction and reactor internals heat transfer.

Although containment analyses are not required to comply with Appendix K, the licensee did include these sources of energy in calculating the mass and energy release to the containment.

Conclusion

The NRC staff has reviewed the licensee's mass and energy release assessment and concludes that the licensee has adequately addressed the effects of the proposed EPU and appropriately accounts for the sources of energy identified in 10 CFR Part 50, Appendix K. Based on this, the NRC staff finds that the mass and energy release analysis meets the requirements in draft GDC-49 for ensuring that the analysis is conservative. Therefore, the NRC staff finds the proposed EPU acceptable with respect to mass and energy release for postulated LOCA.

2.6.4 Combustible Gas Control in Containment

Regulatory Evaluation

Following a LOCA, hydrogen and oxygen may accumulate inside the containment due to chemical reactions between the fuel rod cladding and steam, corrosion of aluminum and other materials, and radiolytic decomposition of water. If excessive hydrogen is generated, it may form a combustible mixture in the containment atmosphere. The NRC staff's review covered (1) the production and accumulation of combustible gases, (2) the capability to prevent high concentrations of combustible gases in local areas, (3) the capability to monitor combustible gas concentrations, and (4) the capability to reduce combustible gas concentrations. The NRC staff's review primarily focused on any impact that the proposed EPU may have on hydrogen release assumptions, and how increases in hydrogen release are mitigated. The NRC's acceptance criteria for combustible gas control in containment are based on (1) 10 CFR 50.44, insofar as it requires that plants be provided with the capability for controlling combustible gas concentrations in the containment atmosphere; (2) draft GDC-62, insofar as it requires that all critical parts of containment air cleanup systems, such as ducts, filters, fans, and dampers be designed to permit physical inspection; and (3) draft GDC-63, 64, and 65, insofar as they require that active components of the air cleanup systems be designed to permit appropriate periodic testing. Specific review criteria are contained in SRP Section 6.2.5.

Technical Evaluation

10 CFR 50.44, "Combustible gas control for nuclear power reactors," was amended effective October 16, 2003. This is after the date of the licensee's EPU application which is dated September 10, 2003. The licensee has not revised the original combustible gas analysis submitted in the PUSAR. This is acceptable since nothing in the amended regulation affects the analyses performed by the licensee to support the EPU in this area.

By letter dated June 17, 2004,[29] the licensee requested an amendment to the VYNPS license to delete the TS requirements associated with the hydrogen and oxygen monitors. This change is

29 Entergy letter (BVY 04-053) to NRC dated June 17, 2004, "Vermont Yankee Nuclear Power Station, Proposed Technical Specification Change No. 265, "Eliminate Requirements for Hydrogen/Oxygen Monitors Using the Consolidated Line Item Improvement Process."

consistent with the revised regulation. The NRC staff approved this request in an amendment dated December 14, 2004.[30]

The post-LOCA production of hydrogen and oxygen by radiolysis increases proportionally with the power level. The hydrogen concentration in containment is controlled by the Containment Atmosphere Dilution System (CADS) which is described in Section 5.2.7 of the VYNPS UFSAR. Because of the increased production of hydrogen and oxygen due to the EPU, the system must be started sooner after the beginning of the accident. This does not significantly affect operator response since the system is not required for many hours after accident initiation.

The licensee analyzed the post-LOCA control of combustible gases at CPPU conditions. The results are given in Section 4.7 and in Figures 4-1 through 4-5 of the PUSAR. For VYNPS at EPU conditions, the start time of the CADS following a LOCA is 37 hours which is adequate time for effective operator response. The licensee states that the addition rate of nitrogen to maintain the containment below the 5% oxygen lower flammability limit remains within the delivery capability of the CADS. The containment repressurization limit of half the design pressure (28 psig) is reached in 35 days due to the increase in pressure due to nitrogen addition to the containment. This is within the VYNPS design basis.

The licensee has not proposed any changes to critical parts of containment air cleanup systems, such as ducts, filters, fans, and dampers. The licensee has also not proposed any changes to the active components of the air cleanup systems.

Conclusion

The NRC staff has reviewed the licensee's assessment related to combustible gas and concludes that the plant will continue to have sufficient capabilities consistent with the requirements in 10 CFR 50.44 and draft GDC-62, 63, 64, and 65 as discussed above. Therefore, the NRC staff finds the proposed EPU acceptable with respect to combustible gas control in containment.

2.6.5 Containment Heat Removal

Regulatory Evaluation

RHR systems are provided to remove heat from the containment atmosphere and from the water in the containment wetwell. The NRC staff's review in this area focused on (1) the effects of the proposed EPU on the analyses of the available NPSH to the RHR system pumps. The NRC's acceptance criteria for containment heat removal at VYNPS are based on draft GDC-41 and 52, insofar as they require that a containment heat removal system be provided, and that its function shall be to prevent exceeding containment design pressure under accident

30 Letter from Richard B. Ennis (NRC) to Michael Kansler (Entergy) dated December 14, 2004, VYNPS Amendment No. 220, "Elimination of Requirements for Hydrogen/Oxygen Monitors."

conditions. Specific review criteria are contained in RG 1.82, Revision 3.[31] This revision of the RG was issued after the licensee's EPU amendment request was submitted to the NRC. However, the guidance in the area of NPSH is the same as that of the draft version of this RG which was issued before the licensee's EPU submittal. The licensee has not requested that RG 1.82, Revision 3, become part of the VYNPS licensing basis; however, the licensee has addressed the degree of conformity with the guidance of this RG in a letter to the NRC and concluded that the request for EPU meets the intent of this RG's positions on NPSH for the RHR and core spray pumps.[32]

Technical Evaluation

The RHR system is described in Section 4.8.5 of the VYNPS UFSAR. The RHR system consists of two trains. Each train consists of two pumps and one heat exchanger. The heat exchanger is cooled by one train of the RHR service water system.

The RHR system has several different modes of operation. It is aligned during normal operation as a low-pressure coolant injection (LPCI) system as part of the ECCS. The RHR pumps are sized for this function. The RHR system also cools the RCS during a normal shutdown and cooldown. The RHR heat exchangers are sized based on this function. The RHR system also cools the suppression pool following a design-basis LOCA by pumping the suppression pool water through the RHR heat exchangers and returning the water to the suppression pool, or by diverting the suppression pool water to spray headers in the drywell and wetwell after it has passed through the RHR heat exchangers.

The core spray system consists of two trains. Each train contains one core spray pump. The core spray pumps take suction from the suppression pool. The water is sprayed into the vessel above the core and is returned to the suppression pool by flowing out the break.

Available NPSH

An important consideration in the operation of the core spray and RHR pumps is the available NPSH. Adequate available NPSH is important in ensuring that the pump will deliver the flow assumed in the safety analyses at the expected discharge pressure. In order to ensure acceptable flow and discharge pressure, the available NPSH must be equal to or greater than the required NPSH. The required NPSH is a function of the pump design and is determined by the pump vendor.

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31 Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident, Regulatory Guide 1.82 Revision 3, US NRC, November 2003.

32 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-21).

The available NPSH is calculated from the equation:

$$\text{Available NPSH} = h_{atm} + h_{static} - h_{loss} - h_{vapor}$$

where:

h_{atm} = head on the surface of the suppression pool

h_{static} = the head due to the difference in elevation between the suppression pool surface and the centerline of the pump suction

h_{loss} = the head loss due to fluid friction, fittings in the flow path from the suppression pool to the pump, and the suction strainers which prevent ingestion of debris into the pumps

h_{vapor} = head due to the vapor pressure of the suppression pool water at the suppression pool water temperature

The increase in power as a result of the EPU results in an increase in the suppression pool temperature following the design-basis LOCA. The increased water temperature reduces the available NPSH of the RHR pumps and the core spray pumps since the vapor pressure of the suppression pool water (or h_{vapor}) increases. The licensee has proposed to compensate for this reduction in available NPSH where necessary by crediting the containment accident pressure, that is, by increasing h_{atm}. Previously, the VYNPS NPSH analyses assumed that the containment pressure remained constant at its pre-accident value.

There is no regulation that prohibits the licensee from crediting containment accident pressure in determining available NPSH for safety related pumps. The NRC has allowed this credit in NPSH analyses for other reactors, for both PWRs and BWRs. To credit containment accident pressure in determining available NPSH, licensees must demonstrate by means of conservative analyses that sufficient containment pressure will be available when required during the postulated accident. In addition, the staff has also examined the increase in risk at EPU conditions due to crediting containment accident pressure. The risk aspects are discussed in SE Section 2.13.

In order to calculate the available NPSH for a specific scenario, the containment conditions must first be determined (i.e., the containment pressure and suppression pool temperature). The licensee calculates the containment conditions for the LOCA events with the GE SHEX computer code. The containment conditions for the ATWS events are calculated with the GE STEMP computer code. As noted in PUSAR Table 1-1, the STEMP code uses fundamental mass and entergy conservation laws to calculate the suppression pool heatup. The use of STEMP was noted in GE Topical Report NEDE-24222, "Assessment of BWR Mitigation of ATWS," dated December 1979. The code has been used in ATWS applications since that time. The analytical models of STEMP have been accepted by the NRC in previous

applications and other ATWS analyses. The containment conditions for the station blackout and Appendix R fire events are calculated with the GOTHIC computer code.[33]

GOTHIC is a general purpose thermal hydraulics computer program for the analysis of nuclear power plant containments. It was developed for the Electric Power Research Institute by Numerical Applications, Incorporated (NAI). NAI validated GOTHIC by comparison with analytical solutions and experimental data. The NRC has previously approved GOTHIC containment analyses. The licensee stated[34] that the Appendix R and station blackout suppression pool temperature analyses are consistent with the conditions and limitations of a previous staff review of GOTHIC 7.0.[35] The licensee stated that VYNPS follows the guidance of Generic Letter (GL) 83-11 and GL 83-11, Supplement 1 and provided supporting information.[36] These generic letters deal with the capability of licensees to perform calculations with complex computer codes including the training and qualification of the licensee personnel, benchmarking calculations and quality assurance. The NRC staff finds the licensee's use of GOTHIC to be acceptable.

The licensee uses the SHEX code to calculate the drywell and wetwell pressures and temperatures as well as suppression pool temperature as a function of time. SHEX has been used in many other similar licensing actions, including power uprates, approved by the NRC staff.

The NRC staff performed an audit calculation of the VYNPS SHEX predictions of containment conditions for NPSH analyses using the NRC computer code CONTAIN 2.0.[37] The results are shown in the following figure. The details are provided in an NRC internal memorandum from the NRC Office of Nuclear Regulatory Research to the NRC Office of Nuclear Reactor Regulation.[38] The figure shows close agreement between the licensee's SHEX calculations and those done with CONTAIN 2.0.

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33 GOTHIC Containment Analysis Package NAI 8907 Version 7.0 Numerical Applications, Inc.

34 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-14).

35 Issuance of Kewaunee Nuclear Power Plant License Amendment No. 169, September 29, 2003.

36 Letter from Jay K. Thayer, Site Vice President, Entergy Nuclear Vermont Yankee, LLC, to US NRC, BVY 04-058, Technical Specification Proposed Change No. 263, Extended Power Uprate, Supplement 8, July 2, 2004, Response to RAI SPSB-C-14.

37 K. K. Murata, et al., "Code Manual for CONTAIN 2.0: A Computer Code for Nuclear Reactor Containment Analysis, Prepared for the US NRC, NUREG/CR-6533, June 1997.

38 Memorandum to James E. Lyons, Director, Division of Systems Safety and Analysis, Office of Nuclear Reactor Regulation, US NRC, from Farouk Eltawila, Director, Division of Systems Analysis and Regulatory Effectiveness, Office of Nuclear Regulatory Research, US NRC, July 7, 2005 (ADAMS Accession Number ML050100337).

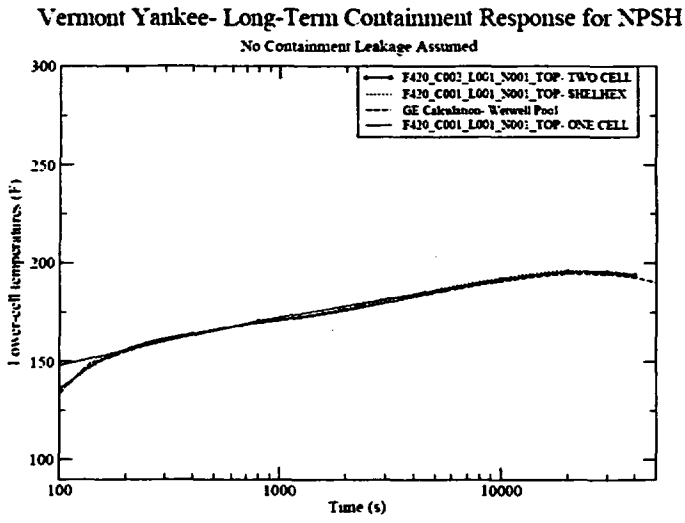

Table SPSB-C-32-1, which is part of the licensee's response to RAI SPSB-C-32[39] provides a summary of the input used to calculate the containment conditions for the NPSH calculations. The licensee, as discussed further below, used conservative assumptions for these calculations, that is, assumptions that underestimate the available NPSH. The licensee submitted Calculation VYC-0808, Revision 8 for staff review.[40] (The licensee had previously submitted an earlier revision of this calculation.[41]) This calculation predicts the available NPSH

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39 Entergy letter (BVY 04-081) to NRC dated August 12, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 11, Extended Power Uprate - Response to Request for Additional Information."

40 Entergy letter (BVY 04-106) to NRC dated October 5, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 18, Extended Power Uprate - ECCS Pump Net Positive Suction Head Margin" (Attachment 1, "Core Spray and Residual Heat Removal Pump Net Positive Suction Head Margin Following a Loss-of-Coolant Accident and an Anticipated Transient Without Scram with Fibrous Debris on the Intake Strainers)."

41 VYC-0808 Revision 6 was submitted by Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information." Entergy letter (BVY 04-071) to NRC dated July 27, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 9, Extended Power Uprate - Revised Containment Overpressure Envelope," provided updated short-term and long-term containment temperature and pressure profiles based on a more conservative assumption of the containment spray mixing efficiency.

of the RHR and ECCS core spray pumps for the LOCA and the ATWS events. The calculation compares the calculated value of available NPSH with the required NPSH for each pump type (RHR and core spray). In cases where the available NPSH is less than the required NPSH, containment accident pressure was credited so that the available NPSH equals the required NPSH. The table below provides a summary of the containment accident pressure credited as a function of time for the large-break LOCA. It is an abbreviated version of a similar table in Calculation VYC-0808, Revision 8.

SUMMARY OF REQUIRED CONTAINMENT ACCIDENT PRESSURE AND THE AVAILABLE CONTAINMENT PRESSURE FOLLOWING A LARGE-BREAK LOCA

Time (sec)	Available Containment Accident Pressure (psig)	RHR Pump: Required Pressure (psig)	Core Spray Pump: Required Pressure (psig)
786	3.01	0.78	0
1098	3.24	1.07	0.23
2962	3.87	2.31	1.47
6275	6.12	3.53	2.7
10220	7.16	4.3	3.47
15170	7.36	4.83	4
25120	7.77	5.06	4.23
30095	7.72	4.98	4.15
35065	7.63	4.85	4.02
40020	7.5	4.66	3.83
70342	6.02	3.05	2.22
100340	4.82	1.8	0.97
140302	3.77	0.88	0.04
150302	3.57	0.71	0
180302	3.02	0.23	0
196552	2.73	0	0
200302	2.67	0	0

Pressure margin remains between the amount of containment accident pressure credited and the containment accident pressure calculated to be available at that time in the accident.

In order to ensure a conservative calculation, the licensee's calculated containment accident pressure is underestimated and the suppression pool temperature is overestimated. Other conservative assumptions affect the loss term in the equation for available NPSH. VYC 0808, Revision 8 describes some of the conservatism in this calculation. A further discussion of the conservatism in the NPSH calculations is discussed in a licensee submittal[42] and evaluated below.

Initial conditions are chosen for the analyses in such a way that containment pressure is minimized and suppression pool temperature is maximized. For example, the initial drywell air temperature and the initial torus air space temperature are at their maximum values, the initial drywell and torus air pressures are at their minimum values and the initial drywell and torus relative humidities are equal to 100%. These values minimize the initial air mass which results in a lower accident pressure. The initial suppression pool temperature and the service water temperature (which is the heat sink for the heat generated in containment) are at their maximum values. This maximizes the suppression pool temperature during the LOCA.

The initial suppression pool level is assumed to be at its minimum TS value. This results in a higher suppression pool accident temperature and also reduces the positive contribution of the water height above the pump suction in the available NPSH calculation (h_{static}). The suppression pool level increases during the LOCA due to thermal expansion and water addition from the ECCS and feedwater. The NPSH calculation is based on a single value of water level corresponding to a pool volume of 77,640 ft^3. At the time of peak torus temperature, the calculated volume is 79,470 ft^3. The difference in water level corresponding to this difference is approximately 0.25 ft.[43] This also reduces the positive contribution of the water height above the pump suction in the calculation of available NPSH.

The mixing of the break flow with the drywell atmosphere and drywell and wetwell spray flow with the respective atmospheres is modeled in a way that minimizes the drywell and wetwell pressures.[44] This can have a significant conservative effect.

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42 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information."

43 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-11).

44 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-11).

Required NPSH increases as the flow rate increases. The flow losses in the suction piping (h_{loss}) also increase with increasing flow rate. Both of these effects reduce the margin between the required and the available NPSH. The licensee states that:[45]

> For conservatism, the [pump flow] values used in the NPSH analysis for LOCA are upper bound values. The upper bound values are based on statistical uncertainty associated with the flow measurements performed during periodic surveillances.

The licensee compared the flow rates assumed in the NPSH analyses to flow rates used in the LOCA analyses which determine compliance with 10 CFR 50.46.[46] The LOCA analyses use lower bound pump flow rates for conservatism.[47] The pump flow rates assumed in the NPSH analyses are upper bound values.

The licensee determined that the worst single failure for the NPSH analyses is loss of one RHR heat exchanger (due, for example, to failure of the heat exchanger outlet valve to open).[48] The licensee showed that without this single failure, credit for containment accident pressure is not required.[49] Since this worst single failure leaves one train for LPCI pumps and two core spray pumps available for injection into the vessel, more heat from the RCS is transferred sooner to the suppression pool (and more pump heat is also transferred to the suppression pool). Licensee calculations show that this results in a significant increase in the suppression pool temperature.[50]

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45 Letter from Jay K. Thayer, Site Vice President, Entergy Nuclear Vermont Yankee, LLC, to US NRC, BVY 04-058, Technical Specification Proposed Change No. 263, Extended Power Uprate, Supplement 8, July 2, 2004, Response to RAI SPSB-C-9.

46 Entergy letter (BVY 05-072) to NRC dated August 1, 2005, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 30, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-51).

47 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-9).

48 Entergy letter (BVY 05-017) to NRC dated February 24, 2005, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 23, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-38).

49 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-10).

50 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-10).

The licensee also discussed several other conservatisms including the use of cycle independent (bounding) values for decay heat and minimum RHR flow and RHR service water flow (as they affect heat exchanger performance).[51]

The containment leakage rate was included in the NPSH analyses. The containment is assumed to leak at a rate of 1.5% per day. This includes leakage of the main steam isolation valves. This has only a minor effect on the containment pressure.

The effectiveness of the RHR heat exchanger is represented by a parameter K. K increases slightly with the EPU due to the calculated increase in suppression pool temperature (approximately 1.9%).

The licensee states that:[52]

> The RHR Hxs [heat exchangers] were previously tested and the test results analyzed per the guidance of NRC Generic Letter 89-13.[53] The testing showed that the heat exchangers' performance easily met their design heat removal requirements (much more than 1.9 percent) based on once per operating cycle cleaning. In accordance with the provisions of GL 89-13 the heat exchangers are cleaned once per operating cycle to maintain this level of performance ...

The licensee includes 5% tube plugging and design fouling in the RHR heat exchanger heat rate so that the K value is underestimated.[54]

The NRC staff finds the licensee's modeling of the RHR heat exchangers to be acceptable since it complies with regulatory guidance and is conservatively determined.

The licensee calculated the head losses due to flow friction in the piping and losses due to fittings (valves, piping tees, etc.) using standard industry methods.[55] By using an upper bound

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51 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-10).

52 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-7).

53 US NRC Generic Letter 89-13, Service Water System Problems Affecting Safety Related Equipment, July 18, 1989.

54 Entergy letter (BVY 04-074) to NRC dated July 30, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 10, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-29).

55 Technical paper No. 410, Flow of Fluids Though Valve Fittings and Pipe, Crane Co., 24th Printing, 1988.

on the pump flow, the licensee also conservatively accounts for flow losses since these vary roughly as the square of the flow rate. The head losses due to friction and fittings are therefore acceptable.

The head loss in available NPSH calculations also considers the debris loading on the ECCS suction strainers. The debris may be generated as a result of the coolant discharge from the ruptured pipe impacting thermal insulation and paint. These strainers are placed in the suppression pool at the pump suction locations to prevent material from entering the pumps which could damage the pumps or in another way interfere with cooling the core. The licensee added large passive strainers in response to NRC Bulletin 96-03.[56] [57] These strainers were designed using BWROG methods which were approved by the NRC.[58]

Sludge is a particulate material which deposits in the suppression pool. It is composed primarily of rust. It is transported to the ECCS suction strainers and combines with the insulation debris on the strainers. The sludge generation rate assumed in the suction strainer head loss calculations was increased from 53 pounds per year to 88 pounds per year for a one-time extension of the suppression pool cleaning. The licensee estimates the actual sludge generation rate as 12.5 pounds per year.[59]

The licensee states that no insulation was added or removed from the drywell which could adversely affect the results of the debris loading calculations. Also, no additional paint has been added to the suppression chamber or drywell that could adversely affect the input to the debris loading calculations.[60]

The ECCS suction strainer head loss values used for the LOCA and ATWS events were provided by the licensee.[61] The licensee states that the maximum predicted head loss values

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56 Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors, NRC Bulletin 96-03, May 6, 1996.

57 Letter from Vermont Yankee Nuclear Power Corporation to US NRC, Suction Strainers in Accordance with NRC Bulletin 96-03, BVY 99-164, December 29, 1999.

58 Utility Resolution Guide for ECCS Suction Strainer Blockage, NEDO 32686-A GE Nuclear Energy, Prepared for the BWR Owners' Group, October 1998.

59 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-6).

60 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-6).

61 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-11).

for the core spray and the RHR strainers are based on conservative debris loads, fluid temperatures, and flow rates. The licensee used these original design basis head loss values rather than revised values which were calculated by the licensee as part of an ECCS suction strainer head loss assessment. The revised values more accurately reflect the debris loading, suppression pool temperature and pump flow. However, the original values were used in the CPPU analyses since they are more conservative. These values are compared in the table below.[62]

Pump Strainer	Debris Head Loss (ft)	
	Used in CPPU analyses	Revised: using more current data
RHR	0.33	0.26
Core Spray	0.21	0.08

For the head loss calculations, the licensee used a suppression pool temperature of 173°F for suppression pool temperatures greater than 173°F. This is conservative since the head loss is inversely proportional to the temperature. When the temperature is less than 173°F, the licensee uses the factor 173/T (where T is the suppression pool temperature) which accounts for the increases in head loss with decreasing temperature. (Note: At 173°F credit for containment pressure is no longer necessary.) The maximum expected flow rates are assumed for the head loss calculations, which is conservative.[63]

The licensee's NPSH calculations assume that, after 10 minutes, one core spray pump provides cooling to the core and one RHR pump cools the suppression pool. The licensee assumes that:

> the remaining debris in the suppression pool and any debris deposited on an active strainer supplying pump(s) in the short-term that is subsequently secured for the long-term is deposited on the two active strainers in proportion to their flow rates. The total debris thus deposited on the two active strainers is used to determine the NPSH margin at the peak suppression pool temperature.

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62 Entergy letter (BVY 04-106) to NRC dated October 5, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 18, Extended Power Uprate - ECCS Pump Net Positive Suction Head Margin" (Calculation VYC-0808 Revision 8 Attachment 2).

63 Entergy letter (BVY 04-106) to NRC dated October 5, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 18, Extended Power Uprate - ECCS Pump Net Positive Suction Head Margin" (Calculation VYC-0808 Revision 8 Section 3.2).

Although this assumed redistribution of debris deposited on the strainers is conservative (since the debris is redistributed from the inactive strainers, where it would have no effect, to the active strainers where it contributes to the strainer head loss), the licensee did not quantify the degree of conservatism. The strainer head loss is not a significant consideration for these VYNPS calculations.

Required NPSH

The required NPSH values for the RHR and core spray pumps are based on curves provided by the pump vendor (Sulzer Bingham Pumps, Inc.) based on information supplied by the licensee. The curves are provided in Attachments 3 and 5 to Calculation VYC-0808, Revision 8 [64] and were the subject of several staff requests for additional information.[65]

The curves are labeled as available NPSH versus pump operating time (in hours). The licensee uses the available NPSH obtained from these curves as the minimum NPSH that must be available; in other words, it is equivalent to the required NPSH. Therefore, the curves provide the required NPSH as a function of the pump operating time that will preclude any damage to the RHR and core spray pumps.

The value of required NPSH supplied by the pump vendor is constant for the first 7 hours of operation. The pump vendor states[66] that when operating the RHR pumps for 7 hours at 7000 gallons per minute (gpm) with the required NPSH at 23 to 24 feet, the pumps will be in the cavitation mode and the head drop will be above 6% but will still be greater than the original minimum operational NPSH. The minimum operational NPSH is one of the acceptance criteria to demonstrate the pump met contractual requirements. When operating the core spray pumps for the first 7 hours at 3000 gpm with an NPSH of 24 to 25 feet, the head drop will be less than

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64 Entergy letter (BVY 04-106) to NRC dated October 5, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 18, Extended Power Uprate - ECCS Pump Net Positive Suction Head Margin" (Calculation VYC 0808 Revision 8 Attachments 3 and 5).

65 Entergy letter (BVY 05-072) to NRC dated August 1, 2005, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 30, Extended Power Uprate - Response to Request for Additional Information" (Responses to RAI SPSB-C-47 to SPSB-C-49).

Entergy letter (BVY 05-074) to NRC dated August 4, 2005, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 31, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-51).

Entergy letter (BVY 05-017) to NRC dated February 24, 2005, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 23, Extended Power Uprate - Response to Request for Additional Information" (Responses to RAI SPSB-C-41 to SPSB-C-44).

66 Entergy letter (BVY 04-106) to NRC dated October 5, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 18, Extended Power Uprate - ECCS Pump Net Positive Suction Head Margin" (Calculation VYC 0808 Revision 8 Attachment 5, Sulzer Bingham Pumps, Inc., Document E12.5.561 NPSH/Minimum Flow Study Summary Report, May 26, 1998).

3%. In both cases, the pump vendor states that the pumps are within acceptable limits of the NPSH knee (the region in which the pump head shows a reduction).

After the first 7 hours, the required NPSH increases in a ramp to a constant value which is maintained from 100 hours to 8000 hours (333 days).[67] The constant value for the RHR pumps corresponds to a 3% drop in head. For the core spray pumps, the constant value corresponds to a head loss of between 1% and 3%. The standard definition of required NPSH, given in industry standards, corresponds to a 3% head loss.

These values are based on testing performed by the pump vendor on the same and similar pumps. Pumps were considered similar when the pump characteristics affecting NPSH were the identical. Other pump characteristics were adjusted to those of the VYNPS pumps (e.g., pump speed and impeller diameter) by pump affinity laws.

Based on the pump vendor's recommendations, which are based on VYNPS pumps and similar pumps, the NRC staff finds the licensee's use of the reduced required NPSH values (i.e., up to a 6% head drop) to be acceptable.

The licensee uses the reduced required NPSH values in determining the NPSH margin for the short-term (<600 seconds) LOCA, ATWS, station blackout and the Appendix R fire.

The licensee uses the constant (8000-hour) required NPSH value for the long-term LOCA NPSH analysis, which is the limiting NPSH event. The 8000-hour required NPSH values for the RHR and core spray pumps provided by the licensee are shown in the following table[68]:

Pumps	Flow Rate (gpm)	Upper Limit Required NPSH (Long-term) (ft)
One RHR pump	7400	31.7
One core spray pump	3500	29.6

Although the use of the reduced required NPSH values is acceptable, adequate NPSH margin is obtained for VYNPS at EPU conditions for the short-term LOCA and ATWS events if, instead

67 Entergy letter (BVY 04-106) to NRC dated October 5, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 18, Extended Power Uprate - ECCS Pump Net Positive Suction Head Margin" (Calculation VYC 0808 Revision 8 Attachment 5, Sulzer Bingham Pumps, Inc., Document E12.5.561 NPSH/Minimum Flow Study Summary Report, May 26, 1998).

68 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-12).

of applying these values, the 8000-hour required NPSH values are used. For example, the tables below for the short-term LOCA response and the ATWS response show that although applying the 8000-hour upper limit required NPSH rather than the reduced required NPSH increases the credited containment accident pressure, the credited pressure remains below the containment accident pressure conservatively calculated to be available.

NPSH Margin for the Short-Term LOCA with and without Reduced Required NPSH						
Pumps	NPSHA (ft)	Reduced required NPSH (ft)	8000-hr value of required NPSH (ft)	Available Containment accident pressure (psig)	Containment pressure credited by licensee using reduced value of required NPSH (psig)	Containment pressure needed if the required NPSH used is the 8000-hr value (psig)
CS	28.44	28	35.0	2.94	0	2.77
1 RHR	31.12	23.8	31.7	2.94	0	0.25
2 RHR	28.75	23.6	30.0	2.94	0	0.53

NPSH Margin for the ATWS Event With and Without Reduced Required NPSH at Peak Suppression Pool Temperature						
Pumps	NPSHA (ft)	Reduced required NPSH (ft)	8000-hr value of required NPSH (ft)	Available containment accident pressure (psig)	Containment pressure credited by licensee using reduced value of required NPSH (psig)	Containment pressure needed if the required NPSH used is the 8000-hr value (psig)
1 RHR	20.53	23.8	31.7	12.5	1.37	4.68

Summarizing, the licensee's use of reduced values of required NPSH is acceptable. However, it is not necessary for the short-term LOCA and the ATWS, as adequate margin is available if, as is done for the long-term large-break LOCA calculations, the upper limit (8000-hr) value of required NPSH is used.

Postulated Accidents Other than the LOCA

The licensee addressed the effect of a stuck open relief valve on available NPSH.[69] The licensee calculated the peak suppression pool temperature to be 182.1°F. This is less than the current peak suppression pool temperature of 182.6°F. Therefore, adequate available NPSH exists without credit for containment accident pressure.

The licensee also calculated the available NPSH at EPU conditions for the Appendix R and station blackout scenarios. As mentioned earlier, the containment calculations were performed using the GOTHIC computer code.

The assumptions used for the Appendix R fire and station blackout containment analyses are discussed in Supplement 8 to the VYNPS EPU license amendment request.[70] The licensee originally credited containment accident pressure for available NPSH for these events. The licensee subsequently revised the analyses for both events by crediting two RHR service water pumps rather than one.[71] These revised analyses showed that credit for containment accident pressure was no longer required. The analyses still credit the reduced required NPSH values for the core spray and RHR pumps. The licensee stated that the use of two RHR service water pumps can be credited during the Appendix R and station blackout events because TS 3.5.C requires a full complement of RHR service water equipment during normal power operation and a postulated loss of an emergency bus would still ensure the availability of two RHR service water pumps.

Because the torus vent valve, which is normally open, does not receive an automatic isolation signal for the Appendix R fire and station blackout events, the licensee originally proposed an operator action to close this valve to maintain containment accident pressure.[72] However, since

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69 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-27).

70 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-14).

71 Entergy letter (BVY 05-030) to NRC dated March 24, 2005, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 25, Extended Power Uprate - Station Blackout and Appendix R Analyses."

72 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information."

credit for containment accident pressure is no longer required, this operator action is no longer required.[73] (For the LOCA, an automatic signal shuts the valve.)

The licensee credits containment accident pressure in calculating the available NPSH of the RHR pumps following an ATWS. The licensee examined two limiting events, main steam isolation valve closure and pressure regulator failed open. The containment temperature and pressure profiles for both events were similar enough that the licensee combined the pressure and temperature values and analyzed the combined event. Both RHR loops are available to cool the suppression pool since ATWS events may be analyzed with realistic assumptions. Nevertheless, the licensee assumed the RHR pump flow rate for the ATWS event is the maximum value assumed in the LOCA analysis (7400 gpm). The suppression pool volume is conservatively assumed to be at the TS minimum value. The licensee accounted for the effect of debris on the suction strainer head loss in the ATWS NPSH calculation. The licensee stated that "there is no high energy line break to dislodge insulation."[74] However, the VYNPS safety valves discharge into the containment.[75] Therefore, the licensee's accounting for debris generation is appropriate.

An NRC inspection conducted at VYNPS found that the licensee had not established the correct condensate storage tank water temperature for use in plant transient analyses.[76] The inspection report discussed the relief valve discharge transient. The finding was of very low safety significance since adequate available NPSH for the core spray pumps remained. As a result of this finding, the licensee revised the ATWS analysis to take into account the higher suppression pool temperature resulting from the assumed change in condensate storage tank temperature (from 117°F to an assumed value of 135°F). The licensee estimated that this 18°F increase in condensate storage tank water temperature results in 0.5°F increase in suppression pool temperature and a change in containment pressure of "no more than 0.2 psi."[77] The effect of the change in condensate storage tank temperature is acceptable since the peak suppression pool temperature as a result of the ATWS was previously calculated to be

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73 Entergy letter (BVY 05-030) to NRC dated March 24, 2005, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 25, Extended Power Uprate - Station Blackout and Appendix R Analyses."

74 Entergy letter (BVY 04-106) to NRC dated October 5, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 18, Extended Power Uprate - ECCS Pump Net Positive Suction Head Margin" (Calculation VYC 0808 Revision 8).

75 VYNPS UFSAR, Revision 19, Section 4.4.5.

76 Letter from Wayne D. Lanning (NRC) to Jay K. Thayer (Entergy) dated December 2, 2004. Vermont Yankee Nuclear Power Station, NRC Inspection Report 05000271/2004008.

77 Entergy letter (BVY 04-106) to NRC dated October 5, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 18, Extended Power Uprate - ECCS Pump Net Positive Suction Head Margin" (Calculation VYC 0808 Revision 8 Attachment 9).

190°F and the peak suppression pool temperature for the limiting event, the LOCA, is 194.7°F. Thus, the limiting temperature will not be exceeded.

Impact on Operator Response

The licensee described the effect on emergency operating procedures of crediting containment accident pressure for available NPSH.[78] The VYNPS emergency operating procedures currently contain a series of curves of suppression pool temperature as a function of pump flow with containment overpressure (containment accident pressure) as a parameter. There are separate curves for the RHR and the core spray pumps. The licensee states that the emergency operating procedures do not require revision to ensure that the containment accident pressure will not be reduced below the pressure required for adequate available NPSH. The emergency operating procedures, as currently written, provide guidance to the operator to ensure containment isolation and to remain aware of the status of RHR and core spray pump NPSH. The licensee described some of the assumptions used to calculate the emergency operating procedure NPSH curves.[79] The NRC staff agrees that credit for containment accident pressure will not adversely affect the VYNPS emergency operating procedures since it is already a part of these procedures at the current power level.

The licensee provided a figure with several curves of containment pressure as a function of time for the LOCA.[80] The figure shows the available containment accident pressure (overpressure) curve. Curves of required overpressure for the RHR and core spray pumps are below this curve. Between the containment overpressure curve and the curves of required overpressure for the RHR and core spray pumps is a stepped line which is the amount of overpressure the licensee is crediting in calculating the available NPSH. The operators would not use this figure. It is not included in the emergency operating procedures and therefore the emergency operating procedures do not require the operators to control the containment pressure to the pressure values on this stepped line or any other curve on this figure.

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78 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-22).

79 Entergy letter (BVY 05-017) to NRC dated February 24, 2005, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 23, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-39).

80 Entergy letter (BVY 04-106) to NRC dated October 5, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 18, Extended Power Uprate - ECCS Pump Net Positive Suction Head Margin" (Calculation VYC 0808 Revision 8, Page 52 of 58, Figure 4-2).

The licensee states that VYNPS procedure ON 3164, "ECCS Suction Strainer Plugging," contains operator guidance on indications of pump cavitation and possible responses.[81] The indications of inadequate available NPSH include:

1. Pump motor current erratic or decreasing;

2. Decreasing pump suction pressure (read locally) with steady state torus pressure/temperature conditions;

3. System flow rate erratic or less than expected for the backpressure to which the system is discharging;

4. Frequent adjustments of ECCS system discharge valve to maintain a constant flow rate at steady state backpressure/level conditions;

5. Audible indications of pump cavitation, such as increasing vibration/rough operation; and

6. Possible opening or cycling of minimum flow valves in response to flow decrease caused by suction strainer plugging.

The possible operator responses include:

1. Remove from service or throttle flow from those ECCS systems not needed to restore and maintain emergency operating procedure parameters. Consider securing one of the two operating RHR pumps within a single loop;

2. If possible, realign the suction of the core spray pump(s) to the condensate storage tank. The condensate storage tank is a nonsafety-related source of water. The core spray flow may also be reduced to maintain adequate available NPSH;

3. If an ECCS pump is aligned to the condensate storage tank, refill the tank; and/or

4. Consider aligning the service water system or fire protection system to the "A" RHR loop. A procedure for this exists.

The NRC staff considers the actions for identifying and mitigating loss of available NPSH to be acceptable since (1) they are contained in written procedures, (2) there are multiple possible indications and possible mitigating actions, and (3) ECCS and suppression pool cooling

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81 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-23).

functions, and hence, the proper functioning of the ECCS and suppression pool cooling pumps, would always be a priority in terms of the operators' attention.

Containment Integrity

Credit for containment accident pressure requires containment integrity. Design basis analyses, such as those supporting the extended power uprate, assume containment integrity. This assumption is justified by the stringent requirements in 10 CFR Part 50 and the VYNPS TSs. 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J require containment leakage rate testing of the containment structure, penetrations and isolation valves at the maximum predicted LOCA pressure. 10 CFR 50.55a(ii)B requires periodic inservice examination of the containment structure in accordance with the ASME Code.

VYNPS containment integrity is continuously monitored during normal operation since the containment is inerted with nitrogen gas. Any significant increase in the amount of nitrogen that must be supplied to the containment might be a sign of degradation in containment integrity and would be observed by the reactor operators. The operators would then take the appropriate action in accordance with the plant's operating procedures. Another sign of loss of integrity would be the presence of oxygen gas in containment. Monitors provide continuous assurance that the oxygen concentration in containment is less than the TS limit. Again, if a greater concentration of oxygen were detected, the operators would take the appropriate action in accordance with the abnormal operating procedures. Thus, there is reasonable assurance the containment will not leak at a rate greater than the limit specified in the TSs during accident conditions and accounted for in the VYNPS NPSH analyses.

Furthermore, using the same analytical methods, the conservatively predicted peak containment pressure at the current power level is 41.6 psig and the predicted peak containment pressure at EPU conditions is 41.8 psig. Thus, the increase in peak containment pressure, and therefore the increased challenge to containment integrity due to the EPU, is minimal.

Conclusion

The NRC staff has reviewed the containment heat removal systems assessment provided by the licensee and concludes that the licensee has adequately addressed the effects of the proposed EPU. The NRC staff finds that the systems will continue to meet draft GDC-41 and 52 with respect to limiting the containment pressure and temperature following a LOCA and maintaining them at acceptably low levels. In addition, the staff finds the proposed VYNPS licensing basis change to credit containment accident pressure to be acceptable since the credited pressure remains below the containment accident pressure conservatively calculated by the licensee to be available. Therefore, the NRC staff finds the proposed EPU acceptable with respect to containment heat removal systems.

2.6.6 Secondary Containment Functional Design

Regulatory Evaluation

The secondary containment structure and supporting systems of dual containment plants are provided to collect and process radioactive material that may leak from the primary containment following an accident. The supporting systems maintain a negative pressure within the secondary containment and process this leakage. The NRC staff's review covered (1) analyses of the pressure and temperature response of the secondary containment following accidents within the primary and secondary containments; (2) analyses of the effects of openings in the secondary containment on the capability of the depressurization and filtration system to establish a negative pressure in a prescribed time; (3) analyses of any primary containment leakage paths that bypass the secondary containment; (4) analyses of the pressure response of the secondary containment resulting from inadvertent depressurization of the primary containment when there is vacuum relief from the secondary containment; and (5) the acceptability of the mass and energy release data used in the analysis. The NRC staff's review primarily focused on the effects that the proposed EPU may have on the pressure and temperature response and drawdown time of the secondary containment, and the impact this may have on offsite dose. The NRC's acceptance criteria for secondary containment functional design are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of a loss-of-coolant accident; and (2) draft GDC-10, insofar as it requires that reactor containment be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain functional capability for as long as the situation requires. Specific review criteria are contained in SRP Section 6.2.3.

Technical Evaluation

An increase in rated thermal power increases the heat load on the secondary containment and affects the drawdown time of the secondary containment. The drawdown time is the time period following the start of the accident during which loss of offsite power causes loss of secondary containment vacuum (relative to atmospheric pressure) which is assumed to result in releases from the primary containment directly to the environment without filtering. The licensee addressed these issues as part of a separate license amendment request to incorporate a full-scope application of an alternative source term (AST) methodology in accordance with 10 CFR 50.67. In that license amendment request, the licensee's analyses, including the secondary containment drawdown time, were performed assuming EPU conditions. The licensee used the GOTHIC containment computer code to calculate the drawdown time. The NRC staff approved the VYNPS AST amendment request on March 29, 2005.[82]

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82 Letter from Richard B. Ennis (NRC) to Michael Kansler (Entergy), dated March 29, 2005, VYNPS Amendment No. 223, "Alternative Source Term (TAC No. MC0253)."

Conclusion

The NRC staff has reviewed the licensee's assessment related to the secondary containment pressure and temperature transient and the ability of the secondary containment to provide an essentially leak-tight barrier against uncontrolled release of radioactivity to the environment. The NRC staff concludes that the licensee has adequately accounted for the increase of mass and energy that would result from the proposed EPU and further concludes that the secondary containment and associated systems will continue to provide an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment following implementation of the proposed EPU. Based on this, the NRC staff also concludes that the secondary containment and associated systems will continue to meet the requirements of draft GDC-10, 40, and 42. Therefore, the NRC staff finds the proposed EPU acceptable with respect to secondary containment functional design.

2.6.7 Additional Review Areas - Containment Review Considerations

Hardened Vent

Regulatory Evaluation

Generic Letter (GL) 89-16[83] discussed the advantages of installing a hardened containment (torus) vent and requested information from licensees on installation of such a vent. This was a result of the NRC BWR Mark I Containment Performance Improvement Program. This is a beyond design basis issue. The licensee installed such a vent on VYNPS.

Technical Evaluation

The hardened vent design criterion is to maintain containment design pressure with the reactor at 1% of rated thermal power. The licensee stated[84] that the actual capability of the VYNPS design was determined to be 1.3% of the CLTP of 1593 MWt (or a margin of 30% above the design criterion). Therefore, a 20% increase in the rated thermal power remains within the capability of the VYNPS hardened vent system.

Conclusion

Since adequate margin remains in the VYNPS hardened vent design after the EPU, the hardened vent is acceptable for EPU conditions.

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83 NRC Generic Letter 89-16, Installation of Hardened Wetwell Vent, September 1, 1989.

84 Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information" (Response to RAI SPSB-C-18).

Containment Isolation

Regulatory Evaluation

The NRC's acceptance criteria for containment isolation are based on draft GDC 53 insofar as the containment isolation function must be protected by redundant valving and associated apparatus.

Technical Evaluation

An important aspect of the effect of containment accident conditions on containment isolation is addressed by NRC GL 96-06[85]. GL 96-06 addressed three subjects. These are: (1) possible water hammer effects in containment air coolers during a LOCA or main steam line break, (2) two phase flow in containment air coolers adversely affecting heat removal assumptions, and (3) thermally induced overpressurization of isolated water filled piping sections in containment which could jeopardize the ability of accident mitigating systems to perform their safety functions and could also lead to a breach of containment integrity via bypass leakage. Only the third item is addressed in this section of the SE. Evaluation of items (1) and (2) is addressed in SE Section 2.5.3.3.

Section 4.1.6 of the PUSAR states that:

> the VYNPS response to GL 96-06 was accomplished in part using the limiting drywell temperature, pressure and steam mass fraction time histories for CLTP [current licensed thermal power] conditions. The results of the containment analysis presented within this section are bounded by the CLTP conditions assumed for the analysis of affected in-containment piping. Therefore, the existing VYNPS response to GL 96-06 remains valid for CPPU.

Since the CLTP analysis remains bounding for EPU conditions, the NRC staff finds the proposed CPPU acceptable with respect to thermally-induced pressurization of containment penetrations.

Conclusion

The NRC staff has reviewed the licensee's assessment related to GL 96-06 and concludes that the licensee has adequately addressed the issue of thermally-induced overpressurization of the affected piping in containment under EPU conditions.

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85 Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions, NRC Generic Letter 96-06, US NRC, September 30, 1996.

2.7 Habitability, Filtration, and Ventilation

2.7.1 Control Room Habitability System

Regulatory Evaluation

The NRC staff reviewed the control room habitability system and control building layout and structures to ensure that plant operators are adequately protected from the effects of accidental releases of toxic and radioactive gases. A further objective of the NRC staff's review was to ensure that the control room can be maintained as the backup center from which technical support center personnel can safely operate in the case of an accident. The NRC staff's review focused on the effects of the proposed EPU on radiation doses, toxic gas concentrations, and estimates of dispersion of airborne contamination. The NRC's acceptance criteria for the control room habitability system are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for engineered safety features (ESFs) against the dynamic effects that might result from plant equipment failures, as well as the effects of a LOCA; and (2) draft GDC-11 and 10 CFR 50.67, insofar as they require that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem Total Effective Dose Equivalent (TEDE) for the duration of the accident. Specific review criteria are contained in SRP Section 6.4 and other guidance provided in Matrix 7 of RS-001.

Technical Evaluation

For control room habitability, the NRC staff reviewed the control room ventilation system and control building layout and structures, as described in the VYNPS UFSAR and the analysis provided by the licensee in support of VYNPS Amendment No. 223, dated March 29, 2005 (Reference 57), which incorporated a full-scope application of an AST methodology in accordance with 10 CFR 50.67. In support of the AST amendment, the licensee re-analyzed the following DBAs: LOCA, main steam line break accident, fuel-handling accident, and control rod drop accident. The licensee performed the AST radiological analyses assuming a reactor power equal to 1950 MWt (i.e., 102% of the proposed EPU power level of 1912 MWt). As discussed in PUSAR Section 4.4, and summarized in PUSAR Table 4-4, the results of these analyses demonstrate that the EPU dose to control room occupants will be less than the 30-day 5 rem TEDE dose for the limiting DBA LOCA. As discussed in the NRC staff's SE for Amendment No. 223, the staff found, with reasonable assurance, that the licensee's estimates of control room doses due to postulated DBAs will comply with the guidance in 10 CFR 50.67. Based on the power levels used in the AST analyses, the NRC staff concludes that the AST analysis is bounding for the proposed EPU and, therefore, is acceptable with respect to radioactive gases. The NRC staff did not identify any aspects of the proposed EPU that would affect control room habitability with respect to toxic gases (e.g., no new system operation or creation of additional chemical sources).

Conclusion

The NRC staff has reviewed the effects of the proposed EPU on the ability of the control room habitability system to protect plant operators against the effects of accidental releases of toxic and radioactive gases. The NRC staff concludes that the control room habitability system will continue to provide the required protection following implementation of the proposed EPU. Based on this, the NRC staff concludes that the control room habitability system will continue to meet the requirements of draft GDC-11, 40, and 42, and 10 CFR 50.67. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the control room habitability system.

2.7.2 Engineered Safety Feature Atmosphere Cleanup

Regulatory Evaluation

ESF atmosphere cleanup systems are designed for fission product removal in post-accident environments. These systems generally include primary systems (e.g., in-containment recirculation) and secondary systems (e.g., standby gas treatment systems and emergency or post-accident air-cleaning systems) for the fuel-handling building, control room, shield building, and areas containing ESF components. For each ESF atmosphere cleanup system, the NRC staff's review focused on the effects of the proposed EPU on system functional design, environmental design, and provisions to preclude temperatures in the adsorber section from exceeding design limits. The NRC's acceptance criteria for ESF atmosphere cleanup systems are based on (1) draft GDC-11 and 10 CFR 50.67, insofar as they require that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident; (2) draft GDC-67, 68, and 69, insofar as they require that systems that may contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions; and (4) draft GDC-17, insofar as it requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences (AOOs), and postulated accidents. Specific review criteria are contained in SRP Section 6.5.1.

Technical Evaluation

The function of the ESF atmosphere cleanup system is to mitigate the consequences of postulated accidents by removing from the atmosphere radioactive material that may be released in the event of an accident. ESF atmosphere cleanup systems should be designed so that they can operate after a design-basis accident (DBA) and can retain radioactive material after a DBA. The system should have provisions to prefilter air, remove moisture and meet the guidance in RG 1.52 for charcoal adsorption.

The ESF atmosphere cleanup system at VYNPS is the standby gas treatment system (SGTS). As discussed in Section 4.5 of the PUSAR, the acceptability of the SGTS at VYNPS was

determined by reviewing plant-specific data at EPU conditions against the criteria stated in Section 4.5 of the PUSAR. With respect to heat loads due to EPU and the basis for determination of system acceptability post-EPU, the SGTS is acceptable for EPU conditions if the SGTS inlet temperature is below [[ ]]. The licensee determined that the secondary containment in both normal and accident conditions was confirmed to be below [[ ]], and therefore, the SGTS is acceptable for EPU.

In addition, in accordance with the PUSAR, the SGTS is designed to maintain secondary containment at a negative pressure and to filter the exhaust air for removal of fission products potentially present during abnormal conditions. By limiting the release of airborne particulate and halogens, the SGTS limits off-site dose following a postulated DBA. The design flow capacity of the SGTS was selected to maintain the secondary containment at the required negative pressure to minimize the potential for exfiltration of air from the reactor building. The PUSAR states that this capability is unaffected by EPU because the [[

]] The PUSAR also states that the results of the AST evaluation, applicable to VYNPS, show that the maximum charcoal loading, based on only [[

]] and that this is well below the 2.5 mg/gm maximum value in RG 1.52. The staff finds this acceptable. The maximum component temperature is approximately [[ ]] with normal flow conditions [[ ]] conditions of a failed fan with minimum cooling flow, which is well below the [[ ]] charcoal ignition temperature. The NRC staff finds this acceptable.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ESF atmosphere cleanup systems. The NRC staff concludes that the licensee has adequately accounted for the increase of fission products and changes in expected environmental conditions that would result from the proposed EPU, and the NRC staff further concludes that the ESF atmosphere cleanup systems will continue to provide adequate fission product removal in post-accident environments following implementation of the proposed EPU. Based on this, the NRC staff concludes that the ESF atmosphere cleanup systems will continue to meet the requirements of draft GDC-11, 17, 67, 68, and 69; and 10 CFR 50.67. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ESF atmosphere cleanup systems.

2.7.3 Control Room Area Ventilation System

Regulatory Evaluation

The function of the control room area ventilation system (CRAVS) is to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components during normal operation, AOOs, and DBA conditions. The NRC's review of the CRAVS focused on the effects that the proposed EPU will have on the functional performance of safety-related portions of the system. The review included the effects of radiation, combustion, and other toxic products; and the expected environmental conditions in areas served by the CRAVS. The NRC's acceptance criteria for the CRAVS are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of a loss of coolant accident; (2) draft GDC-11 and 10 CFR 50.67, insofar as they require that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident; and (3) draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 9.4.1.

Technical Evaluation

As indicated above, the function of the CRAVS is to provide a controlled environment for the comfort and safety of control room personnel and to assure the operability of control room components during normal operation, AOOs, and DBA conditions.

The NRC staff reviews the CRAVS from the air intake to the point of discharge where the system connects to the gaseous cleanup and treatment system or station vents to assure conformance with the requirements of draft GDCs 11, 40, 42, and 70, and 10 CFR 50.67. The review includes components such as air intakes, ducts, air conditioning units, filters, blowers, isolation dampers or valves, and exhaust fans. The review of the CRAVS covers the control room, switchgear and battery room, access control area, control building heating, ventilating, and air conditioning (HVAC) equipment room, and computer room.

In a letter dated July 30, 2004 (Reference 11), the licensee stated that "the heat loads in the control room are not a function of power level. Heat sources in the control room are from electrical equipment, ambient outside air temperature, and personnel. None of these sources are expected to increase at CPPU conditions. Therefore, the control room HVAC system's ability to provide appropriate temperature and humidity conditions for personnel and equipment during all modes of operation and emergency conditions is not impacted by CPPU. In addition, CPPU has no impact on the control room HVAC system's ability to provide for heating during cold weather conditions." The NRC staff finds this acceptable.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ability of the CRAVS to provide a controlled environment for the comfort and safety of control room personnel and to support the operability of control room components. The NRC staff also evaluated the effects of the proposed EPU with respect to the control room environment and the accidental releases of toxic and radioactive gases as discussed in SE Section 2.7.1. The NRC staff concludes that the CRAVS will continue to provide an acceptable control room environment for safe operation of the plant following implementation of the proposed EPU. The NRC staff also concludes that the system will continue to suitably control the release of gaseous radioactive effluents to the environment. Based on this, the NRC staff concludes that the CRAVS will continue to meet the requirements of draft GDC-11, 40, 42, and 70, and 10 CFR 50.67. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the CRAVS.

2.7.4 Spent Fuel Pool Area Ventilation System

Regulatory Evaluation

The function of the spent fuel pool area ventilation system (SFPAVS) is to maintain ventilation in the spent fuel pool equipment areas, permit personnel access, and control airborne radioactivity in the area during normal operation, AOOs, and following postulated fuel handling accidents. The NRC staff's review focused on the effects of the proposed EPU on the functional performance of the safety-related portions of the system. The NRC's acceptance criteria for the SFPAVS are based on (1) draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents, and (2) draft GDC-67, 68, and 69, insofar as they require that systems which contain radioactivity be designed with appropriate confinement and containment. Specific review criteria are contained in SRP Section 9.4.2.

Technical Evaluation

The function of the SFPAVS is to maintain ventilation in the spent fuel pool (SFP) equipment area, to permit personnel access, and to control airborne radioactivity in the area during normal operation, AOOs, and following postulated fuel-handling accidents.

The NRC staff reviews the SFPAVS from the air intake to the point of discharge where the system connects to the gaseous cleanup and treatment system or the station vents to assure conformance with the requirements of draft GDCs 67, 68, 69, and 70. The review includes components such as air intakes, ducts, air conditioning units, filters, blowers, isolation dampers, and exhaust fans. The review of the SFPAVS covers all areas containing or adjacent to the SFP, including the spent fuel cooling pump room.

In the letter dated July 30, 2004 (Reference 11), the licensee stated that VYNPS does not have a separate SFP area ventilation system and that the SFP area is served by the reactor building HVAC system. The licensee also stated that the fuel pool cooling and demineralizer system (FPCDS) was evaluated for the EPU for both batch and full core off-loads. For normal operation, it was determined that although the decay heat load would increase for the EPU, the SFP temperature would remain within current limits. Consequently, the licensee concluded that there is no impact on the heat load to the reactor building HVAC during normal operation. The NRC staff finds this acceptable.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the SFPAVS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system's capability to maintain ventilation in the SFP equipment areas, permit personnel access, control airborne radioactivity in the area, control release of gaseous radioactive effluents to the environment, and provide appropriate containment. Based on this, the NRC staff concludes that the SFPAVS will continue to meet the requirements of draft GDC-67, 68, 69, and 70. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the SFPAVS.

2.7.5 Auxiliary and Radwaste Area and Turbine Areas Ventilation Systems

Regulatory Evaluation

The function of the auxiliary and radwaste area ventilation system (ARAVS) and the turbine area ventilation system (TAVS) is to maintain ventilation in the auxiliary and radwaste equipment and turbine areas, permit personnel access, and control the concentration of airborne radioactive material in these areas during normal operation, during AOOs, and after postulated accidents. The NRC staff's review focused on the effects of the proposed EPU on the functional performance of the safety-related portions of these systems. The NRC's acceptance criteria for the ARAVS and TAVS are based on draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Sections 9.4.3 and 9.4.4.

Technical Evaluation

The NRC staff reviews the ARAVS from air intake to the point of discharge where the system connects to the gaseous cleanup and treatment system or station vents to assure conformance with the requirements of draft GDC 70. The review includes components such as air intakes, ducts, air conditioning units, blowers, isolation dampers, and roof exhaust fans. The review of the ARAVS covers the radwaste areas and controlled access nonradioactive areas and their relationship to safety-related areas in the auxiliary building.

The licensee stated in Reference 11 that the heat loads in the radwaste building are not a function of power level. Therefore, the radwaste building HVAC system is acceptable for EPU operation, and its ability to vent potentially contaminated air is not affected by EPU.

The licensee also stated that the offgas building ventilation system maintains a suitable environment for operation personnel and equipment as required to ensure proper operation of the equipment. The EPU evaluation noted that while hydrogen production is linear with respect to core thermal power, the operating temperatures of the recombiner, following EPU, will remain at or below the design-basis temperature of 655°F. An evaluation of the operating temperature of the recombiner room indicates an increase of 3°F or less at EPU conditions, which is within the capabilities of the offgas ventilation system. In addition, the licensee stated that the radwaste building HVAC and offgas ventilation are not credited during post-accident conditions.

The NRC staff reviews the TAVS from air intake to the point of discharge to assure conformance with the requirements of draft GDC 70. The review includes components such as air intakes, ducts, cooling units, blowers, isolation dampers, and roof exhaust fans. The review of the TAVS includes systems contained in the turbine building and their relationship, if any, to safety-related equipment areas.

With respect to the TAVS, the licensee stated in Reference 11 that increases in area heat gain and ambient air temperature, as a result of the EPU, are predominantly caused by increases in operating temperature of piping systems, equipment, and air-cooled motors operating under increased loads. For the EPU, it was determined that the following areas serviced by the turbine building HVAC would experience temperature increases as indicated.

Area	EPU Ambient Temperature Increase (°F)
Low Pressure Heater area	4.1
High Pressure Heater Area	1.7
Feedwater Pump Room	7.6
Condensate Pump Room	3.5

It was noted in Section 6.6 of the PUSAR that the 105°F design ambient room temperature may be exceeded for the condensate pump and feedwater pump rooms during the summer under EPU conditions. This aspect of the EPU was subsequently evaluated by the licensee as discussed in Reference 8. The licensee's evaluation determined that the affected equipment was acceptable for operation at the temperatures expected under EPU conditions. The licensee also stated that the turbine building HVAC is not credited during post-accident conditions. Based on the above, the NRC staff finds that the licensee has adequately addressed the effects of the proposed EPU on the ARAVS and the TAVS.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ARAVS and TAVS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the capability of these systems to maintain ventilation in the auxiliary and radwaste equipment areas and in the turbine area, permit personnel access, control the concentration of airborne radioactive material in these areas, and control release of gaseous radioactive effluents to the environment. Based on this, the NRC staff concludes that the ARAVS and TAVS will continue to meet the requirements of draft GDC-70. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ARAVS and the TAVS.

2.7.6 Engineered Safety Feature Ventilation System

Regulatory Evaluation

The function of the engineered safety feature ventilation system (ESFVS) is to provide a suitable and controlled environment for ESF components following certain anticipated transients and DBAs. The NRC staff's review for the ESFVS focused on the effects of the proposed EPU on the functional performance of the safety-related portions of the system. The NRC staff's review also covered (1) the ability of the ESF equipment in the areas being serviced by the ventilation system to function under degraded ESFVS performance; (2) the capability of the ESFVS to circulate sufficient air to prevent accumulation of flammable or explosive gas or fuel-vapor mixtures from components (e.g., storage batteries and stored fuel); and (3) the capability of the ESFVS to control airborne particulate material (dust) accumulation. The NRC's acceptance criteria for the ESFVS are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a LOCA; (2) draft GDC-24 and 39, insofar as they require onsite and offsite electric power systems be provided to permit functioning of the ESFs and protection systems; and (3) draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 9.4.5.

Technical Evaluation

As stated above, the function of the ESFVS is to provide a suitable and controlled environment for ESF components following certain anticipated transients and DBAs.

The NRC staff reviews the ESFVS from air intake to the point of discharge to the atmosphere to assure conformance with the requirements of draft GDCs 24, 39, 40, 42, and 70. The review includes components such as air intakes, ducts, air conditioning units, flow control devices, isolation dampers, exhaust vents, and exhaust fans.

The review of the ESFVS covers all ventilation systems utilized to maintain a controlled environment in areas containing safety-related equipment. These include the diesel generator area, emergency core cooling system (ECCS) pump rooms, and other areas containing equipment essential for safe shutdown of the reactor or necessary to prevent or mitigate the consequences of an accident.

The licensee stated that the ECCS corner rooms are cooled by reactor recirculation units (RRU)-5, RRU-6, RRU-7, and RRU-8, in addition to outside air provided by reactor building HVAC. At EPU conditions, normal heat loads and ambient temperatures do not increase. Therefore, the ability of RRU-5, RRU-6, RRU-7, and RRU-8 to maintain acceptable area temperatures during normal operation is unchanged. In addition, the licensee stated that there is no change in the environments controlled by the diesel generator room HVAC. The NRC staff finds this acceptable.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on the ESFVS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the ability of the ESFVS to provide a suitable and controlled environment for ESF components. The NRC staff further concludes that the ESFVS will continue to assure a suitable environment for the ESF components following implementation of the proposed EPU. The NRC staff also concludes that the ESFVS will continue to suitably control the release of gaseous radioactive effluents to the environment following implementation of the proposed EPU. Based on this, the NRC staff concludes that the ESFVS will continue to meet the requirements of draft GDC-24, 39, 40, 42, and 70. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the ESFVS.

2.8 Reactor Systems

The licensee's application for the proposed EPU was prepared following the guidelines contained in GE Licensing Topical Report (LTR) NEDC-33004P-A, "Constant Pressure Power Uprate," Revision 4, dated July 31, 2003 (Reference 51). The constant pressure power uprate (CPPU) LTR (CLTR) was approved by the NRC in an SE dated March 31, 2003 (Reference 52).

Attachment 4 to Reference 1 contains GE Report NEDC-33090P (proprietary) which is the Power Uprate Safety Analysis Report (PUSAR) for VYNPS. This report summarizes the results of the safety analyses and evaluations performed by GE specifically to justify the proposed EPU for VYNPS. The report follows the generic content and format using the CPPU approach to uprating reactor power, as described in the CLTR. A non-proprietary (i.e., publicly available) version) of the PUSAR is contained in Attachment 6 to Reference 1. The PUSAR describes in general, the plant's ability to operate at the higher power level and to respond to anticipated operational occurrences, transients and accident conditions as designed and analyzed. The PUSAR also summarizes GE's evaluation of the effect of the increased thermal power level on

the capability and performance of systems, structures, and components important to safe operation of the plant.

Section 11.2 of the NRC staff's SE for the CLTR, titled, "Application of the CPPU LTR," states that each of the sections of the CPPU LTR were in one of two disposition categories: generic assessment or plant-specific evaluation. The NRC staff's safety conclusions with regard to reactor core-related technical areas for VYNPS EPU operation are based on either the generic assessment or the plant-specific evaluation.

In general, the licensee's plant-specific engineering evaluations supporting the EPU were performed in accordance with guidance contained in the NRC-approved GE LTR NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate (ELTR1)" (Reference 63). For some items, bounding analyses and evaluations provided in NRC-approved GE LTR NEDC-32523P-A, "Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate (ELTR2)," (Reference 64) were cited. The ELTR2 generic evaluations assume (a) a 20% increase in the thermal power, (b) an increase in operating dome pressure up to 1095 psia, (c) a reactor coolant temperature increase to 556°F, and (d) a steam and feedwater flow increase of about 24%.

The approach to achieving the EPU consists of (1) an increase in the core thermal power with a more uniform power distribution achieved by better fuel management techniques to create increased steam flow, (2) a corresponding increase in the feedwater system flow, (3) no increase in maximum core flow, and (4) reactor operation primarily along the maximum extended load line limit analysis (MELLLA) rod/flow lines. This approach is based on, and is consistent with, the NRC-approved BWR EPU guidelines that are given in the CLTR.

The current MELLLA power/flow map was approved in VYNPS Amendment No. 219, "Average Power Range Monitor, Rod Block Monitor TSs/Maximum Extended Load Line Limit Analysis" (ARTS/MELLLA), dated April 14, 2004 (Reference 53).

The NRC staff's review of the VYNPS EPU amendment request used applicable rules, RGs, SRP sections, and NRC staff positions on the topics being evaluated. The staff also used RS-001, Revision 0, "Review Standard For Extended Power Uprate," December 2003. Additionally, the NRC staff evaluated the EPU application for conformance with the CPPU BWR EPU program as defined in the staff's SE for the CLTR (Reference 52). The CLTR provides appropriate guidelines for constant pressure EPU applications with a core exclusively using GE fuel.

The scope of the NRC staff's review for the VYNPS EPU request included "lessons learned" from past power uprate amendment reviews. In reviewing the licensee's request for EPU, the staff considered the recommendations of the report of the Maine Yankee Lessons Learned Task Group (SECY-97-042, "Response to OIG Event Inquiry 96-04S Regarding Maine Yankee," February 18, 1997). The task group's main findings centered on the use and applicability of the computer codes and analytical methodologies used for power uprate

evaluations. The staff requested that the licensee identify all codes and methodologies used to obtain safety limits and operating limits and explain how it verified that these limits were correct for the uprated core. The licensee was also requested to identify and discuss any limitations imposed by the staff on the use of these codes and methodologies. During the review, the NRC staff evaluated several areas related to application of GE methods used for EPU evaluations. The evaluation of the GE methods is shown in SE Section 2.8.7.

The VYNPS EPU reload core for Cycle 25 (the cycle following the fall 2005 refueling outage) will consist of all Global Nuclear Fuel (GNF) GE-14 (10x10) fuel. The EPU safety analyses and the cycle-specific reload analyses will be performed in accordance with NRC-approved GE analytical methodologies described in the latest version of GESTAR II (Reference 68). The LTR specifying the codes and methodologies used for performing the safety analyses are documented in VYNPS TS Section 6.6.C. The limiting anticipated operational occurrence (AOO) and accident analyses are reanalyzed or confirmed to be valid for every reload, and the safety analyses of transients and accidents are documented in Chapter 14 of the VYNPS UFSAR (Reference 50). Limiting transient or accident analyses are generally defined as analyses of events that could potentially affect the core operating and safety limits that ensure the safe operation of the plant.

2.8.1 Fuel System Design

Regulatory Evaluation

The fuel system consists of arrays of fuel rods, burnable poison rods, spacer grids and springs, end plates, channel boxes, and reactivity control rods. The NRC staff reviewed the fuel system to ensure that (1) the fuel system is not damaged as a result of normal operation and AOOs, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. The NRC staff's review covered fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, AOOs, and postulated accidents. The NRC's acceptance criteria are based on (1) 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits; and (3) draft GDC-37, 41, and 44, insofar as they require that a system to provide abundant emergency core cooling be provided to prevent fuel damage following a LOCA. Specific review criteria are contained in SRP Section 4.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The fuel system design at VYNPS is described in Section 3.2 of the VYNPS UFSAR. The core thermal-hydraulic design and fuel performance characteristics are evaluated for each reload

fuel cycle. The following sections address the effect of the EPU on fuel design performance and thermal limits.

Fuel Design and Operation

The PUSAR states that a CPPU increases the average power density proportional to the power increase and has some effects on operating flexibility, reactivity characteristics and energy requirements. The peak bundle power will increase from 7.02 MWt before the EPU to 7.37 MWt after the EPU. The power distribution in the core is changed to achieve increased core power, while limiting the Minimum Critical Power Ratio (MCPR), Linear Heat Generation Rate (LHGR), and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) in any individual fuel bundle to be within its operating limits as defined in the core operating limits report (COLR).

As discussed in the NRC's SE for the CLTR (Reference 52), licensees using GE fuel, up through GE-14 fuel, may reference the CLTR as the basis for their EPU. As of RFO 25 (fall 2005), the VYNPS core utilizes GE-14 fuel only (Reference 33). Section 2.1 of the PUSAR states that [[

]] The fuel design limits are established for all new fuel product line designs as a part of the fuel introduction and reload analyses. [[

]]

The PUSAR further states that the percent power level above which fuel thermal margin monitoring is required may change with a CPPU. The original plant operating licenses set this monitoring threshold at a typical value of 25% of Rated Thermal Power (RTP). [[

]]

For a CPPU, the fuel thermal margin monitoring threshold is scaled down, if necessary, to ensure that monitoring is initiated [[

]], then the existing power threshold value must be lowered by a factor of $1.2/P_{25}$. The licensee stated that for VYNPS, the CPPU fuel thermal monitoring threshold is established at 23% of CPPU RTP. A change in the fuel thermal monitoring threshold also requires a corresponding change to the TS reactor core safety limit for reduced pressure or low core flow.

Because the licensee will continue to use approved analytical methods, and will continue to ensure that the results of those analyses remain within currently acceptable limits, the NRC staff finds the proposed EPU acceptable with respect to fuel design and operation.

Thermal Limits Assessment

The NRC's acceptance criteria require that the reactor core and the associated control and instrumentation systems be designed with appropriate margin to ensure that fuel design limits are not exceeded during normal operation, including AOOs. Operating limits are established to assure that regulatory or safety limits are not exceeded for a range of postulated events (transients and accidents).

The safety limit minimum critical power ratio (SLMCPR) ensures that 99.9% of the fuel rods are protected from boiling transition during normal operation and AOOs. The operating limit minimum critical power ratio (OLMCPR) assures that the SLMCPR will not be exceeded as the result of an AOO. NRC staff experience with several power uprates has shown that the change in OLMCPR resulting from a constant-pressure EPU is small. The CLTR SE (Reference 52) stated that this [[ ]] When the core design is complete, the OLMCPR will be determined with the "real" core design parameters. Because the licensee will use approved methods to evaluate these parameters, this is acceptable to the staff. As required by the CLTR SE, the licensee will perform [[ ]] to demonstrate that the SLMCPR and OLMCPR are appropriate for establishing the CPPU thermal limits.

The maximum average planar linear heat generation rate (MAPLHGR) operating limit is based on the most limiting LOCA conditions, and ensures compliance with the ECCS acceptance criteria in 10 CFR 50.46. For every new fuel type, GE performs LOCA analyses to confirm compliance with the LOCA acceptance criteria, and for every reload licensees confirm that the MAPLHGR operating limit for each reload fuel bundle design remains applicable.

In general, the licensee must ensure that plant operation is in compliance with the cycle-specific thermal limits (SLMCPR, OLMCPR, MAPLHGR, and maximum LHGR) and specify the thermal limits in a cycle-specific COLR as required by VYNPS TSs. In addition, while EPU operation may result in an increase in fuel burnup, the licensee cannot exceed the NRC-approved burnup limits. In accordance with VYNPS TS Section 6.6.C, cycle-specific analyses are performed using NRC-reviewed and-approved methodologies. The NRC staff finds that the licensee has appropriately considered the potential effects of EPU operation on the fuel design limits, and the generic thermal limits assessment show that the VYNPS can operate within the fuel design limits during steady state operation, AOOs, and accident conditions.

The TS 1.1.A requirements for SLMCPR assure that the fuel system is not damaged as a result of normal operation and AOOs. Compliance with 10 CFR 50.46, as discussed in SE

Section 2.8.5.6.2, assures that the fuel system damage will never be so severe as to prevent control rod insertion, and that core coolability is maintained.

Conclusion

The NRC staff has reviewed the licensee's [[ ]] assessment and concludes that it is consistent with the staff's understanding described in the CLTR SE. In addition, the licensee will perform plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the NRC staff concludes that the fuel system and associated analyses will continue to meet the requirements of 10 CFR 50.46, draft GDC-6, 37, 41, and 44 following implementation of the proposed EPU. Therefore, the staff finds the proposed EPU acceptable with respect to the fuel system design.

2.8.2 Nuclear Design

Regulatory Evaluation

The NRC staff reviewed the nuclear design of the fuel assemblies, control systems, and reactor core to ensure that fuel design limits will not be exceeded during normal operation and anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. The NRC staff's review covered core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality, burnup, and vessel irradiation. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits during any condition of normal operation, including the effects of AOOs; (2) draft GDC-8, insofar as it requires that the reactor core be designed so that the overall power coefficient in the power operating range shall not be positive; (3) draft GDC-7, insofar as it requires that the reactor core be designed to ensure that power oscillations, which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed; (4) draft GDC-12, insofar as it requires that instrumentation and controls be provided as required to monitor and maintain variables within prescribed operating ranges; (5) draft GDC-14 and 15, insofar as they require that the protection system be designed to initiate the reactivity control systems automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and to initiate operation of ESFs under accident situations; (6) draft GDC-31, insofar as it requires that the reactivity control systems be capable of sustaining any single malfunction without causing a reactivity transient which could result in exceeding acceptable fuel damage limits; (7) draft GDC-27 and 28, insofar as they require that at least two independent reactivity control systems be provided, with both systems capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits; (8) draft GDC-29, insofar as it requires that at least one of the reactivity control systems be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits; and (9) draft GDC-32, insofar as it requires

that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling. Specific review criteria are contained in SRP Section 4.3 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The VYNPS nuclear design is described in Section 3.6 of the UFSAR. As required by the CLTR SE, the topics addressed by the licensee in this evaluation are hot excess reactivity and shutdown margin.

The higher core energy requirements of a power uprate may affect the hot excess core reactivity and can also affect operating shutdown margins. The general effect of a power uprate on core reactivity, as described in Section 5.7.1 of ELTR1 (Reference 63), is also applicable to a CPPU. Based on experience with previous plant-specific power uprate submittals, the required hot excess reactivity and shutdown margin can typically be achieved for power uprates through the standard approved fuel and core reload design process. Plant shutdown and reactivity margins must meet NRC-approved limits established in GESTAR-II (Reference 68) on a cycle-specific basis and are evaluated for each plant reload core, and [[
]]

The VYNPS EPU reload core design will account for any loss of margin for future cycles. The reload core analysis will ensure that the minimum shutdown margin requirements are met for each core design and that the current design and TS cold shutdown margin will be met. Since the licensee will continue to confirm that the TS cold shutdown margin requirements will be met for each reload core operation, the NRC staff finds this acceptable, and concludes that the existing NRC acceptance criteria outlined in the Regulatory Evaluation section will continue to be satisfied.

As stated in the CLTR SE, the NRC staff agreed that [[

]] and that the licensee will evaluate the shutdown margin for the uprated reload core prior to CPPU implementation.

Conclusion

The NRC staff has reviewed the licensee's [[ ]] assessment and concludes that it is consistent with the staff's understanding described in the CLTR SE. In addition, the licensee will perform plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, and in coordination with the reviews of the fuel system design, thermal and hydraulic design, and transient and accident

analyses, the NRC staff concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable requirements of draft GDC-6, 7, 8, 12, 14, 15, 27, 28, 29, 31, and 32, and is acceptable to the staff.

2.8.3 Thermal and Hydraulic Design

Regulatory Evaluation

The NRC staff reviewed the thermal and hydraulic design of the core and the RCS to confirm that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to or a justified extrapolation from proven designs, (3) provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermal-hydraulic instability. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits during any condition of normal operation, including the effects of AOOs; and (2) draft GDC-7, insofar as it requires that the reactor core, together with reliable controls, ensure that power oscillations, which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed. Specific review criteria are contained in SRP Section 4.4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The thermal and hydraulic design for the VYNPS core is described in Section 3.7 of the UFSAR. As required by the CLTR SE, the evaluation for thermal hydraulic stability will be performed during the reload analysis. As stated in the CLTR SE, this is acceptable because the equilibrium cycle analysis is not necessary to demonstrate that the applicable stability solution will provide thermal-hydraulic stability protection for a CPPU, and the necessary analysis will be performed during the reload process.

A generic evaluation was performed for the interim corrective actions as documented in Section 3.2.1 of ELTR2. This generic evaluation is applicable for the CPPU. Interim corrective action stability boundaries are the same in terms of absolute core power and flow. The listed power levels, as a percentage of rated power, are scaled downwards based on the new uprated power.

The licensee stated that VYNPS has adopted the Option I-D solution. It was stated in the CLTR SE that for the plants with Option I-D solution, the exclusion region may change and SLMCPR protection may be affected by the CPPU. Option I-D is a solution combining prevention and detect-and-suppress elements. The prevention portion of the solution is an administratively controlled exclusion region. The detect-and-suppress feature is a demonstration that regional mode reactor instability is not probable and that the existing flow-biased flux trip provides adequate SLMCPR protection for events that initiate along the rated rod line. The CLTR SE requires that these features be analyzed for the first core reload analysis that incorporates the

new rated power level. The NRC staff has reviewed PUSAR Section 2.4, "Stability," and the licensee's responses to staff RAIs SRXB-A-13 through SRXB-A-16 in Reference 31 to address the issues of: 1) the dominance of the core-wide mode oscillation; 2) dependent parameters for the hot bundle oscillation magnitude portion of the detect-and suppress calculation with respect to the core and fuel design; 3) the impact of the EPU on relative stability of the plant as compared to pre-EPU conditions; and 4) a 10 CFR Part 21 notification from GE dated June 29, 2001, "Stability Reload Licensing Calculation using Generic DIVOM [Delta CPR over Initial CPR Versus Oscillation Magnitude]."

Based on its review, the NRC staff has found that the thermal-hydraulic stability analysis for the VYNPS EPU is acceptable because:

(1) a cycle-specific DIVOM curve or a conservative [[ ]] will be used to resolve the DIVOM 10 CFR Part 21 issue;

(2) cycle-specific ODYSY code calculations will be performed;

(3) the calculations show that the hot channel Decay Ratio (DR) is [[ ]], but the core-wide DR is [[ ]] Therefore, the oscillations are very likely to be core-wide and not out-of-phase;

(4) there is not any significant de-stabilizing trend in the EPU design by comparing Cycle 24 to Cycle 23;

(5) VYNPS has modified the flow-biased scram line to account for the reduced DIVOM performance. The APRM has to oscillate now by only 4.6% to scram.

The CLTR SE further stated that CPPU will also affect the SLMCPR protection confirmation. Changes to the nominal flow-biased APRM trip setpoint or the rated rod line require the hot bundle oscillation magnitude portion of the detect-and-suppress calculation to be recalculated. This calculation is not dependent upon the core and fuel design. However, the SLMCPR protection calculation is dependent upon the core and fuel design and is performed for each reload. These features will be analyzed for the first reload analysis that incorporates the new rated power level.

The licensee has determined that the [[ ]] is applicable to VYNPS. As stated in the CLTR SE, the long-term stability solution option evaluations are reload core dependent, and therefore, the licensee will perform plant-cycle-specific evaluations for each reload fuel cycle. In addition, the licensee will demonstrate that the prevention and detection/suppression features of the long-term stability solutions are either unaffected by the EPU or are modified and validated in accordance with the solution methodology. This approach is acceptable to the NRC staff.

Conclusion

The NRC staff has reviewed the licensee's [[ ]] and concludes that it is consistent with the staff's understanding described in the CLTR SE. In addition, the licensee will perform plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions, and that the core design is not susceptible to thermal-hydraulic instability. Based on this, the NRC staff concludes that the thermal and hydraulic design will continue to meet the requirements of draft GDC-6 and 7 following implementation of the proposed EPU, and is acceptable to the staff.

2.8.4 Emergency Systems

2.8.4.1 Functional Design of Control Rod Drive System

Regulatory Evaluation

The NRC staff's review covered the functional performance of the control rod drive (CRD) system to confirm that the system can effect a safe shutdown, respond within acceptable limits during AOOs, and prevent or mitigate the consequences of postulated accidents. The NRC's acceptance criteria are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a LOCA; (2) draft GDC-26, insofar as it requires that the protection system be designed to fail into a safe state; (3) draft GDC-31, insofar as it requires that the reactivity control systems be capable of sustaining any single malfunction without causing a reactivity transient which could result in exceeding acceptable fuel damage limits; (4) draft GDC-27 and 28, insofar as they require that at least two independent reactivity control systems be provided, with both systems capable of making and holding the core subcritical from any hot standby or hot operating condition, sufficiently fast to prevent exceeding acceptable fuel damage limits; (5) draft GDC-29 , insofar as it requires that at least one of the reactivity control systems be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits; (6) draft GDC-32, insofar as it requires that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling; and (7) 10 CFR 50.62(c)(3), insofar as it requires that all BWRs have an alternate rod injection (ARI) system diverse from the reactor trip system, and that the ARI system have redundant scram air header exhaust valves. Specific review criteria are contained in SRP Section 4.6 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The CRD system is described in Section 3.4 of the VYNPS UFSAR. The CRD system controls gross changes in core reactivity by positioning neutron-absorbing control rods within the reactor. The CRD system is also required to scram the reactor by rapidly inserting withdrawn rods into the core. The scram, rod insertion and withdrawal functions of the CRD system depend on the operating reactor pressure and the pressure difference between the CRD system hydraulic control unit (HCU) and the reactor vessel bottom head pressure.

The CRD system was [[ ]] evaluated in Section 5.6.3 and J.2.3.3 of ELTR1 and Section 4.4 of Supplement 1 to ELTR2. The [[ ]] evaluation concluded that the CRD systems for BWR/2-6 are acceptable for an EPU as high as 20% above the original rated power. Also, in the CLTR SE, the NRC staff concluded that no additional plant-specific calculations are required beyond confirmatory evaluations. In Section 2.5 of the PUSAR, the licensee confirmed that the [[ ]] evaluation of the scram time response, CRD positioning, CRD cooling and CRD integrity are applicable to VYNPS.

[[

]] Therefore, the current TS scram requirements are still valid.

Based on the [[ ]] in ELTR1/2 and the CLTR, the NRC staff concludes that for CRD insertion and withdrawal, there will be a minimum pressure of 250 psid between the HCU and the vessel bottom head. VYNPS UFSAR Section 3.4.5.3.1 states that "Drive pressure of about 250 psi above reactor pressure is required at a flow rate of approximately 4 GPM to insert a control rod and 2 GPM to withdraw a control rod during normal operation." The staff concludes that the VYNPS CRD pump capability and capacity are sufficient to provide the required pressure difference for operation at the EPU conditions. Based on the [[ ]] in ELTR1/2 and the CLTR, and the plant-specific evaluations, the staff concludes that the performance of the VYNPS CRD insert, withdraw, cooling and drive functions at EPU conditions will be adequate. The automatic operation of the system flow control valve maintains the required drive water pressure and the CRD positioning function is not affected.

Based on the [[ ]] in ELTR1/2, the NRC staff concludes that the required CRD cooling and drive flows are sufficient for EPU operation. The cooling and drive flows are assured by the automatic operation of the CRD system flow control valve, which would compensate for any changes in the reactor pressure.

Fuel channel bow is elongation of one fuel channel face relative to the opposite face on the same fuel channel. Fuel channel bow has been known to occur, and has been modeled in fuel licensing (thermal limits) analysis, and mitigated in core design. Previous occurrences of fuel channel bow have been known to arise from these sources: initial manufacturing, residual stress relaxation under irradiation, and differential irradiation caused by fast fluence gradients. Corrosion of the fuel channel outer surface can occur when a control blade is inserted next to

the fuel channel. Corrosion can result in [[

]] channel bowing.

On March 3, 2003, GE issued a 10 CFR Part 21 notification concerning a reportable condition of fuel channel bow. GE recommended an interim penalty of 0.02 on the operating limit MCPR (OLMCPR) for BWR/6 plants affected by the fuel channel bow phenomenon in order to maintain operation within acceptable limits. Although VYNPS is a BWR/4 plant, the licensee adopted the 0.02 penalty in the OLMCPR. On April 30, 2003, GE recommended an interim surveillance program for fuel channel bow monitoring for BWR/6 and BWR/4&5 C-lattice plants. The interim surveillance program was intended to permit affected licensees to detect fuel channel-control blade friction and take compensatory actions before reaching excessive control blade friction. GE indicated that BWR/2, 3, and 4 D-lattice plants were excluded from the interim surveillance program. Because VYNPS is a BWR/4 D-lattice plant, the licensee did not implement the recommended interim surveillance program and removed the penalty of 0.02 from the OLMCPR.

By letter dated July 14, 2005, GE revised the surveillance program for fuel channel-control blade interference. The revised surveillance program included a surveillance plan for the BWR/6 S-lattice plants and another surveillance plan for the BWR/2-5 C/D-lattice plants. VYNPS was one of the plants that GE recommended for implementation of the revised surveillance program. However, GE did not include any recommendation for additional MCPR penalties for D-lattice plants at that time.

During a telephone conversation on September 14, 2005, between the NRC staff, Entergy, and GE, the licensee indicated that it plans to implement the revised surveillance program after GE completes its assessment of the susceptible fuel cells for monitoring fuel channel-control blade interference. The licensee also indicated that, in the fuel reload analyses, there is an R-factor uncertainty which accounts for some variance of fuel power distribution, fuel assembly geometry, and fuel channel bow.

GE indicates in PUSAR Section 2.5 that the postulated abnormal operating condition for the CRD design assumes a failure of the CRD system pressure-regulating valve that applies the maximum pump discharge pressure to the CRD mechanism internal components. [[

]] With respect to external mechanical loads, PUSAR Section 3.3.2 indicates that [[ ]] No modifications or changes are required as a result of the proposed EPU. The NRC staff agrees with the licensee's determination that the CRD system integrity will remain acceptable under EPU conditions.

The CRD system capability to sustain any single malfunction without causing a reactivity transient is unaffected by the EPU. Two independent reactivity control systems (CRD system and SLCS) are still provided. The capability of either of these systems to make the core

subcritical under any conditions is unaffected by the proposed EPU. Control rod worth limits which include considerable margin are unaffected.

VYNPS has installed an alternate rod injection (ARI) system which is diverse from the reactor trip system, and the VYNPS ARI system has redundant scram air header exhaust valves.

Conclusion

The NRC staff has reviewed the licensee's plant-specific evaluation related to the effects of the proposed EPU on the functional design of the CRDS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the system's ability to perform a safe shutdown, respond within acceptable limits, and prevent or mitigate the consequences of postulated accidents will be maintained following the implementation of the proposed EPU. The NRC staff further concludes that the licensee has demonstrated that sufficient technical basis exists to ensure the system's design bases will continue to be followed upon implementation of the proposed EPU.

The present design satisfies the draft GDCs under which the plant was licensed. No system changes are required for the EPU, so the system design will continue to meet draft GDCs and current licensing bases in this technical area. Based on this, the NRC staff concludes that the CRD system and associated analyses will continue to meet the requirements of draft GDC-26, 27, 28, 29, 31, 32, 40, and 42, and 10 CFR 50.62(c)(3) following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the functional design of the CRDS.

2.8.4.2 Overpressure Protection During Power Operation

Regulatory Evaluation

Overpressure protection for the RCPB during power operation is provided by relief and safety valves and the reactor protection system. The NRC staff's review covered relief and safety valves on the main steam lines and piping from these valves to the suppression pool. The NRC's acceptance criteria are based on (1) draft GDC-9, insofar as it requires that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (2) draft GDC-33, 34, and 35, insofar as they require that the RCPB be designed to assure that it behaves in a non-brittle manner and that the probability of rapidly propagating type failures is minimized. Specific review criteria are contained in SRP Section 5.2.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

Nuclear system pressure relief system is discussed in Section 4.4 of UFSAR. The safety/relief valves provide over-pressure protection for the nuclear steam supply system (NSSS), preventing failure of the nuclear system pressure boundary and an uncontrolled release of

fission products. VYNPS has three spring safety valves (SSVs) and four safety relief valves (SRVs). These valves, together with the reactor scram function, provide over-pressure protection. The SSV and SRV setpoints are established to provide the over-pressure protection function while ensuring that there is adequate pressure difference (simmer margin) between the reactor operating pressure and the SRV actuation set points. The setpoints are also selected to be high enough to prevent unnecessary valve actuations during normal plant maneuvers. As discussed in PUSAR Section 3.1, no SSV or SRV setpoint increase is needed as a result of the proposed EPU because there is no change in dome pressure.

Since the licensee performed limiting ASME Code over-pressure analyses based on 102% of the EPU power level, and the current SSV and SRV setpoints and upper tolerance limits will not change, the NRC staff accepts the licensee's assessment that the valves will have sufficient capacity to handle the increased steam flow associated with the operation at the EPU power level. The ASME over-pressure situation is evaluated during each cycle-specific reload analysis. Therefore, the capability of the valves to ensure ASME over-pressure protection will be confirmed in the all subsequent reload analysis.

The design pressure of the reactor vessel and RCPB remains at 1250 psig. The ASME Code allowable peak pressure for the reactor vessel and the RCPB is 1375 psig (110% of the design pressure of 1250 psig), which is the acceptance limit for pressurization events. The most limiting pressurization transient is analyzed on a cycle specific basis and this approach would be applicable for each EPU reload cycle.

Section 5.5.1.4 and Appendix E of ELTR1 evaluated the ASME overpressure analysis in support of a 20% power increase, stating that the limiting pressurization transients events are the MSIV closure and turbine trip with turbine bypass failure. [[ ]] As required by the CLTR SE, the licensee analyzed the MSIV closure event based on an initial dome pressure of 1040 psia with one SRV out of service (OOS), at 102% of the EPU rated thermal power. The MSIV-position signal scram was assumed to fail and the high-flux signal scram was assumed to shut down the reactor. The MSIV closure event resulted in a maximum reactor dome pressure of 1304 psig, which corresponds to a vessel bottom head pressure of 1328 psig. Therefore, the peak calculated vessel pressure (1328 psig) remains below the ASME limit of 1375 psig. The licensee used the NRC staff-approved evaluation model ODYN with the equilibrium core to perform the EPU overpressure protection analysis consistent with the [[ ]] in Section 3.8 of ELTR2 (Reference 64).

Flow-induced vibration (FIV) may increase incidents of valve leakage. However, VYNPS currently has procedures to address a leaking SRV. FIV on the Target Rock 3-Stage safety/relief design may result in an inadvertent SRV opening and a "stuck open" SRV event. This characteristic has previously been identified and is addressed in plant procedures. The consequences of a stuck open SRV have been previously considered in the plant-specific safety analyses and have been demonstrated to be non-limiting. Increased main steam line flow may affect FIV of the piping and safety/relief valves during normal operation. The vibration

frequency, extent and magnitude depend upon plant-specific parameters, valve locations, the valve design and piping support arrangements. The FIV of the piping will be addressed by the licensee by vibration testing during initial plant operation at the higher steam flow rates (Reference 16). Attachment 1 of Reference 16 describes the FIV testing during power ascension, and the Attachment 2 provides a regulatory commitment to implement this testing.

For the VYNPS over-pressure analysis with equilibrium core, the maximum calculated pressure meets the ASME Code. In addition, the most limiting pressurization transient is analyzed for each EPU reload cycle. Therefore, the NRC staff finds that the licensee has demonstrated an acceptable analysis of the plant response to over-pressure conditions for the proposed EPU. This provides reasonable assurance that the probability of gross rupture of the RCPB or significant leakage throughout its design lifetime will continue to be exceedingly low. Since the operating ranges of RPV pressure and temperature at the proposed EPU conditions remain unchanged, the RCPB design requirement to behave in a non-brittle manner to minimize rapidly propagating failures is unaffected.

Conclusion

The NRC staff has reviewed the licensee's plant-specific analyses with equilibrium core related to the effects of the proposed EPU on the overpressure protection capability of the plant during power operation. In addition, the licensee will perform plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the NRC staff concludes that the overpressure protection features will continue to meet draft GDC-9, 33, 34, and 35 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to overpressure protection during power operation.

2.8.4.3 Reactor Core Isolation Cooling System

Regulatory Evaluation

The reactor core isolation cooling (RCIC) system serves as a standby source of cooling water to provide a limited decay heat removal capability whenever the main feedwater system is isolated from the reactor vessel. In addition, the RCIC system may provide decay heat removal necessary for coping with a station blackout. The water supply for the RCIC system comes from the condensate storage tank, with a secondary supply from the suppression pool. The NRC staff's review covered the effect of the proposed EPU on the functional capability of the system. The NRC's acceptance criteria are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of a LOCA; (2) draft GDC-37, insofar as it requires that ESFs be provided to back up the safety provided by the core design, the RCPB, and their protective systems; (3) draft GDC-51 and 57, insofar as they require that piping systems penetrating containment be designed with appropriate features as necessary to protect from an accidental rupture outside containment and the capability to periodically test the

operability of the isolation valves to determine if valve leakage is within acceptable limits; and (4) 10 CFR 50.63, insofar as it requires that the plant withstand and recover from an SBO of a specified duration. Specific review criteria are contained in SRP Section 5.4.6 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The RCIC is described in Section 4.7 of the VYNPS UFSAR. The VYNPS RCIC system provides core cooling in the event of a transient where the reactor pressure vessel (RPV) is isolated from the main condenser concurrent with the loss of feedwater (LOFW) flow, and the RPV pressure is greater than the maximum allowable for the initiation of a low-pressure core cooling system.

The maximum injection pressure for the RCIC is conservatively based on the upper analytical setpoint for the lowest available group of SRVs operating in the relief mode. For the VYNPS EPU, the normal reactor operating pressure and the SRV/SSV setpoints are unchanged, and there are no changes to the maximum specified reactor pressure for RCIC operation. The licensee states that there are [[

]]

The licensee further states that EPU operation [[

]] The required EPU surveillance testing and system injection demands would occur at the same reactor operating pressures, so there would be no change to existing system and component reliability. The LOFW transient event was evaluated for an equilibrium core, and the acceptance criterion (maintain reactor water level above top of active fuel) continues to be met for EPU conditions.

Because the licensee has analyzed the LOFW transient for EPU operation, consistent with the CLTR guidelines, and has conservatively evaluated the pressure performance requirements of the VYNPS RCIC system, and no RCIC system power dependent functions or operating requirements (flows, pressure, temperature, and NPSH) are added or changed from the original design or licensing bases, the NRC staff accepts the licensee's assessment that the RCIC will continue to meet the NRC's acceptance criteria as delineated in the Regulatory Evaluation section above.

Conclusion

The NRC staff has reviewed the licensee's plant-specific analyses with equilibrium core related to the effects of the proposed EPU on the ability of the RCIC system to provide decay heat removal following an isolation of main feedwater event and a station blackout event and the ability of the system to provide makeup to the core following a small break in the RCPB. The

NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on these events and demonstrated that the RCIC system will continue to provide sufficient decay heat removal and makeup for these events following implementation of the proposed EPU. Based on this, the NRC staff concludes that the RCIC system will continue to meet the requirements of draft GDC-37, 40, 42, 51, and 57, and 10 CFR 50.63 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the RCIC system.

2.8.4.4 Residual Heat Removal System

Regulatory Evaluation

The RHR system is used to cool down the RCS following shutdown. The RHR system is typically a low pressure system which takes over the shutdown cooling function when the RCS pressure and temperature are reduced. The NRC staff's review covered the effect of the proposed EPU on the functional capability of the RHR system to cool the RCS following shutdown and provide decay heat removal. The NRC's acceptance criteria are based on draft GDC-40 and 42, insofar as they require that ESFs be protected against dynamic effects. Specific review criteria are contained in SRP Section 5.4.7 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The RHR system is described in Section 4.8 of the VYNPS UFSAR. The RHR system is designed to (1) restore and maintain the reactor coolant inventory and to remove sensible and decay heat from the primary system and containment following reactor shutdown for both normal shutdown and post accident conditions. The RHR system is designed to operate in the low-pressure coolant injection (LPCI) mode, shutdown cooling (SDC) mode, suppression pool cooling (SPC) mode, containment spray cooling (CSC) mode and fuel pool cooling (FPC) assist mode. The LPCI mode, as it relates to the LOCA response, is discussed in Section 2.8.5.6.2 of this SE. The effects of the EPU on the other modes are described below. The results of the following evaluations are consistent with the generic evaluation in Section 4.1 of ELTR2.

The operational objective of normal shutdown is to reduce the bulk reactor temperature after scram to 125°F within approximately 20 hours using [[ ]] Single loop operation of RHR SDC is assumed for decay heat removal as part of the VYNPS Appendix R analysis in order to achieve cold shutdown within the time required by Appendix R (i.e., 72 hours). An underlying assumption in the Appendix R analysis is that one loop of RHR is unavailable due to the postulated event. The licensee's analysis shows that the time required to achieve cold shutdown (i.e., 212°F) under the Appendix R scenario conditions is less than 24 hours, and therefore, cold shutdown is achieved well within the 72-hour requirement assuming the operation of a single loop of RHR SDC. Since the SDC evaluation at EPU conditions demonstrated that the plant can meet this cooldown time, the NRC staff finds it acceptable.

The SPC and CSC modes of the RHR system cool the suppression pool following a design-basis LOCA by pumping the suppression pool water through the RHR heat exchangers and returning the water to the suppression pool, or by diverting the suppression pool water to spray headers in the drywell and wetwell after it has passed through the RHR heat exchangers. The effect of the proposed EPU with respect to these two RHR operating modes is discussed in Section 2.6 of this SE.

The FPC assist mode uses the RHR heat removal capacity to provide supplemental fuel pool cooling in the event that the fuel pool heat load exceeds the heat removal capacity of the fuel pool cooling and cleanup system. This mode can be operated separately or along with the fuel pool cooling and cleanup system to maintain the fuel pool temperature within acceptable limits. The effect of the proposed EPU with respect to FPC is discussed in Section 2.5.3 of this SE.

The licensee's application stated that the higher suppression pool temperature (194.7°F) and containment pressure during a postulated LOCA do not affect hardware capabilities of the RHR equipment, except for the RHR pump seals. The peak suppression pool temperature during a limiting LOCA remains below the RHR pump seal design temperature of 210°F. However, this temperature exceeds the maximum operating temperature of 185°F analyzed for the pump seals. In Reference 6, the licensee confirmed that the seals have been re-qualified for the increased suppression pool temperature under accident conditions.

Conclusion

The NRC staff has reviewed the licensee's plant-specific evaluation related to the effects of the proposed EPU on the RHR system. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the RHR system will maintain its ability to cool the RCS following shutdown and provide decay heat removal. Based on this, the NRC staff concludes that the RHR system will continue to meet the requirements of draft GDC-40 and 42 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the RHR system.

2.8.4.5 Standby Liquid Control System

Regulatory Evaluation

The standby liquid control system (SLCS) provides backup capability for reactivity control independent of the control rod system. The SLCS functions by injecting a boron solution into the reactor to effect shutdown. The NRC staff's review covered the effect of the proposed EPU on the functional capability of the system to deliver the required amount of boron solution into the reactor. The NRC's acceptance criteria are based on (1) draft GDC-27 and 28, insofar as they require that at least two independent reactivity control systems, preferably of different design principles, be provided, with both systems capable of making and holding the core subcritical from any hot standby or hot operating condition, sufficiently fast to prevent exceeding acceptable fuel damage limits; (2) draft GDC-29, insofar as it requires that at least one of the

reactivity control systems be capable of making the core subcritical under any condition sufficiently fast to prevent exceeding acceptable fuel damage limits; and (3) 10 CFR 50.62, insofar as it requires that the SLCS be capable of reliably injecting a borated water solution into the reactor pressure vessel at a boron concentration, boron enrichment, and flow rate that provides a set level of reactivity control. Specific review criteria are contained in SRP Section 9.3.5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The SLCS is described in Section 3.8 of the VYNPS UFSAR. The licensee evaluated the effect of the EPU on the SLC system injection and shutdown capability. The VYNPS SLC is a manually operated system that pumps concentrated sodium pentaborate solution into the vessel in order to provide neutron absorption and is capable of bringing the reactor to a subcritical shutdown condition from rated thermal power.

As discussed in PUSAR Section 6.5, the licensee stated that an increase in the core thermal power does not by itself directly affect the ability of the SLC boron solution to bring the reactor subcritical and to maintain the reactor in a safe-shutdown condition. The SLC system shutdown capability is reevaluated for each reload core.

The licensee performed a plant-specific ATWS analysis assuming EPU conditions. As a result of the analysis, the licensee has proposed changes to the TSs associated with the SLCS pump discharge pressure and the SLCS operability factors. These TS changes are evaluated in SE Sections 3.12 and 3.13, respectively. Based on the discussion in those SE sections, the NRC staff concludes that the SLCS will be capable of meeting its function of delivering the required amount of boron solution to the reactor under EPU conditions.

Two independent reactivity control systems (the CRD system and SLCS) are provided. The capability of either of theses systems to make the core subcritical under any conditions is unaffected by the EPU.

Conclusion

The NRC staff has reviewed the licensee's plant-specific evaluation related to the effects of the proposed EPU on the SLCS and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the system and demonstrated that the system will continue to provide the function of reactivity control independent of the CRD system following implementation of the proposed EPU. Based on this, the NRC staff concludes that the SLCS will continue to meet the requirements of draft GDC-27, 28, and 29, and 10 CFR 50.62(c)(4) following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the SLCS.

2.8.5 Accident and Transient Analyses

Anticipated operational occurrences (AOOs) are abnormal transients which are expected to occur one or more times in the life of a plant and are initiated by a malfunction, a single failure of equipment, or a personnel error. The applicable acceptance criteria for the AOOs are based on 10 CFR Part 50, Appendix A, Draft GDCs 6, 9, 14, 15, 27, 28, 31, and 32.

Design-basis accidents (DBAs) are not expected to occur but are postulated to occur because their consequences would include the potential for the release of significant amounts of radioactive material. They are analyzed to determine the extent of fuel damage expected and to assure that the radiological dose is maintained within 10 CFR Part 100 guidelines. The applicable acceptance criteria for DBAs such as a LOCA are based on 10 CFR 50.46, Appendix K to 10 CFR Part 50, and draft GDCs 40, 42, and 44.

The SRP provides further guidelines that (1) pressure in the reactor coolant and main steam system should be maintained below 110% of the design values according to the ASME Code, Section III, Article NB-7000, "Overpressure Protection;" (2) fuel cladding integrity should be maintained to ensure that acceptable fuel design limits are not exceeded during normal operating conditions and AOOs; (3) an incident of moderate frequency should not generate a more serious plant condition unless other faults occur independently; and (4) an incident of moderate frequency, in combination with any single active component failure or single operator error, should not result in the loss of function of any fission product barrier other than the fuel cladding. A limited number of fuel cladding perforations are acceptable.

VYNPS UFSAR Section 14.5 contains the design basis analyses that evaluate the effects of an AOO resulting from changes in system parameters such as: (1) a decrease in core coolant temperature, (2) an increase in reactor pressure, (3) a decrease in reactor core coolant flow rate, (4) reactivity and power distribution anomalies, (5) an increase in reactor coolant inventory, and (6) a decrease in reactor coolant inventory. The plant's responses to the most limiting transients are analyzed each reload cycle and are used to establish the thermal limits. A potentially limiting event is an event or an accident that has the potential to affect the core operating and safety limits.

The generic guidelines for an EPU evaluation (Appendix E of ELTR1) identified the set of limiting transients to be considered in each event category. However, VYNPS is following the CPPU approach approved by the staff's SE for the CLTR dated March 31, 2003 (Reference 52). As discussed in the staff's SE:

> The CPPU approach takes an exception to the guidelines given in ELTR1. The staff SE for ELTR1 states that: "- - -the staff agrees with the minimum set of limiting transients to be analyzed, which is contained in Appendix E of ELTR1." [[

]]

For the VYNPS proposed EPU, the licensee is following the CPPU approach which [[

]]

As discussed in Attachment 5 to VYNPS EPU Supplement 32 (Reference 33), and consistent with the discussion in Section 9.1 of the NRC staff's SE for the CLTR, the licensee will reanalyze the following transients [[ ]] for the first VYNPS EPU core:
[[

]]

In addition, the [[ ]] will be analyzed for the first VYNPS EPU core.

VYNPS UFSAR Section 14.6 evaluates the following DBAs: control rod drop accident (CRDA), LOCA, refueling accident, and main steam line break accident

The following sections provide the NRC staff's evaluation of the licensee's accident and transient analyses for the proposed EPU.

2.8.5.1 Decrease in Reactor Water Temperature

Regulatory Evaluation

Excessive heat removal causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. Any unplanned power level increase may result in fuel damage or excessive reactor system

pressure. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) postulated initial core and reactor conditions, (2) methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor system components, (5) functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; (3) draft GDC-14 and 15, insofar as they require that the core protection system be designed to act automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs; and (4) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.1.1-4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

Transients in this group included loss of feedwater heating, shutdown cooling (i.e., RHR) malfunction, and inadvertent RCIC/HPCI pump start. A feedwater heater can be lost in at least two ways: (1) if the steam extraction line to the heater is shut, the heat supply to the heater is removed, producing gradual cooling of the feedwater heater, and (2) a bypass line is usually provided so that the feedwater flow can be bypassed around rather than through the heater. In either case, the reactor vessel receives cooler feedwater which produces an increase in core inlet subcooling. Due to the negative reactivity coefficient, an increase in power results. This event will be analyzed [[ ]] as required by the CLTR SE. A shutdown cooling malfunction leading to a moderator temperature decrease could result from misoperation of the cooling water for the RHR heat exchangers. The resulting temperature decrease causes a slow insertion of positive reactivity into the core. If the reactor were critical or near critical, a very slow reactor power increase could result. If no operator action were taken to control the power level, a high neutron flux reactor scram would terminate the transient without fuel damage and without any measurable nuclear system pressure increase. This is a mild event and is bounded by the limiting events and hence need not be analyzed. The introduction of cold water to the reactor vessel will result in an increase in core power as a result of increased inlet subcooling. An inadvertent startup of the RCIC or HPCI pumps would introduce cold water to the vessel. Inadvertent startup of HPCI is severe and this event will be analyzed [[ ]] as required by the CLTR SE.

A reliable reactor protection system is provided for VYNPS. Two independent reactivity control systems (the CRD system and SLCS) are provided. The capability of either of theses systems to make the core subcritical under any conditions is unaffected by the EPU.

Conclusion

The NRC staff SE for the CLTR requires that staff-approved analytical methods will be used for the EPU core reload analysis. The NRC staff has reviewed the licensee's generic assessment and concludes that it is consistent with the staff's understanding described in the CLTR SE. In addition, the licensee will perform plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 9, 14, 15, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the events stated.

2.8.5.2 Decrease in Heat Removal

2.8.5.2.1 Increase in Reactor Pressure

Regulatory Evaluation

A number of initiating events may result in unplanned increases in reactor pressure and decreases in heat removal from the core. These events result in a sudden reduction in steam flow and, consequently, result in pressurization events. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered the sequence of events, the analytical models used for analyses, the values of parameters used in the analytical models, and the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (3) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.2.1-5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

Transients in this group included load rejection no bypass, turbine trip no bypass, and MSIV closure - direct scram. A loss of generator electrical load from high power conditions initiates main turbine control valve fast closure. Turbine control valve closure is sensed by the reactor protection system activating the reactor scram. This transient will be evaluated [[
]] A variety of turbine or nuclear system malfunctions can initiate a turbine trip. Once initiated, all of the main turbine stop valves achieve full closure within about 0.1 seconds. This event is one of the severe nuclear pressure events and will be evaluated [[
]] Generic analyses performed in ELTR1 indicated that MSIV closure with flux scram is the limiting transient for the pressurization transients and bounds other transients with respect

to RCPB overpressurization. The load rejection no bypass and turbine trip no bypass transients will be evaluated to determine the operating limit MCPR after the core configuration is finalized before the start-up.

In the CPPU approach, as part of the EPU application overpressure protection analyses, the licensee performed the analysis with an equilibrium core for the MSIV closure event with flux scram. The MSIV closure event with flux scram is described in Section 2.8.4.2 of this SE.

A reliable reactor protection system is provided for VYNPS. Two independent reactivity control systems (the CRD system and SLCS) are provided. The capability of either of these systems to make the core subcritical under any conditions is unaffected by the EPU.

Conclusion

The NRC staff has reviewed the results of the licensee's plant-specific reactor overpressure protection analyses for an equilibrium core of the events described above and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. For other events, the NRC staff expects that staff approved acceptable analytical methods will be used for the EPU core reload analysis. The NRC staff has reviewed the licensee's generic assessment and concludes that it is consistent with the staff's understanding described in the CLTR SE. In addition, the licensee will perform plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 9, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the events stated.

2.8.5.2.2 Loss of Auxiliary Power

Regulatory Evaluation

The loss of non-emergency ac power is assumed to result in the loss of all power to the station auxiliaries and simultaneous tripping of all reactor coolant recirculation pumps. This causes a flow coast-down as well as a decrease in heat removal by the secondary system, a turbine trip, an increase in pressure and temperature of the coolant, and a reactor trip. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; and (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; (3) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of

making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.2.6 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The reactor is subjected to a complex sequence of events when the station loses all auxiliary power. This can occur if all external grid connections are lost or if faults occur in the auxiliary power system itself. As discussed in Attachment 5 to Reference 5, loss of auxiliary power to the station auxiliaries is [[

]] The turbine trip with no bypass event is addressed in Section 2.8.5.2.1 of this SE.

A reliable reactor protection system is provided for VYNPS. Two independent reactivity control systems (the CRD system and SLCS) are provided. The capability of either of theses systems to make the core subcritical under any conditions is unaffected by the EPU.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the loss of non-emergency ac power to station auxiliaries event and concludes that the licensee's evaluation has adequately accounted for operation of the plant at the proposed power level. The NRC staff has reviewed the licensee's generic assessment and concludes that it is consistent with the staff's understanding described in the CLTR SE. In addition, the licensee will perform plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 9, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the loss of non-emergency ac power to station auxiliaries event.

2.8.5.2.3 Loss of Feedwater Flow

Regulatory Evaluation

A loss of normal feedwater flow could occur from pump failures, valve malfunctions, or a loss-of-offsite power (LOOP). Loss of feedwater flow results in an increase in reactor coolant temperature and pressure which eventually requires a reactor trip to prevent fuel damage. Decay heat must be transferred from fuel following a loss of normal feedwater flow. Reactor protection and safety systems are actuated to provide this function and mitigate other aspects of the transient. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6,

insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (3) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.2.7 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

Feedwater control system failures or reactor feedwater pump trips can lead to partial or complete loss of feedwater flow. Loss of feedwater flow results in a situation where the mass of steam leaving the reactor vessel exceeds the mass of water entering the vessel, resulting in a decrease in the coolant inventory available to cool the core. Generic analyses performed in ELTR1 indicated that this transient is not limiting and is bounded by other transients. VYNPS confirmed that the generic analyses performed for the loss of one feedwater pump event apply to VYNPS. The licensee performed a plant-specific calculation with a representative equilibrium core for loss of feedwater flow as required by the CPPU approach. The increased decay heat due to EPU operation results in a lower reactor water level. This analysis assumed failure of the HPCI system and used only the RCIC system to restore the reactor water level. The reactor level is automatically maintained above the top of the active fuel without any operator actions. The results of the analysis show that the minimum water level inside the core shroud is 80 inches above the top of the fuel. The core remains covered and hence no fuel cladding failure would occur.

A reliable reactor protection system is provided for VYNPS. Two independent reactivity control systems (the CRD system and SLCS) are provided. The capability of either of theses systems to make the core subcritical under any conditions is unaffected by the EPU.

Conclusion

The NRC staff has reviewed the licensee's analyses of the loss of normal feedwater flow event and concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and were performed using acceptable analytical models. The NRC staff has reviewed the licensee's generic assessment and concludes that it is consistent with the staff's understanding described in the CLTR SE. In addition, the licensee will perform plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 9, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the loss of normal feedwater flow event.

2.8.5.3 Decrease in Reactor Coolant System Flow

2.8.5.3.1 Loss of Forced Reactor Coolant Flow

Regulatory Evaluation

A decrease in reactor coolant flow occurring while the plant is at power could result in a degradation of core heat transfer. An increase in fuel temperature and accompanying fuel damage could then result if fuel design limits are exceeded during the transient. Reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) the postulated initial core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) assumed reactions of reactor systems components, (5) the functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (3) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.3.1-2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

Events in this group include recirculation flow control failure, trip of one recirculation pump and trip of two recirculation pumps. Several varieties of recirculation flow control malfunctions can cause a decrease in core coolant flow. Although the manual loading station output values are adjustable based on selectable high and low limits, the manual loading station could malfunction in such a way that a zero speed signal is generated for both recirculation flow control loops. This scenario is no more severe than the simultaneous trip of both recirculation pumps.

Normal trip of one recirculation loop is accomplished through the drive motor breaker. This transient is bounded by the trip of two recirculation pumps.

The trip of both recirculation pumps is mainly due to loss of non-emergency AC power. When the drive motor breakers are tripped, the motor-generators will continue to supply some reduced power to their respective recirculation pump motors, due to the time required for the motor-generator sets to coast down. As the core flow decreases additional voids will be formed causing a decrease in reactor power. Reactor power will decrease approximately 50% within a short time. The time constants of the fuel will cause thermal power to lag behind the neutron flux and core flow decay and the mismatch between reactor thermal power and recirculation

flow results in a decrease in critical power ratio. The MCPR will reach its lowest value in a very short time. The fuel thermal margin is provided by the rotating inertia of the motor-generator sets. [[

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A reliable reactor protection system is provided for VYNPS. Two independent reactivity control systems (the CRD system and SLCS) are provided. The capability of either of these systems to make the core subcritical under any conditions is unaffected by the EPU.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the decrease in reactor coolant flow event and concludes that the licensee's evaluation has adequately accounted for operation of the plant at the proposed power level. The NRC staff has reviewed the licensee's generic assessment and concludes that it is consistent with the staff's understanding described in the CLTR SE. In addition, the licensee will perform plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 9, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the decrease in reactor coolant flow event.

2.8.5.3.2 Recirculation Pump Rotor Seizure and Reactor Recirculation Pump Shaft Break

Regulatory Evaluation

The event postulated is an instantaneous seizure of the rotor or break of the shaft of a recirculation pump. Flow through the affected loop is rapidly reduced, leading to a reactor and turbine trip. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer which could result in fuel damage. The initial rate of reduction of coolant flow is greater for the rotor seizure event. However, the shaft break event permits a greater reverse flow through the affected loop later during the transient and, therefore, results in a lower core flow rate at that time. In either case, reactor protection and safety systems are actuated to mitigate the transient. The NRC staff's review covered (1) the postulated initial and long-term core and reactor conditions, (2) the methods of thermal and hydraulic analyses, (3) the sequence of events, (4) the assumed reactions of reactor system components, (5) the functional and operational characteristics of the reactor protection system, (6) operator actions, and (7) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-32, insofar as it requires that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling; and (2) draft GDC-33, 34, and 35, insofar as they require that the RCPB be designed

with margin sufficient to assure that, under specified conditions, it will behave in a non-brittle manner and the probability of rapidly propagating fractures is minimized. Specific review criteria are contained in SRP Section 15.3.3-4 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

As discussed in Attachment 5 to Reference 5, events in this category, [[

]]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the sudden decrease in core coolant flow events and concludes that the licensee's evaluation has adequately accounted for operation of the plant at the proposed power level. The NRC staff has reviewed the licensee's generic assessment and concludes that it is consistent with the staff's understanding described in the CLTR SE. In addition, the licensee will perform plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-32, 33, 34, and 35 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the sudden decrease in core coolant flow events.

2.8.5.4 Reactivity and Power Distribution Anomalies

2.8.5.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal from subcritical or low power startup conditions may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covered (1) the description of the causes of the transient and the transient itself, (2) the initial conditions, (3) the values of reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the

transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-14 and 15, insofar as they require that the core protection systems be designed to act automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs; and (3) draft GDC-31, insofar as it requires that the reactivity control systems be capable of sustaining any single malfunction without causing a reactivity transient which could result in exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.4.1 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

UFSAR Section 14.5.3.2, "Continuous Rod Withdrawal During Reactor Startup," states that the peak fuel enthalpies resulting from this transient event are less than 60 calories/gram (cal/gm), which is significantly less than the licensing basis criteria of 170 cal/gm. As discussed in Attachment 4 to Reference 33 (response to RAI SRXB-A-17), the current licensing basis for this event is not being changed for the EPU. Because this event is considered a non-limiting transient, it is not required to be analyzed for the EPU per the CLTR. However, the licensee did perform an evaluation of this transient for the EPU. The licensee's response to the RAI stated that peak fuel enthalpy is not expected to increase for the EPU by itself. However, indirectly, EPU fuel and core designs may lead to higher rod worth, and, therefore, higher peak enthalpy at low power. The licensee conservatively assumed that a 20% increase in rated power would increase peak fuel enthalpy at low power by 20%, resulting in a peak fuel enthalpy for this transient event of 72 cal/gm (i.e. 60 cal/gm x 1.2), which is still far below the peak fuel enthalpy limit of 170 cal/gm.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition and concludes that the licensee's evaluation has adequately accounted for the changes in core design necessary for operation of the plant at the proposed power level. The NRC staff has reviewed the licensee's generic assessment and concludes that it is consistent with the staff's understanding described in the CLTR SE. In addition, the licensee will perform plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 14, 15, and 31 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition.

2.8.5.4.2 Continuous Rod Withdrawal During Power Range Operation

Regulatory Evaluation

An uncontrolled control rod assembly withdrawal at power may be caused by a malfunction of the reactor control or rod control systems. This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The NRC staff's review covered (1) the description of the causes of the AOO and the description of the event itself, (2) the initial conditions, (3) the values of reactor parameters used in the analysis, (4) the analytical methods and computer codes used, and (5) the results of the associated analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-14 and 15, insofar as they require that the core protection systems be designed to act automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs; and (3) draft GDC-31, insofar as it requires that the reactivity control systems be capable of sustaining any single malfunction without causing a reactivity transient which could result in exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.4.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

As discussed in UFSAR Section 14.5.3.1, in the analysis of this event, it is assumed that while operating in the power range, the reactor operator makes a procedural error and fully withdraws the maximum worth control rod. Due to the positive reactivity insertion, the core average power increases. If the rod withdrawal error is severe enough, the rod block monitor (RBM) will sound alarms, at which time the operator will take corrective actions. Even for extremely severe conditions (i.e., for highly abnormal control rod patterns, operating conditions, and assuming that the operator ignores all the alarms and warnings and continues to withdraw the control rod), the fuel cladding integrity safety limit (i.e., the MCPR) or fuel rod mechanical overpower limits will not be exceeded.

As discussed in Attachment 5 to Reference 5, this event will be reanalyzed as part of the VYNPS reload evaluation.

Conclusion

The NRC staff expects that staff approved analytical methods will be used for the EPU core reload analysis. The NRC staff has reviewed the licensee's generic assessment and concludes that it is consistent with the staff's understanding described in the CLTR SE. In addition, the licensee will perform plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 14, 15, and 31

following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the uncontrolled control rod assembly withdrawal at power.

2.8.5.4.3 Startup of an Idle Recirculation Pump or Recirculation Flow Controller Failure

Regulatory Evaluation

A startup of an idle recirculation pump or a recirculation flow controller failure may result in either an increased core flow or the introduction of cooler water into the core. This event causes an increase in core reactivity due to decreased moderator temperature and core void fraction. The NRC staff's review covered (1) the sequence of events, (2) the analytical model, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (3) draft GDC-14 and 15, insofar as they require that the core protection systems be designed to act automatically to prevent or suppress conditions that could result in exceeding acceptable fuel damage limits and that protection systems be provided for sensing accident situations and initiating the operation of necessary ESFs; (3) draft GDC-32, insofar as it requires that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling; and (4) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.4.4-5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

Events in this category include recirculation flow controller failure - increasing flow, and startup of an idle recirculation pump. As discussed in Attachment 5 of Reference 5, failure of a recirculation flow controller can result in either a slow or fast recirculation increase. [[

]] Startup of an idle recirculation pump is [[

]]

Conclusion

The NRC staff has reviewed the licensee's evaluation of the increase in core flow event and concludes that the licensee's evaluation has adequately accounted for operation of the plant at the proposed power level. The NRC staff has reviewed the licensee's generic assessment and concludes that it is consistent with the staff's understanding described in the CLTR SE. In addition, the licensee will perform plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 9, 14, 15, 27, 28, and 32 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the increase in core flow event.

2.8.5.4.4 Control Rod Drop Accident

Regulatory Evaluation

The NRC staff evaluated the consequences of a control rod drop accident (CRDA) in the area of reactor physics. The NRC staff's review covered the occurrences that lead to the accident, safety features designed to limit the amount of reactivity available and the rate at which reactivity can be added to the core, the analytical model used for analyses, and the results of the analyses. The NRC's acceptance criteria are based on draft GDC-32, insofar as it requires that limits, which include considerable margin, be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling. Specific review criteria are contained in SRP Section 15.4.9 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

A CRDA is a DBA and is described in UFSAR Section 14.6.2. As discussed in Attachment 5 to Reference 5, the licensee's evaluation of a CRDA compared the maximum increase in fuel enthalpy for the proposed EPU against an acceptance criterion of 280 calories/gram (cal/gm). The 280 cal/gm acceptance criterion is identified in UFSAR Section 14.3 as a safety design limit for reactivity accidents. As discussed in UFSAR Section 3.6.6, test data indicates that the sudden fuel pin rupture threshold is about 425 cal/gm. In order to provide margin to the 425 cal/gm value, a limit on peak fuel enthalpy of 280 cal/gm was selected.

As discussed in Attachment 5 to Reference 5, if the peak fuel rod enthalpy is conservatively increased by a factor of 1.2, the CRDA peak fuel enthalpy at EPU will be 162 cal/gm. This enthalpy is well below the acceptance criterion of 280 cal/gm.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the CRDA and concludes that the licensee's evaluation has adequately accounted for operation of the plant at the proposed power level. The NRC staff has reviewed the licensee's generic assessment and concludes that it is consistent with the staff's understanding described in the CLTR SE. In addition, the licensee will perform plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-32 following implementation of the EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the CRDA.

2.8.5.5 Inadvertent Operation of ECCS or Malfunction that Increases Reactor Coolant Inventory - Feedwater Controller Failure

Regulatory Evaluation

Equipment malfunctions, operator errors, and abnormal occurrences could cause unplanned increases in reactor coolant inventory. Depending on the temperature of the injected water and the response of the automatic control systems, a power level increase may result and, without adequate controls, could lead to fuel damage or overpressurization of the RCS. Alternatively, a power level decrease and depressurization may result. Reactor protection and safety systems are actuated to mitigate these events. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; and (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (3) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.5.1-2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

As discussed in UFSAR Section 14.5.8, a feedwater controller failure transient is initiated when the feedwater flow controller may fail to the maximum demand value. This causes a quick increase in feedwater flow. The reactor water level increases until the high water level (L8) trip is initiated. When the L8 trip setpoint is reached, a high level main turbine trip occurs, the feedwater pumps are tripped and a reactor scram is initiated as a consequence of the turbine trip. The feedwater controller failure to maximum demand event is the most limiting of the

vessel inventory increase transients. As discussed in Attachment 5 to Reference 33, this transient [[ ]]

The evaluation of an inadvertent RCIC/HPCI pump start transient is included in SE Section 2.8.5.1.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the inadvertent operation of ECCS (i.e., inadvertent RCIC/HPCI pump start) or a malfunction that increases reactor coolant inventory (i.e., feedwater controller failure) and concludes that the licensee's evaluation has adequately accounted for operation of the plant at the proposed power level and was performed using acceptable analytical models. The NRC staff has reviewed the licensee's generic assessment and concludes that it is consistent with the staff's understanding described in the CLTR SE. In addition, the licensee will perform plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 9, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the inadvertent operation of the ECCS or a malfunction that increases reactor coolant inventory.

2.8.5.6 Decrease in Reactor Coolant Inventory

2.8.5.6.1 Inadvertent Opening of a Pressure Relief Valve

Regulatory Evaluation

The inadvertent opening of a pressure relief valve results in a reactor coolant inventory decrease and a decrease in RCS pressure. The pressure relief valve discharges into the suppression pool. Normally there is no reactor trip. The pressure regulator senses the RCS pressure decrease and partially closes the turbine control valves to stabilize the reactor at a lower pressure. The reactor power settles out at nearly the initial power level. The coolant inventory is maintained by the feedwater control system using water from the condensate storage tank via the condenser hotwell. The NRC staff's review covered (1) the sequence of events, (2) the analytical model used for analyses, (3) the values of parameters used in the analytical model, and (4) the results of the transient analyses. The NRC's acceptance criteria are based on (1) draft GDC-6, insofar as it requires that the reactor core be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits; (2) draft GDC-9, insofar as it requires that the reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime; and (3) draft GDC-27 and 28, insofar as they require that at least two reactivity control systems be provided and be capable of making and holding the core subcritical from any hot standby or hot operating condition sufficiently fast to prevent exceeding

acceptable fuel damage limits. Specific review criteria are contained in SRP Section 15.6.1 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

As discussed in UFSAR Section 14.5.4.2, inadvertent opening of a safety relief valve or safety valve will cause a decrease in reactor coolant inventory and result in mild depressurization. The pressure regulator senses the reactor pressure decrease and closes the turbine control valves far enough to maintain constant reactor vessel pressure. Reactor power settles out at nearly the initial power level. Automatic recirculation flow control increases the recirculation flow to the maximum. Because the recirculation flow control cannot satisfy the additional load demand, the pressure regulator setpoint is automatically reduced to its lower limit, and the reactor pressure decreases. This event will have only a slight effect on fuel thermal margins. Any change in surface heat flux is expected to be negligible indicating an insignificant change in the MCPR. As discussed in Attachment 5 to Reference 5, this event is [[

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Conclusion

The NRC staff has reviewed the licensee's evaluation of the inadvertent opening of a safety relief valve or safety valve event and concludes that the licensee's evaluation has adequately accounted for operation of the plant at the proposed power level and was performed using acceptable analytical models. The NRC staff has reviewed the licensee's generic assessment and concludes that it is consistent with the staff's understanding described in the CLTR SE. In addition, the licensee will perform plant-specific reload analyses to confirm that fuel design limits and RCPB pressure limits will not be exceeded under EPU conditions. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-6, 9, 27, and 28 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to the inadvertent opening of a pressure relief valve event.

2.8.5.6.2 Emergency Core Cooling System and Loss-of-Coolant Accidents

Regulatory Evaluation

LOCAs are postulated accidents that would result in the loss of reactor coolant from piping breaks in the RCPB at a rate in excess of the capability of the normal reactor coolant makeup system to replenish it. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. The reactor protection and ECCS systems are provided to mitigate these accidents. The NRC staff's review covered (1) the licensee's determination of break locations and break sizes; (2) postulated initial conditions; (3) the sequence of events; (4) the analytical model used for analyses, and calculations of the reactor power, pressure, flow, and temperature transients; (5) calculations of

peak cladding temperature, total oxidation of the cladding, total hydrogen generation, changes in core geometry, and long-term cooling; (6) functional and operational characteristics of the reactor protection and ECCS systems; and (7) operator actions. The NRC's acceptance criteria are based on (1) 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) 10 CFR Part 50, Appendix K, insofar as it establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a LOCA; (3) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects that might result from plant equipment failures, as well as the effects of a LOCA; and (4) draft GDC-37, 41, and 44, insofar as they require that a system to provide abundant emergency core cooling be provided so that fuel and clad damage that would interfere with the emergency core cooling function will be prevented. Specific review criteria are contained in SRP Sections 6.3 and 15.6.5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The VYNPS ECCS is described in Section 6 of the UFSAR. ECCS components are designed to provide protection in the event of a LOCA due to a rupture of the primary system piping. For a LOCA, 10 CFR 50.46 specifies design acceptance criteria based on (a) the peak cladding temperature (PCT); (b) local cladding oxidation; (c) total hydrogen generation; (d) coolable core geometry; and (e) long-term cooling. The LOCA analysis considers a spectrum of break sizes and locations, including a rapid circumferential rupture of the largest recirculation system pipe. Assuming a single failure of the ECCS, the LOCA analysis identifies the break sizes that most severely challenge the ECCS systems and the primary containment. The MAPLHGR operating limit is based on the most limiting LOCA analysis, and licensees perform LOCA analyses for each new fuel type to demonstrate that the 10 CFR 50.46 acceptance criteria can be met.

The ECCS for VYNPS includes the high-pressure coolant injection system, the low-pressure coolant injection mode of the RHR, the core spray system and the automatic depressurization system.

High Pressure Coolant Injection (HPCI)

The HPCI system is designed to pump water into the reactor vessel over a wide range of operating pressures. The primary purpose of the HPCI system is to maintain reactor vessel coolant inventory in the event of a small-break LOCA that does not immediately depressurize the reactor vessel. In this event, the HPCI system maintains reactor water level and helps depressurize the reactor vessel. The HPCI system performance was [[ ]] evaluated in the CLTR SE. For a CPPU, there is no change to the maximum nominal reactor operating pressure, and the SRV setpoints remain the same. [[

]] The NPSH required by the HPCI pump [[

]]
Since the licensee's ECCS-LOCA analysis (see section below titled, "ECCS Performance"), based on the current HPCI capability, demonstrates that the system provides adequate core cooling, the NRC staff finds the evaluation acceptable, and agrees with the licensee's assessment that the HPCI will continue to meet the NRC's acceptance criteria, as outlined in the Regulatory Evaluation section above.

Core Spray (CS)

The CS system sprays water into the reactor vessel after it is depressurized. The primary purpose of the CS system is to provide reactor vessel coolant inventory makeup for a large-break LOCA and for any small-break LOCA after the reactor vessel has depressurized. It also provides long-term core cooling in the event of a LOCA. For the proposed EPU, there is no change in the reactor pressures at which the CS is required.

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]] The NRC staff, therefore, accepts the licensee's assessment that the EPU does not significantly impact operation of the CS system. Since the licensee's ECCS-LOCA analysis (see section below titled, "ECCS Performance"), based on the current CS capability, demonstrates that the system provides adequate core cooling, the staff finds the evaluation acceptable, and agrees with the licensee's assessment that the CS will continue to meet the NRC's acceptance criteria.

Low Pressure Coolant Injection (LPCI)

The LPCI mode of the RHR system is automatically initiated in the event of a LOCA. The primary purpose of the LPCI mode is to help maintain reactor vessel coolant inventory for a large-break LOCA and for any small-break LOCA after the reactor vessel has depressurized. The LPCI operating requirements are not affected by CPPU. As required by the CLTR SE, [[

]] Since the licensee's ECCS-LOCA analysis (see section below titled, "ECCS Performance"), based on the current LPCI capability, demonstrates that the system provides adequate core cooling, the NRC staff finds the evaluation acceptable, and agrees with the licensee's assessment that the LPCI will continue to meet the NRC's acceptance criteria.

Automatic Depressurization System (ADS)

The ADS uses SRVs to reduce the reactor pressure following a small-break LOCA when it is assumed that the high-pressure systems have failed. After a specified delay, the ADS actuates either on low water level plus high drywell pressure or on sustained low water level alone. This allows the CS and LPCI to inject coolant into the reactor vessel. The licensee stated that [[

]] Since the licensee's ECCS-LOCA analysis (see section below titled, "ECCS Performance"), based on the current ADS capability, demonstrates that the system provides adequate core cooling, the NRC staff finds the evaluation acceptable, and agrees with the licensee's assessment that the ADS will continue to meet the NRC's acceptance criteria.

The EPU does not affect the protection provided for any of the ECCS features (HPCI, CS, LPCI and ADS) against the dynamic effects and missiles that might result from plant equipment failures.

ECCS Performance

The ECCS is designed to provide protection against postulated LOCAs caused by ruptures in the primary system piping. The ECCS performance under all LOCA conditions and the analysis models must satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K.

The following NRC staff approved codes were used for the equilibrium core LOCA analysis:

SAFER

The SAFER code was used to calculate the long-term-thermal-hydraulic behavior of the coolant in the vessel during a LOCA. Some important parameters calculated by SAFER are vessel pressure, vessel water level, and ECCS flow rates. The SAFER code also calculates PCT and local maximum oxidation.

LAMB

The LAMB code is used to analyze the short-term thermal-hydraulic behavior of the

coolant in the vessel during a postulated LOCA. In particular, LAMB predicts the core flow, core inlet enthalpy, and core pressure during the initial phase of the LOCA event (i.e., the first 5 seconds).

GESTR

The GESTR code is used to provide best-estimate predictions of the thermal performance of GE nuclear fuel rods experiencing variable power histories. For the LOCA analysis, the GESTR code is used to initialize the fuel stored energy and fuel rod fission gas inventory at the onset of a postulated LOCA.

TASC

The TASC code has been accepted for transient analysis and LOCA analysis. TASC is a functional replacement of the SCAT code. TASC is an improved version of the NRC-approved SCAT code, with the added capability to model advanced fuel features (partial length rods and new critical power correlation). TASC is a detailed model of an isolated fuel channel. It is used to predict the time to boiling transition for a large-break LOCA. This value is used in subsequent codes to turn off nucleate boiling heat transfer models and turn on transition boiling models.

In the CPPU approach, the LOCA analysis description is based on a [[

]] The CPPU approach [[
]] is acceptable for the following reasons:

a) The NRC staff evaluations of several requests for stretch power and extended power uprates at BWRs have shown that the change of [[

]]

b) [[

]]

c) The limiting break sizes are well known and have been shown not to be a function of reactor power level.

d) [[

]]

e) [[

]]

f) The reload evaluation confirms that the MAPLHGR for each fuel type in the specific reload core is bounded by the MAPLHGR used in the ECCS-LOCA performance analysis.

g) If the plant is MAPLHGR-limited or if the LOCA analysis results are at (or above) the acceptance criteria limits, a detailed plant-specific analysis for the licensing basis PCT will be performed.

The LOCA analysis for a CPPU builds on the existing SAFER/GESTR LOCA analyses for a plant. The CLTR SE states that the NRC staff evaluations of past EPUs at BWRs have shown that [[

]] The licensing basis PCT is based on the Appendix K PCT. [[

]] will ensure continued compliance with the requirements for the SAFER/GESTR LOCA application methodology as approved by the NRC.

The licensing basis peak clad temperature (LBPCT) for VYNPS was determined by the licensee based on the calculated Appendix K PCT at rated core flow with an adder to account for uncertainties. At both current licensing basis and EPU conditions, the limiting LOCA case for VYNPS is the large-break LOCA (LBLOCA) with maximum recirculation line break with a DC power source (battery) failure. The CPPU GE-14 LBPCT is 1960°F [[

]] This is 50°F greater than the LBPCT at the pre-CPPU conditions. Although the PCT changes due to the CPPU at VYNPS are greater than those typically seen, these changes are small compared to the margin to the 2200°F licensing limit that the bounding LBPCT provides.

[[

]] of break sizes, as required by the CLTR SE, in order to assure adequate ADS capacity. [[

]] there is sufficient ADS capacity at EPU conditions. In addition, the current VYNPS design analysis of one SRV out-of-service (OOS) out of four SRVs remains valid with the EPU.

The EPU will make a negligible effect on compliance with the other acceptance criteria of 10 CFR 50.46 (local cladding oxidation, core-wide metal-water reaction, coolable geometry). Long-term cooling is assured when the core remains flooded to the jet pump top elevation and when a core spray system is operating.

As part of its confirmatory evaluations, the NRC staff performed audit evaluations of the double-ended guillotine break in the recirculation line for the VYNPS at the EPU conditions. The

RELAP5/MOD3 code was used to investigate the effect of axial power distributions on the peak clad temperature for a double-ended recirculation line break. Based on the licensee's plant-specific LOCA analysis with an equilibrium core, the staff agrees with the licensee that the VYNPS ECCS-LOCA performance complies with 10 CFR 50.46 and Appendix K requirements.

Conclusion

The NRC staff has reviewed the licensee's plant-specific analyses of the LOCA events and the ECCS with equilibrium core. The NRC staff concludes that the licensee's analyses have adequately accounted for operation of the plant at the proposed power level and that the analyses were performed using acceptable analytical models. In addition, the licensee will perform plant-specific reload analyses to confirm that the peak cladding temperature, total oxidation of the cladding, total hydrogen generation, and changes in core geometry, and long-term cooling will remain within acceptable limits. Based on this, the NRC staff concludes that the plant will continue to meet the requirements of draft GDC-37, 40, 41, 42, and 44, and 10 CFR 50.46 following implementation of the proposed EPU, and is acceptable to the staff.

2.8.5.7 Anticipated Transients Without Scram (ATWS)

Regulatory Evaluation

ATWS is defined as an AOO followed by the failure of the reactor portion of the protection system specified in draft GDC-14 and 15. The provisions of 10 CFR 50.62 require that:

- Each BWR have an ARI system that is designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

- Each BWR have a standby liquid control system (SLCS) with the capability of injecting into the reactor vessel a borated water solution with reactivity control at least equivalent to the control obtained by injecting 86 gpm of a 13 weight-percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor vessel.

- Each BWR have equipment to trip the reactor coolant recirculating pumps automatically under conditions indicative of an ATWS.

The NRC staff's review was conducted to ensure that (1) the above requirements are met, (2) sufficient margin is available in the setpoint for the SLCS pump discharge relief valve such that SLCS operability is not affected by the proposed EPU, and (3) operator actions specified in the plant's Emergency Operating Procedures are consistent with the generic emergency procedure guidelines/severe accident guidelines (EPGs/SAGs), insofar as they apply to the plant design. In addition, the NRC staff reviewed the licensee's ATWS analysis to ensure that (1) the peak vessel bottom pressure is less than the ASME Service Level C limit of 1500 psig;

(2) the peak clad temperature is within the 10 CFR 50.46 limit of 2200°F; (3) the peak suppression pool temperature is less than the design limit; and (4) the peak containment pressure is less than the containment design pressure. The NRC staff also evaluated the potential for thermal-hydraulic instability in conjunction with ATWS events using the methods and criteria approved by the NRC staff. For this analysis, the NRC staff reviewed the limiting event determination, the sequence of events, the analytical model and its applicability, the values of parameters used in the analytical model, and the results of the analyses. Review guidance is provided in Matrix 8 of RS-001.

Technical Evaluation

The ATWS prevention/mitigation systems for VYNPS are discussed in UFSAR Section 7.18. The licensee's EPU evaluation for ATWS is provided in PUSAR Section 9.3.

The ATWS analyses assume that the SLCS will inject within a specified time to bring the reactor subcritical from hot full power and maintain the reactor subcritical after the reactor has cooled to the cold-shutdown condition. For every reload, the licensee evaluates how plant modifications, reload core designs, changes in fuel design, and other reactor operating changes affect the applicability of the ATWS analysis of record.

The licensee stated that VYNPS meets the ATWS mitigation requirements defined in 10 CFR 50.62, because (a) an ARI system is installed, (b) the boron injection capability is equivalent to 86 gpm, and (c) an automatic ATWS recirculating pump trip has been installed. Section L.3 of ELTR1 discusses the ATWS analyses and provides a generic evaluation of the following limiting ATWS events in terms of overpressure and suppression pool cooling: (1) MSIV closure; (2) pressure regulator failure to open; (3) loss of offsite power, and (4) inadvertent opening of a relief valve. The licensee performed plant-specific ATWS analyses at the EPU operating conditions to demonstrate that VYNPS meets the ATWS acceptance criteria.

As noted in the CLTR SE, boron injection from the SLCS is assumed to start at the later of either (a) reaching the boron injection initiation temperature (BIIT) or (2) 2 minutes after the ATWS recirculation pump trip on either low reactor water level or high reactor pressure. As discussed in Attachment 10 to Reference 31, the ATWS analyses of VYNPS at EPU conditions, the SLCS initiation is assumed to occur at 2 minutes after the recirculation pump trip. In addition to boron injection, a number of operator actions (consistent with the emergency operating procedures (EOPs)) are assumed in the VYNPS ATWS analyses. These operator actions are assumed to occur at the same time or later than the timing assumed in the pre-uprate ATWS analyses, consistent with the CLTR SE. The ATWS analysis methodology assumes operator action to reduce feedwater flow to the reactor in order to decrease reactor water level. This action occurs at the later of either reaching the BIIT or 90 seconds after the MSIV closure. In the ATWS analyses of VYNPS at EPU conditions, this event was assumed to be initiated by operator action at the BIIT. The ATWS methodology also assumes operator action to initiate torus cooling. For VYNPS the time at which operators initiate torus cooling was

increased from the 10 minutes assumed in the pre-uprate ATWS analysis to 15 minutes at EPU conditions. This assumption, while increasing margin for operator action, is more conservative because it allows additional torus water heat-up prior to initiating cooling.

Table 9-4 of the PUSAR lists the key input parameters used in the ATWS analyses and Table 9-5 lists the corresponding results (peak vessel bottom pressure, peak cladding temperature, peak suppression pool temperature, and peak containment pressure). The licensee stated that the results of the ATWS analyses meet the ATWS acceptance criteria.

Since the ATWS analyses are based on NRC-approved methods and the licensee performed the ATWS analyses at the EPU conditions, the NRC staff finds the licensee evaluation to be acceptable.

Conclusion

The NRC staff has reviewed the information submitted by the licensee related to ATWS and concludes that the licensee has adequately accounted for the effects of the proposed EPU on ATWS. The NRC staff has reviewed the licensee's plant-specific ATWS analyses with an equilibrium core. The NRC staff concludes that the licensee confirmed that ARI, SLCS, and recirculating pump trip systems will continue to meet the requirements of 10 CFR 50.62. Therefore, the NRC staff finds the proposed EPU acceptable with respect to ATWS.

2.8.6 Fuel Storage

2.8.6.1 New Fuel Storage

Regulatory Evaluation

Nuclear reactor plants include facilities for the storage of new fuel. The quantity of new fuel to be stored varies from plant to plant, depending upon the specific design of the plant and the individual refueling needs. The NRC staff's review covered the ability of the storage facilities to maintain the new fuel in a subcritical array during all credible storage conditions. The review focused on the effect of changes in fuel design on the analyses for the new fuel storage facilities. The NRC's acceptance criteria are based on draft GDC-66, insofar as it requires the prevention of criticality in fuel storage systems by physical systems or processes, preferably utilizing geometrically safe configurations. Specific review criteria are contained in SRP Section 9.1.1 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

As discussed in Attachment 10 to Reference 31 and Attachment 6 to Reference 33, the licensee performed an evaluation to assess the impact of the EPU on VYNPS new fuel storage. On the basis of this assessment, the licensee has determined that for EPU, VYNPS is bounded by the requirements of current licensing basis, and that there is no need to change the licensing

basis requirements for the new fuel storage, as currently listed in TS 5.5. These requirements are:

a) The new fuel storage facility shall be such that the effective multiplication factor (Keff) of the fuel when dry is less than 0.90 and when flooded is less than 0.95.

b) The maximum core geometry infinite lattice multiplication factor of any segment of the fuel assembly stored in the new fuel storage facility shall be less than or equal to 1.31 at 20°C.

Based on the NRC staff's review of the licensee's evaluation and rationale, the NRC staff concurs with the licensee that plant operation at the proposed EPU level will have an insignificant impact on the new fuel storage discussed above, and therefore, no modifications are necessary. Since it is not necessary to change the original design or licensing bases, the staff accepts the licensee's assessment that the new fuel storage will continue to meet the NRC's acceptance criteria as delineated in the Regulatory Evaluation section above.

Conclusion

The NRC staff has reviewed the licensee's evaluation related to the effect of the EPU on the analyses for new fuel storage facilities and concludes that the new fuel storage facilities will continue to meet the requirements of draft GDC-66 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to new fuel storage.

2.8.6.2 Spent Fuel Storage

Regulatory Evaluation

Nuclear reactor plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel pool and storage racks is to maintain the spent fuel assemblies in a safe and subcritical array during all credible storage conditions and to provide a safe means of loading the assemblies into shipping casks. The NRC staff's review covered the effect of the proposed EPU on the criticality analysis (e.g., reactivity of the spent fuel storage array and boraflex degradation or neutron poison efficacy). The NRC's acceptance criteria are based on (1) draft GDC-40 and 42, insofar as they require that protection be provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a LOCA; and (2) draft GDC-66, insofar as it requires that criticality in the fuel storage systems be prevented by physical systems or processes, preferably by use of geometrically safe configurations. Specific review criteria are contained in SRP Section 9.1.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

As discussed in Attachment 10 to Reference 31 and Attachment 6 to Reference 33, the licensee performed an evaluation to assess the impact of the EPU on VYNPS spent fuel storage. On the basis of this assessment, the licensee has determined that for the EPU, VYNPS is bounded by the requirements of the current licensing basis, and that there is no need to change the licensing basis requirements for spent fuel storage, as currently listed in TS 5.5. These requirements are:

a) The Keff of the fuel in the spent fuel storage pool shall be less than or equal to 0.95.

b) Spent fuel storage racks may be moved (only) in accordance with written procedures which ensure that no rack modules are moved over fuel assemblies.

c) The number of spent fuel assemblies stored in the spent fuel pool shall not exceed 3353.

d) The maximum core geometry infinite lattice multiplication factor of any segment of the fuel assembly stored in the spent fuel storage pool shall be less than or equal to 1.31 at 20°C.

The licensee has performed analysis which shows that ensuring the infinite multiplication factor (Kinf) of any fuel segment less than 1.31 will ensure that the Keff remains below 0.95. For each reload, the fuel vendor, currently Global Nuclear Fuel, calculates Kinf at 20°C for each different fuel lattice type to be utilized, as a function of void history and lattice exposure. The calculations address the change in elements/isotopes including plutonium. VYNPS ensures that the peak Kinf is less than 1.31 for all fuel lattice types used in the reload.

In addition, the EPU does not affect the protection provided for ESFs against the dynamic effects and missiles that might result from plant equipment failures, as well as the effects of a LOCA.

Based on the NRC staff's review of the licensee's evaluation and rationale, the NRC staff concurs with the licensee that plant operation at the proposed EPU level will have an insignificant impact on the spent fuel storage discussed above, and therefore, no modifications are necessary. Since it is not necessary to change the original design or licensing bases, the staff accepts the licensee's assessment that spent fuel storage will continue to meet the NRC's acceptance criteria as delineated in the Regulatory Evaluation section above.

Conclusion

The NRC staff has reviewed the licensee's evaluation related to the effects of the proposed EPU on the spent fuel storage capability and concludes that the licensee has adequately accounted for the effects of the proposed EPU on the spent fuel rack temperature and criticality analyses. The NRC staff also concludes that the spent fuel pool design will continue to ensure an acceptably low temperature and an acceptable degree of subcriticality following

implementation of the proposed EPU. Based on this, the NRC staff concludes that spent fuel storage at VYNPS will continue to meet the requirements of draft GDC-40, 42, and 66 following implementation of the proposed EPU. Therefore, the NRC staff finds the proposed EPU acceptable with respect to spent fuel storage.

2.8.7 Additional Review Area - Methods Evaluation

2.8.7.1 Application of NRC-approved Analytical Methods and Codes

The analyses supporting safe operation at EPU conditions are required to be performed using NRC-approved licensing methodology, analytical methods and codes. In general, the analytical methods and codes are assessed and benchmarked against measurement data, comparisons to actual nuclear plant test data and research reactor measurement data. The validation and benchmarking process provides the means to establish the associated biases and uncertainties. The uncertainties associated with the predicted parameters and the correlations modeling the physical phenomena are accounted for in the analyses. NRC-approved licensing methodology, topical reports and codes specify the applicability ranges. The generic licensing topical reports (LTR) covering specific analytical methods or code systems quantify the accuracy of the methods or the code used. The safety evaluation reports approving topical reports include restrictions that delineate the conditions that warrant specific actions, such as obtaining measurement data or obtaining further NRC approval. In general, the use of NRC-approved analytical methods is contingent upon application of these methods and codes within the ranges for which the data were provided and against which the methods were evaluated. Thus, a plant-specific application does not entail review of the NRC-approved analytical methods and codes.

To implement the proposed EPU and maintain the current 18-month cycle, a higher number of maximum powered bundles are loaded into the core and the power of the average bundles is also increased, making the core radial power distribution flatter. Due to an increased two-phase pressure drop and higher coolant voiding, the flow in the maximum powered bundles decreases. This effect leads to a higher bundle power-to-flow ratio and higher exit void fraction. Since the maximum powered bundles set the thermal limits, EPU operation reduces the margins to thermal limits.

Table 2.8.7-1 below shows the predicted operating conditions for the maximum powered bundles for VYNPS as shown in Table 6-2 of Attachment 3 to Reference 25. Figures 2.8.7-1 through 2.8.7-4 show plots for some of these parameters for VYNPS throughout the core cycle.

Table 2.8.7-1 Ranges of Operational Experience

Metric	VYNPS Prediction
[[	
	]]

As shown, the VYNPS maximum exit void fraction is 87% and the core average bundle exit void fraction is 76%.

2.8.7.2 Applicability of Neutronic Methods

2.8.7.2.1 Methods Review Topics

In Enclosure 3 to a letter dated March 4, 2004, (Reference 69) GE provided its evaluation of the impact of operation at higher void conditions on all of GE's licensing methodologies. The generic evaluation was also based on core thermal-hydraulic conditions that bound the EPU conditions (void fraction 90% or greater). Specifically, operation with a large number of bundles operating at high in-channel void fractions could potentially affect the following topics:

1. Assumptions made in the generation of the lattice physics data that establish the neutronic feedback (see SE Section 2.8.7.2.2).

2. Accuracy of the fuel isotopics generated considering the method employed in the lattice physics (see SE Section 2.8.7.2.2).

3. Assumptions made in the generation of the neutronic parameters in assuming 0% bypass voiding, although voiding is present during some transients (see SE Section 2.8.7.2.2).

4. Applicability of the thermal-hydraulic correlations used to model physical phenomena (see SE Section 2.8.7.3).

5. Reliability of the response and prediction of the instrumentation located in the out-channel regions (traversing in-core probes (TIPs) and local power range monitors (LPRMs), see SE Section 2.8.7.4).

Section 2.8.7.2 of this SE evaluates the applicability of GE's neutronic methods to the EPU core thermal-hydraulic conditions. Entergy proposed an "Alternative Approach," which involves evaluating available margins in key safety parameters that are important to safety analyses supporting operation at the EPU conditions. This section evaluates the viability of the "Alternative Approach." The NRC staff's evaluation also includes the impacts of bypass voiding on the accuracy of the generated lattice physics data. SE Section 2.8.7.3 covers the applicability of some of the thermal-hydraulic models and correlations that support the EPU analyses. The evaluation confirms whether the thermal-hydraulic models are being applied within the ranges that the correlations and models were developed, benchmarked and approved. SE Section 2.8.7.4 focuses on the impact of potential bypass voiding during transient conditions on the accuracy of the neutron monitoring system. Specifically, that section covers the impact of potential bypass voiding on the Option I-D stability solution. SE Section 2.8.7.5 presents the conclusions of the SE Section 2.8.7 review. To support this review, an audit of the reactor neutronic and thermal-hydraulic analyses was performed by the NRC staff and a contractor from Oak Ridge National Laboratory (ORNL) at GE's Washington, DC office on September 7, 2005.

2.8.7.2.2 Neutronic Methods Evaluation

2.8.7.2.2.1 Extrapolation of Neutronic Parameters and Code Qualification

The neutronic parameters feed into other codes that are used to perform the steady state, transient and accident conditions and establish the core operating thermal limits. Therefore, the accuracy of the methods to calculate the neutronic parameters affects the analyses supporting operation at EPU conditions.

Neutronic Methods Not Assessed For Void Fraction Greater than 70%

The GE lattice physics (TGBLA) and core simulator code (PANACEA) have been assessed for void fractions up to a void fraction of 70%. The neutronic method employed generates the cross-sections at 0%, 40%, and 70% void fractions. The neutronic data (e.g., Kinf, cross-sections and pin power peaking) are parameterized by a [[ ]] For nodes operating above 70% void conditions, the neutronic parameters are obtained by extrapolating the [[ ]] however, the NRC staff determined that the extrapolation process used by GE was not evaluated or documented by the staff.

The NRC staff review focused on ensuring that the impact of any extrapolation errors in the Kinf and the cross-sections are accounted in the bundle power calculations, which in turn affect the steady-state core thermal-hydraulic conditions (e.g., radial and axial power distribution and peaking, void fraction). The steady-state conditions set the initial conditions for the steady-state

reactivity analyses (e.g., shutdown margin, standby liquid control shutdown capability, ATWS boron worth calculations), the transients (e.g., ASME overpressure, feedwater failure at maximum demand, instability) and accident (e.g., LOCA) conditions. Extrapolation and depletion errors for operation at voids greater than 70% in the bundle and pin power predictions also affect the calculation of the core operating steady-state limits such as the SLMCPR, the operating LHGR kW/ft, the operating MAPLHGR and the pellet nodal exposure accounting.

Errors associated with the predictions of the neutronic methods for operation at high void conditions are not limited to the biases associated with the lattice physics data generated by TGBLA or the errors associated with fit/extrapolation techniques employed by PANACEA, but include the additional inherent errors and biases associated with the neutronic and thermal-hydraulic method employed by the core simulator code (PANACEA). Therefore, establishing the errors associated with the neutronic method in its predictions of bundle and pin powers as depleted for the current operating strategies would require performing actual bundle and pin power measurement data (e.g., gamma scans and heavy isotopic and fission inventory measurement).

Impact of Using of 40% Depletion in Generating the Branch Cases

In the GE approach, the instantaneous branch cases are based on the [[ ]] void fraction (VF) conditions. In addition, the instantaneous cases are based on isotopic depletion assuming cross sections generated at a 40% void fraction condition for the curve.

The branch cases establish the core neutronic response to sudden changes in the void fractions during transients (e.g., a pressurization transient) and ATWS. Specifically, any errors in the neutronic method used in the instantaneous cases would affect the key parameters such as the void reactivity coefficient, which in turn affects changes in the critical power ratio during transients, reactivity/power response during overpressure and instability response and the thermal and mechanical LHGR overpower response.

Comprehensive qualification of the GE steady-state neutronic method (TGBLA04/PANAC10) was last performed in 1985. In LTR NEDE-30130-P-A, "Steady State Nuclear Methods," GE qualified its methods for fuel designs and operating strategies of the time with TIP measured/calculated comparisons (core follow data), gamma scan comparisons, cold critical measurements and isotopic burnup verifications. The initial TGBLA/PANAC LTR also included measured fissionable nuclide densities (isotopic inventory) and rod exposure measurements.

Since the initial qualification of the steady-state neutronic methods in 1985, GE did not perform any gamma scans to benchmark the codes' adequacy in predicting the bundle and pin powers for the current fuel designs and for the current operating strategies (depletion at higher void conditions). Without measurement data, the neutronic methods' capability to predict bundle and pin powers or the impact of depletion at high void conditions cannot be fully assessed and the use of specific values for computational uncertainties cannot be established.

GE relies heavily on TIP measured/calculated 4 bundle power peaking and code-to-code comparisons (MCNP/TGBLA). Recently, GE had compiled comprehensive core follow TIP comparisons for plants that have uprated and for high density plants. However, core follow data, while useful for monitoring core performance, is not always sufficient for qualifying neutronic code systems. Section 5.2 of the initial TGBLA/PANAC licensing report compared the relative merits of using TIP comparisons (core follow) for validating neutronic code systems, stating, "The TIP signals provide a good picture of the axial power distribution, but do not provide a detailed bundle by bundle distribution, because there is only 1 TIP location for every 16 bundles. A more accurate estimate of the reactor power distribution can be obtained just prior to a reactor shutdown by the procedure known as gamma scanning ..."

Similar to the statement in the TGBLA LTR in Section 5.2, the NRC-approved SLMCPR technical evaluation report recognizes the limitation of core follow data and code-to-code comparisons to establish the bundle and pin power uncertainties.

In a letter dated June 20, 2005 (Reference 70), GE committed to perform gamma scan measurements to confirm that the assumptions used in the neutronic method are still appropriate. GE had also presented plans for gamma and plenum fission gas scans. The plan did not include isotopic inventory measurement.

Given that the specific measurement data would not be available for some time, the NRC staff review shifted and focused on the impacts and sensitivity of the safety limits to assure that sufficient margins are available to compensate for potential underpredictions until such time that the neutronic methods are confirmed against measurement data on a plant-specific basis.

2.8.7.2.2.2 Alternative Approach

In Reference 31, Entergy provided a response to NRC staff RAI SRXB-A-6. In the submittal, Entergy proposed an alternative approach to address the lack of measurement data to benchmark the neutronic method used to perform the safety analyses supporting the VYNPS EPU application. The licensee identified those fuel parameters that cannot be measured directly by the plant nuclear instrumentation as follows:

1. Critical power (controlled by SLMCPR and OLMCPR)

2. Shutdown margin (controlled with TS limit of 0.38% ΔK/K)

3. Fuel rod thermal mechanical performance (controlled by limits on LHGR)

4. LOCA-related nodal power limits (controlled via the MAPLHGR)

5. Stability (protected by the SLMCPR, OLMCPR and stability solutions), and

6. Licensed pellet exposure [[ ]]

The licensee states that other nodal fuel and bundle powers parameters (e.g., lattice reactivity, bundle power and bundle axial power shape) are adequately confirmed by comparisons to plant operating instrumentation (i.e., TIP) and through the shutdown margin demonstrations. The fuel design limits or safety analyses that are affected by the uncertainties in the important parameters are:

1. SLMCPR
2. Shutdown margin
3. Fuel rod thermal mechanical performance (steady-state and transient LHGR)
4. LOCA (MAPLHGR)
5. Stability
6. Licensed pellet exposure [[ ]]

Entergy proposed evaluating the adequacy of the margins available in these fuel design limits and safety analyses. The NRC staff concurs with this approach, because potential inaccuracies in the cross-sections and the pin powers would predominantly affect (1) the calculations of the steady-state thermal limits (SLMCPR, LHGR); (2) the predictions of the initial steady-state conditions assumed in the transient analyses; and (3) the key parameter that influences the transient response (void coefficient) and its impact on the reactivity initiated events, transients, instability and LOCA. Evaluation of the available margins in these key principal topics would provide reasonable assurance that the impact of potential underpredictions are accounted for during this interim period (i.e., until measurement data is available).

In addition, Entergy was in a position to perform lattice physics code-to-code comparisons, using another independent code (CASMO/4) to assess the impact of depletion under high void conditions. In Reference 35, Entergy's response to NRC staff RAI SRXB-A-66 provides the CASMO/4 and TGBLA comparisons. The staff had also performed confirmatory lattice physics calculations in order to evaluate the adequacy of the lattice physics techniques employed in GE's neutronic method for operation at high void conditions. The objective of the code-to-code comparisons was not to establish the uncertainties in GE's lattice physics calculation or benchmark the code, but to trend the lattice physics code's performance for the current operating strategy.

The following sections describe the licensee and NRC staff evaluations of the margins available in the (1) SLMCPR; (2) LHGR; (3) exposure limit; (4) MAPLHGR; (5) shutdown margin and related subjects; and (6) void coefficient. To support the margin evaluations, the licensee provided code-to-code comparisons using CASMO/4.

2.8.7.2.2.3 Fuel Design Limits and Safety Analyses Margin Evaluation

2.8.7.2.2.3.1 SLMCPR

The SLMCPR methodology and the associated uncertainty treatments are specified in GE LTRs NEDC-32601P-A and NEDC-32694P-A. It is the NRC staff opinion that the technical evaluation report approving the SLMCPR licensing methodology stated that the pin and [[ ]] would be confirmed through gamma scan for new fuel designs and operating strategy. Review of the SLMCPR methodology indicates that the [[ ]] and the peak pin uncertainty P_{peak} should have been confirmed by GE through pin and [[ ]] gamma scans for the GE-14 fuel.

Entergy proposed [[

]] Therefore for VYNPS, the licensee will take a penalty on the SLMCPR and increase the calculated SLMCPR by 0.02 for additional safety margin. This penalty will be established by a license condition as shown in SE Section 3.17.1.

The NRC staff raised a concern regarding what confidence is provided that gamma scans of the GE-14 fuel operated at conditions similar to VYNPS core conditions would not yield higher pin and [[ ]] uncertainties. This concern is addressed below.

TGBLA and CASMO/4 are two independent production codes. Although, CASMO/4 cannot be used to benchmark TGBLA, trending of the performance of the two lattice physics codes provide some useful insights (see SE Section 2.8.7.2.2.3.7 on code-to-code comparisons). Trending of the local power peaking in TGBLA/CASMO comparisons shows that the two codes are mostly consistent for depletion at 70% VF for low exposures, with TGBLA overpredicting at high exposures.

SE Section 2.8.7.2.2.3.7, concerning the code-to-code comparison, discusses the consistency of the performance of TGBLA and CASMO/4 pin power peaking and Kinf with depletion. The [[ ]] cannot be assessed because the cross-section comparisons were not provided. In summary, there are some consistent deviations that can be seen between the codes that reflect known TGBLA behavior, however, with increased uncertainties, TGBLA pin power peaking is acceptable (i.e., increased uncertainties would account for any potential underprediction).

Considering that the additional margin was obtained by a more conservative statistical treatment than currently used in the NRC-approved process, a 0.02 margin is considered to be a good SLMCPR margin. The code-to-code trending does not indicate degraded performance of the corrected TGBLA at high void conditions. Therefore, the staff accepts the 0.02 SLMCPR increase as sufficient in providing adequate margin, until the neutronic method is confirmed against appropriate measurement data.

2.8.7.2.2.3.2. LHGR Limit

The linear heat generation rate (LHGR) is a thermal-mechanical limit that assures the integrity of the fuel cladding during steady-state and transient conditions. During heat-up, a limit is placed on the peak pin nodal power to assure that the diametric strain would not result in [[ ]] (e.g., due to differential pellet/cladding creep and swelling). During a transient, the fuel pellet experiences overpower, which could result in fuel centerline melt. Therefore, a limit is also placed on the peak pin nodal power to prevent fuel centerline melting during any transient event. The peak kW/ft limit is exposure dependent and the thermal and mechanical limit establishes the steady-state kW/ft value. The peak kW/ft limit is an indicator of the peaking in the core since it comprises the combination of radial, axial, and local (pin) power peaking.

Margins in the operating LHGR kW/ft are of interest because the accuracy of the local pin peaking and the bundle power are contributors to the nodal pin kW/ft value. The table below shows the power/exposure dependent LHGR limit for GE-14 uranium dioxide (UO2) and gadolinium (Gd) rods. [[

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In general, core monitoring operating data indicate that plants can operate with the peak pin at the LHGR limit for some limited amount of time. Therefore, any underpredictions in the nodal peak pin power peaking, the nodal bundle power and its operating history would translate to errors in the calculations of the operating pellet kW/ft with depletion. Peak rods in a bundle could be operating at the LHGR limit because of high bundle and pin power peaking. The peak

rods could also be operating at the LHGR limit because the limit is low for a given exposure due to the impact of burnup on the internal rod pressures. The once-burned fuel (at end-of-cycle (EOC)) and twice-burned fuel (early in the cycle) is expected to be operating at peak bundle powers. Therefore, for twice-burned fuel, the peak rod would be close to the LHGR limit, assuming a UO2 rod is setting the limit. The presence of poison (Gd) at low burnups also reduces the LHGR limit. However, initially the Gd rods would be operating at lower power levels than the UO2 rods, until the Gd burns out with exposure, depending on the initial concentration of the Gd (e.g., 13 GWd/MTU).

Figure 2.8.7-1 shows the VYNPS predicted peak kW/ft of the maximum powered bundles against the cycle exposure. The operating LHGR limit calculations shown in Figure 2.8.7-1, for VYNPS at a 120% power level, are not based on cycle-specific core design but rather a projected core design (reference core design). For VYNPS at a 115% power level, the predicted cycle and core specific operating LHGR kW/ft appears to remain around 12 kW/ft. Note that Figure 2.8.7-1 plots the peak LHGR against cycle exposure. Therefore, the exposure accumulated by the peak bundle that is setting the bundle peak LHGR and its corresponding LHGR limit is not apparent from the plots. The figure also does not show if the UO2 rods or the Gd rods are setting the peak kW/ft and what the corresponding exposures are.

As discussed earlier, the NRC staff evaluation focused on the available margins in the predicted operating kW/ft value to ensure that potential underpredictions, due to the lack of pin and bundle axial power measurements data, would not lead to exceeding the LHGR limit and the peak pellet exposure.

The code-to-code comparison section discusses the consistency of the performance of TGBLA and CASMO/4 pin power peaking and Kinf with depletion. The bundle powers cannot be assessed because the cross-section comparisons were not provided. In summary, some consistent deviations can be seen between the codes, which reflect known TGBLA performance as well as some overpredictions by TGBLA. The increased pin and bundle power uncertainties in the proposed Alternative Approach is limited to the SLMCPR calculations. The code-to-code comparisons do show for both codes increased peaking at high exposures with high void conditions. This holds for 70% VF as well, although it is more pronounced for the 90% VF. Although the highly exposed bundles would be expected to operate at lower power levels, the twice-burned and thrice-burned peak rod set the operating peak kW/ft. Therefore, increases in the pin power peaking with exposure affects the margins to the exposure dependent LHGR limit.

The Alternative Approach submitted by VYNPS stated that the key conservatism in the development of the LHGR limit is that the peak power node is assumed [[

]] It also states that (1) a [[ ]] conservative bias is applied in the fuel rod internal pressure calculation, and (2) an additional power uncertainty of [[ ]] is applied that is not specifically assigned to any cause.

In Reference 31, Entergy's response to NRC staff RAI SRXB-A-41 provided a kW/ft uncertainty treatment, including a higher [[ ]] and peak pin uncertainty P_{peak}. The response stated that [[ ]] The licensee's response to NRC staff RAI SRXB-A-65 in Reference 35 stated that [[ ]] The NRC staff finds this explanation reasonable, although no specific additional "no cause" margin is available.

In Reference 35, Entergy's response to the NRC staff's RAI SRXB-A-65 added clarification stating that the [[ ]] uncertainty applied to the fuel rod internal pressure calculations accounts for [[ ]] Therefore, no uncertainty is applied to ensure that the operating histories (e.g., radial and axial pin power distribution and peaking) assumed in development of the LHGR limit bounds all plant operating history (e.g., effects on internal rod pressures).

The licensee's response to RAI SRXB-A-65 provided a detailed discussion of the inherent conservative assumptions in the generation of the limit. The licensee cited the following justifications for ensuring that (1) the operation at the LHGR limit would not result in exceeding the fuel thermal-mechanical acceptance criteria, and (2) the assumed operating history in development of the LHGR limit bounds VYNPS actual operating history:

- At any point, the fuel rod power level could potentially approach or even be at the LHGR operating limit, [[ ]]
- [[ ]] The licensee provided the actual operating history of a VYNPS pin against the LHGR envelope (see Figure 2.8.7-5). Although, the pin selected was not the rod that operated at peak power or the pin that experienced the highest power at the higher exposures, the comparisons of the selected actual VYNPS pin operating history showed that there was sufficient margin to the LHGR envelope.

To demonstrate the conservatism [[

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The responses also included demonstration of how a [[ ]] additional uncertainty is included in the GESTR-Mechanical statistical treatment of uncertainties. Based on its review, the NRC staff confirmed that the [[ ]] uncertainty applied in the LHGR limit calculations, although not intended for the core monitoring uncertainty, is not applied due to other considerations. As such, this uncertainty adds conservatism to the calculation of the LHGR limit that may offset other potential uncertainties in the prediction of the steady-state operating kW/ft.

As mentioned earlier, the process of establishing the errors associated with the neutronic method (both the lattice physics and core simulator) in its predictions of bundle and pin powers as depleted for the current operating strategies would require performing actual bundle and pin power measurement data (e.g., pin and bundle axial power gamma scans and heavy isotopic and fission gas inventory measurement (rod internal pressure)). Specifically, when bundles are operating at the limit, it is important to have assurance that the accuracies of the analytical tools used are validated.

Considering GE's commitment to perform the needed measurement data, the conservatism introduced in the calculation of the LHGR limit that may offset other potential uncertainties in the predictions of the steady-state operating kW/ft, the power distribution uncertainty applied to the generation of the LHGR limit and [[

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the NRC staff accepts that no additional uncertainties need to be applied to the LHGR limit.

2.8.7.2.2.3.3. Exposure Limit

The GE fuel designs are licensed to a peak pellet exposure limit of [[ ]] which is equivalent to a GE-14 rod average exposure of approximately [[ ]] However, there is no explicit rod average exposure limit for the GE fuel design method. The exposure limit assures that the fuel rod is not operated beyond the bases supporting the thermal-mechanical limit (e.g., fuel rod internal pressure acceptance criteria). The local pin power peaking, the bundle power and the void coefficient are all factors that contribute to meeting the LHGR limit and thus operating within the exposure limit.

The fuel rod internal pressure is the limiting criterion for the end-of-life for the GE fuel thermal-mechanical performance. The fuel rod internal pressure is limited such that [[

]] With exposure, the fission gas release and hence fuel rod internal pressure increase. Therefore, the fuel rod internal pressure is a key factor in the fuel rod thermal-mechanical performance at the exposures near the peak pellet exposure limit

[[ ]] Underpredictions in the exposure due to potential underpredictions in the fuel rod and bundle powers would affect the fuel thermal-mechanical design acceptance criteria.

The fuel exposure is monitored by the core simulator [[ ]] To demonstrate that operating at the exposure limit would not result in exceeding the fuel design acceptance criteria, the licensee cited the internal rod pressure analysis performed (see SE Section 2.8.7.2.2.3.2 on LHGR Limit). [[ ]] In Reference 35 (response to RAI SRXB-A-65), the licensee concluded that, therefore, no additional conservatism in local exposure is required to maintain fuel integrity.

At the EPU conditions, the plant can operate with the limiting bundles (or rods) operating at the LHGR limit, depending on the initial core design and the control rod patterns employed by the licensee. The VYNPS lattice calculations show that for nodes depleting at high void fraction conditions, the pin power peaks with exposure. In addition, more fission gas is released during the second and third fuel cycle, than in the first cycle. Top-peaked power shapes can be attained, through control rod pattern and depletion. Depending on the control rod pattern employed, the potential exists that the highly exposed fuel could experience high bundle powers, thereby accumulating higher exposure for the proposed operating strategy. However, the licensee demonstrated that [[ ]] Therefore, the NRC staff agrees that [[ ]] In addition, the exposure limit (e.g., peak pellet exposure limit of [[ ]] is an NRC-approved limit for each vendor's fuel, and therefore the licensee is obligated to ensure that the plant is operated and the cycle-specific core designed in a manner that the fuel exposure does not exceed this limit.

Considering the margins demonstrated in the internal pressures calculation, the fact that the exposure limit is an NRC-approved limit, and [[ ]] the NRC staff accepts that no additional margins need to be included in the operating kW/ft. Therefore, the staff concludes that there is reasonable assurance that the [[ ]] peak pellet exposure limit will not be exceeded.

2.8.7.2.2.3.4 MAPLHGR

MAPLHGR is a LOCA fuel design limit that minimizes gross fuel failures due to the severe cladding heat-up or fuel fragmentation as result of the quenching of the ECCS flow. The

amount of stored energy in the fuel is proportional to the average kW/ft in each node (bundle-wise) before the scram. Gross cladding failure is prevented by limiting the power level which would result in a PCT of 2200°F during a DBA LOCA. The LOCA MAPLHGR is obtained by averaging the LHGR over each fuel rod in a given plane of a particular fuel bundle and selecting a limiting value as a function of fuel burnup. The PCT is considered to be a function of the average kW/ft of all the rods in a bundle at a given axial location. Amendment 19 to the GE Standard Application for Reactor Fuels (GESTAR) approved selecting the MAPLHGR limit based on (1) the LOCA PCT being below 2200°F and (2) not exceeding the maximum LHGR.

Similar to the LHGR, the local pin power peaking and the bundle powers factor in the generation of the MAPHGR limit. The licensee's Alternative Approach discussed the conservatisms in the development of the MAPLHGR that would compensate for potential inaccuracies in the pin and bundle powers with depletion. The conservatisms cited are discussed below.

In the SAFER/GESTR methodology, the hot bundle is initialized assuming a hot rod at the LHGR limit and the average rod at the MAPLHGR limit. In the Appendix K calculation, a 2% core thermal power uncertainty is applied to the hot rod. Note that for those plants that implemented improved feedwater measurement instrumentation (i.e., for measurement uncertainty power uprate purposes), a lower power uncertainty is applied. However, VYNPS did not implement a measurement uncertainty power uprate, and the full 2% power uncertainty was applied for the VYNPS Appendix K calculation.

Since total bundle power is important to ECCS-LOCA response, the SAFER/GESTR methodology maximizes the initial operating power of the hot bundle. In an iterative calculation assuming different OLMCPR and a low R-factor, the power peaking is maximized. The R-factor represents the influence of the rod pattern peaking on the critical power. An ECCS-LOCA analysis is not performed every reload, but only during new fuel introduction or if new operating conditions are implemented (e.g., higher operating domains). Therefore, the hot bundle operating power is maximized such that the ECCS-LOCA OLMCPR bounds the OLMCPR calculated from the limiting cycle-and core-specific AOO analyses.

To ensure that the ECCS-LOCA results are bounding, the hot rod power peaking is minimized so that the average power (average planar linear heat generation rate, APLHGR) is maximized.

The Appendix K PCT calculations include the conservative Appendix K modeling assumption. In calculating the upper bound SAFER/GESTR PCT, the nominal PCT is adjusted to account for model uncertainties (at 95% probability). The 95% probability PCT includes a 2.2 σ [[ ]] applied to the LHGR. Based on the Appendix K modeling assumptions, the VYNPS PCT is 1960°F.

Review of core follow data of EPU plants showed that the axial bundle power and nodal power uncertainties increase with high bundle power/flow ratio characteristic of operation with high powered bundles and a flatter core design. In Reference 31, the licensee responses to NRC staff RAIs SRXB-A-29, 30, and 58, indicate that no axial power uncertainties are added to the

calculations of the MAPLHGR. In addition, the responses also state that ECCS-LOCA response is not too sensitive to the power profile and the mid-peaked power shape results in a more limiting PCT. However, the NRC staff was not convinced that uncertainties should not be applied to the axial power profile to account for higher uncertainties. In addition, the staff believed that the ECCS-LOCA is not highly sensitive to the axial power distribution, but could make a difference if the plant is MAPLHGR-limited.

To address this concern, the NRC staff performed confirmatory LOCA analysis that did show sensitivity to the axial power profile, with the top-peak profile being limiting. The staff's confirmatory ECCS-LOCA analyses, which assumed a top peaked power profile, resulted in a PCT less than 2200°F. Therefore, the staff accepts the VYNPS MAPLHGR calculation based on the current SAFER/GESTAR methodology. The staff accepts the current MAPLHGR calculation methods, which does not include any axial power uncertainties, because (1) there are conservatisms in the ECCS-LOCA calculations, as cited, (2) the staff's confirmatory analyses, based on a different power profile, resulted in a PCT below the limit, and (3) the sensitivity of the ECCS-LOCA calculation to the axial power profile and the need for axial power uncertainty is being addressed generically.

2.8.7.2.2.3.5 Shutdown Margin, Standby Liquid Control and Rod Withdrawal Error

Shutdown Margin Demonstration

The shutdown margin (SDM) is typically defined as the amount of reactivity by which the reactor is subcritical or would be subcritical assuming: (1) all control rods are fully inserted, except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn; (2) the reactor is xenon free; and (3) the moderator temperature is 68°F.

Since the core reactivity is greatly influenced by the isotopic composition of the exposed fuel, the approach for ensuring sufficient SDM was investigated. The plutonium isotopic content, in particular, is sensitive to the void content during depletion ("void history"), which may be increased for uprated conditions.

The VYNPS TSs require that an SDM of greater than or equal to 0.38% $\Delta k/k$ be maintained throughout the cycle when the highest worth control rod is determined analytically (by calculation rather than by direct measurement). It is standard practice for GE to increase this value to 1% $\Delta k/k$ for design purposes to account for manufacturing tolerances, changes in operation from planned conditions, control rod worth reduction due to depletion, methodology approximation, inexact tracking of actual plant parameters, and other identified factors. VYNPS adds an additional 0.1% $\Delta k/k$ to account for a potential SDM loss from inverted B_4C control rod tubes in their control blades. Therefore, while the TSs require a margin of 0.38%, GE and VYNPS design to a SDM value of 1.1%.

The highest core reactivity point occurs at cold (nominally 68°F) and xenon-free conditions, frequently, but not necessarily at beginning of cycle (BOC). The SDM is demonstrated for each cycle by performing an in-sequence measurement of the cold critical state. Since the

measured state may be at slightly higher temperatures and for a slightly super-critical state, temperature and period corrections are made to determine the SDM. An additional correction is applied if the most reactivite point in the cycle is determined to be other than BOC. The following equation shows the calculation of the SDM:

$$SDM = k_{crit} - k_{sro} - R + \Delta k_{temp} - \Delta k_{per}$$

Where:

k_{crit} is the multiplication factor for the critical rod pattern;

k_{sro} is the multiplication factor for the strongest rod out;

R is the maximum decrease in SDM throughout the cycle;

Δk_{temp} is the temperature correction; and

Δk_{per} is the period correction.

The period and temperature corrections are determined by pre-computed tables and are confirmed by calculations at the temperature and rod-pattern positions corresponding to the critical configuration measured.

This equation can also be written as:

$$SDM = \Delta k_{crp} - \Delta k_{sro} - R + \Delta k_{temp} - \Delta k_{per}$$

Where:

Δk_{crp} is the difference in multiplication factor from the critical state to all rods in state ($= k - k_{ari}$); and

Δk_{sro} is the difference in multiplication factors from the all rods in state to strongest rod out state ($= k_{ari} - k_{sro}$).

Written in this form, the equation shows that the biases that occur in the calculation of multiplication factors for the critical configuration and all-rods-in configuration cancel out since these configurations both represent "distributed" critical states that generally have similar calculational biases. The biases for the change in multiplication factor for the strongest-rod-out case do not cancel out since this represents a difference in the multiplication factor for a "distributed" critical state (all rods in) and a "local" critical state (strongest rod out). Per GE Technical Design Procedure, an appropriately conservative value of the bias is assumed for the strongest-rod-out configuration, which is typically computed as the distributed bias along with an additional strongest-rod-out bias. Therefore, biases are included in the strongest-rod-out reactivity effect. In general, the differences in multiplication factors can be computed more

accurately than absolute multiplication because the biases discussed subtract out, resulting in an accurate value for the SDM.

Cold Critical Eigenvalue

The cold critical eigenvalue predictions were reviewed for VPNPS cycles 23 and 24. The results are shown in Table 2.8.7-2. The cycle 24 core was designed for EPU operation and therefore, provides a direct indication of the predictions for the EPU designs. The BOC difference in cold critical eigenvalue for cycle 24 is 0.07%. The results in this table show good agreement between the predicted and critical eigenvalues. These cold critical eigenvalue predictions can be compared to those for several plants presented in Figure 2.8.7-6. This figure shows the variation in the prediction of the cold critical eigenvalue which results in bias and uncertainty values. However, as can be observed, the deviation of the cold critical eigenvalue from unity in some cases is significant. This difference from unity indicates a bias in the calculational model resulting from unidentified sources. While the GE procedure for determining a design eigenvalue includes an eigenvalue trend line to account for this recurring bias, significant deviation from unity indicate that there is a significant unknown reactivity effect that is not accounted for in the model. VYNPS, however, has nearly no bias at the beginning of cycle as shown in Table 2.8.7-2, and therefore the GE methods appear to perform well in predicting the absolute criticality (without bias).

Table 2.8.7-2
Comparison of Predicted and Critical Eigenvalues for VYNPS Cycles 23 and 24
(Note - Cycle 23 BOC not evaluated with Cycle 23)

Cycle	Cycle Exposure (MWd/ST)	Predicted Eigenvalue	Critical Eigenvalue	Difference (Dk)
23	BOC	-	1.0006	-
	7417	0.9976	0.9957	-0.0019
24	BOC	1.0000	1.0007	+0.0007
	961	0.9996	0.9972	-0.0024

The In-Sequence Critical/SDM Worksheet for the startup of cycle 24 of VYNPS was reviewed and the demonstrated SDM was shown to be 1.291%, which significantly exceeds the TS requirement of 0.38% as well as the VYNPS-imposed requirement (1.1%).

After a review of the procedures and results for predicting the SDM and the cold critical eigenvalue, it was determined that the biases and uncertainties in the predictions for VYNPS are relatively small and well known. The resulting biases and predictive capabilities are similar for an uprated core design (cycle 24) as for previous core designs (e.g., cycle 23). In addition, the methodology for determining SDM is such that the biases in the calculational procedures cancel out, providing a direct indication of the actual SDM, rather than predicted values. The demonstrated SDM for cycle 24 shows considerable margin to the TS limit and meets the

VYNPS-imposed requirement as well. Therefore, it was found that the ability to demonstrate the SDM is not impacted by the proposed VYNPS EPU.

Standby Liquid Control System (SLCS) Calculation

In addition to the SDM, the potential impact of high-void operation on the Standby Liquid Control System (SLCS) was also investigated. A cycle-specific SLCS calculation is performed to assure that the reactor will remain sub-critical in the most reactivity condition when the TS minimum soluble boron is introduced into the core. Based on the analysis of the SDM calculation, described above, the biases and uncertainties for the EPU core designs are similar to those of the non-EPU core designs. In addition, the GESTAR methodology makes conservative assumptions that no credit be given for the minimum control rod inventory at the exposure condition being analyzed, which can amount to 1% at BOC where the SLCS is most frequently limiting. Therefore, the SLCS calculational procedure remains applicable to VYNPS and provides sufficient margin to ensure the shutdown of the reactor.

Impact on Rod Withdrawal Error (RWE) Transient

The RWE transient is driven by the worth of the withdrawn control blade. Errors in the prediction of the worth of the control blade can potentially have an impact on the transient resulting from the withdrawal of that control blade. The total reactivity worth of the withdrawn rod was compared to modeling errors. The potential errors in the prediction of the rod worth were found to be small in comparison to the overall worth of the rod, resulting in little impact on the transient response. The impact is further minimized by the fact that the control rod will be adjacent to both low and high exposure bundles, with the error for the lower exposures being smaller. And finally, the errors appearing as biases in calculated eigenvalues will subtract out since they occur in both the controlled and uncontrolled conditions. For these reasons, potential impacts on the RWE are minimal for VYNPS.

2.8.7.2.2.3.6 Void Coefficient

Void Reactivity Coefficient Evaluation

The reactor core response during transient situations is highly dependent upon the changes in reactivity with changes in void content in the coolant. The impact is measured by computing a void reactivity coefficient, which is defined as follows:

$$\text{Coolant Void Reactivity Coefficient} = 1/k\,(dk/d\alpha)$$

In the above equation, k is the multiplication factor and α is the void fraction. Since a derivative is involved, the coolant void reactivity coefficient is essentially proportional to the slope of the k versus void fraction curve. A different shape in the Kinf vs. void fraction for a particular lattice, Kinf, would be expected for lattices that have been depleted with different historical void fractions because they will have a different plutonium content. Therefore, it is expected that the

void reactivity coefficient will have some variation with the historical void fraction used in the lattice depletion.

The transient analysis code ODYN performs a one-dimensional analysis of the core using cross sections that are determined via GE's cross section model. This approach involves using the TGBLA lattice physics code to perform depletion calculations for a particular lattice at 0%, 40%, and 70% void fraction. These "historical" cases are important for capturing the change in the isotopic composition of the fuel with exposure and void fraction. In addition, instantaneous changes in the void fraction are performed for the 40% void history case to capture the change in the cross sections for the instantaneous void changes that would occur in transients. The GE methodology assumes that the change in the cross sections with respect to changes in the instantaneous void fraction for the 0% and 70% void history cases is the same as the changes for the 40% void history case. In addition quadratic interpolation/extrapolation is used to determine cross sections at other instantaneous and historical void fractions (the fitting procedure actually uses water density as the independent variable).

Therefore, while ODYN does not specifically use a void reactivity coefficient, the sensitivity study/uncertainty studies performed with ODYN consider a specific value for the uncertainty in the void reactivity coefficient. For example, the analysis performed in response to NRC staff RAI SRXB-A-68, as shown in Reference 36, assumes a [[ ]] uncertainty. However, given that the void reactivity coefficient is sensitive to the plutonium content and that specific instantaneous calculations are not performed for the 0% and 70% branches, an investigation of the potential error in the void reactivity coefficient is warranted. In addition, errors related to the extrapolation beyond the 70% historical void fraction were evaluated.

To assess the potential impact of the 40% historical assumption on the void reactivity, the NRC staff requested ORNL to perform confirmatory analyses. ORNL performed confirmatory analyses using the HELIOS code system. The calculations were performed for a lattice with high enrichment, gadolinium loading, and with vanished rods typical of the upper portion of the fuel bundles. The calculations were performed for bundle exposures up to 60 GWd/t and for instantaneous and historical void fractions ranging from 0 to 90%.

Comparisons of the void reactivity coefficients for the different void history cases and exposures are presented in Figures 2.8.7-7 through 11. Note that additional instantaneous branch points were calculated at the higher void conditions to resolve the non-linearity of the void coefficient with respect to instantaneous void fraction at high burnups. The impacts of the different void histories are evident at the higher void fractions with the increasing differences in the void reactivity profiles with increasing exposure (see Figure 2.8.7-11, for example). The difference in the profiles from that at a void history of 40% represents a potential error resulting from GE's methodology. In addition, the deviation of the void reactivity from a linear variation represents an error that results from the quadratic extrapolation used in GE's methodology. As shown in these figures, at high void fractions and high exposures the profile deviates from linear.

An evaluation of the potential errors in GE's methodology has been performed evaluating the void reactivity coefficients obtained using the GE approach (quadratic fitting and instantaneous

void branches only at 40% void history) against void reactivity coefficients calculated at different void histories with exposure. A plot showing the potential errors is presented in Figures 2.8.7-7 through 2.8.7-11. These figures show that the fitting process results in increasing errors with exposure and burnup (as previously discussed, based on the void reactivity coefficient plots). The errors for exposures of about 30 GWd/t and less are consistent with the [[ ]] uncertainty assumed in the ODYN analysis. However, for higher exposures at high void fractions the error increases to more than 30%.

The GE approach to cross-section parameterization and fitting (quadratic fitting with values at 0%, 40%, and 70% void fraction) combined with the assumption that the change in cross sections with instantaneous change in voids is not sensitive to the void history, results in a substantially larger error in the void reactivity coefficient than assumed in the ODYN uncertainty analysis, particularly at high fuel exposures. The NRC staff determined that the impact of these increased errors on the response of the core during a transient needs to be evaluated to ensure that there is no impact on the core response. GE's position is that the errors at the higher exposures are not significant because the power generation in those bundles would be low. However, criteria applied to the fuel are also exposure dependent (e.g., LHGR), and therefore it is important to provide a demonstration that the fuel integrity is not compromised in the event of a transient event.

GE's Evaluation of Increased Void Coefficient Uncertainty

In response to NRC staff RAI-SRXB-A-68 (Reference 36), GE performed an evaluation of the errors in the void coefficient resulting from the cross section model as described above. The model assumptions that have a significant impact on the void coefficient are:

- The assumption that the cross sections can be parameterized with respect to void history using a quadratic fit to the 0%, 40%, and 70% instantaneous void fraction values with extrapolation to higher instantaneous void fractions. This results in a linear variation in coolant void reactivity with respect to void fraction, whereas the results show a significant deviation from the linear at high void fractions.

- The assumption that the void reactivity coefficient determined at a 40% void history condition applies to all other void histories. At high exposures the difference in isotopic compositions resulting from differing void histories results in significantly different void reactivity coefficients.

In the RAI response, GE considered the cross section model impacts separately for exposures less than 25 GWd/ST and greater than 25 GWd/ST, up to 65 GWd/ST. The calculation and comparison of the void coefficients at exposures of less than 25 GWd/ST indicated that the void reactivity coefficient errors were within those assumed in the ODYN ΔCPR/initial critical power ratio (ICPR) uncertainty analysis (see Figure 2.8.7-17). The results for exposures greater than 25 GWd/ST are shown in Figure 2.8.7-18 as a ratio of the MCNP to TGBLA06 void coefficients, and are quite similar to the confirmatory results discussed above. These relatively large

differences therefore required additional analysis to ensure that they do not result in a significant impact on safety parameters.

GE considered two transients (generator load rejection without bypass and MSIV closure with flux scram) for analysis assuming a void coefficient bias with exposure as shown in Figure 2.8.7-18. [[

]] The changes in the other parameters (thermal over-power, mechanical over-power, and peak pressure) showed relatively small differences in comparison to the available margin.

The cross section model utilized in GE's methodology introduces a relative large bias in the void reactivity coefficient at high void and exposure conditions. GE performed specific TRACG transient calculations incorporating this bias to determine the impact of the increased bias on the transient behavior for two particular transients. The calculational results indicate that the increased bias at high exposure does have an effect on the transient response, but the magnitude of the response is sufficiently small such that the impact is not significant for VYNPS.

2.8.7.2.2.3.7 Code-to-Code Comparisons

Recent gamma-scan measurements or isotopic assays have not been performed. In order to determine the fidelity of the results from GE's methods, the only remaining method is to perform a comparison with other codes that are of higher fidelity (e.g., the MCNP code) or with other codes that have been validated. Code-to-code comparisons were included in Reference 69, Enclosure 3, comparing TGBLA to MCNP for particular lattices. Additional code-to-code comparisons were included in the licensee's RAI SRXB-66 response (Reference 33) comparing CASMO and TGBLA. For this evaluation, comparisons were performed for representative lattices and comparisons of Kinf, peak pin power, plutonium isotopic inventory, void reactivity coefficient, and cross sections were compared.

Kinf Comparisons

Regarding Kinf, the results agree very well at high exposures. The differences are larger in the 0 – 20 GWd/ST burnup range, which is when the gadolinium is burning out. The burnout of Gd, particularly at high void fractions, results in differences between TGBLA-6 and CASMO-4. The differences are smaller for void fractions of 40% and 70%, which represent the average void fraction value for a core. The differences for the results at high void fractions will not have a significant impact on the overall core results because the contribution of power in these regions will be relatively small.

Figures 2.8.7-19 and 2.8.7-20 provides two Kinf vs. exposure curves for historical void fractions of 40% and 70%. The calculations were performed with an instantaneous void fraction of 0%. Therefore, the difference at high exposure is caused by differing isotopic compositions resulting from the depletion at different void histories. As these figures show, the impact of void history at high exposures is significant (greater than 5% delta k). Confirmatory analyses performed with HELIOS, presented in Figure 2.8.7-21, show a similar trend and include data for a historical void fraction of 90%.

Pin Power Distribution Comparisons

The pin peaking factor (Figure 2.8.7-22) shows good agreement at zero exposure, when the power peaking is the most significant. Differences increase with exposure resulting in several percent at burnups of 60 GWd/ST. Generally, the TGBLA-6 local peaking factors are larger than CASMO-4 and therefore will be more conservative in regards to pin exposure and LHGR. Note that at high exposures (greater than 30 GWd/ST), the power peaking is the largest for bundles with the highest void history. However, given that the power generation in this highly burned fuel is low, the net result is that the overall LHGR (a product of the region average power and the local power peaking) will be lower than that earlier in the exposure history, where the differences are smaller. However, the LHGR limit is lower at higher fuel exposures. Therefore, if spectral shift strategy is employed, as proposed in the expanded operating domains for EPU plants, with the upper nodes depleting at high void conditions and a top-peaked power distribution, both the bundle power and the pin power peaking would be high, at the most limiting kW/ft exposures.

Figure 2.8.7-23 shows the root-mean-square (RMS) of the differences between TGBLA-6 and CASMO-4. The RMS error is used in development of the SLMCPR. Previous analysis comparing TGBLA-6 and MCNP (Reference 69, Enclosure 3) had shown that for a variety of lattices and exposures, that the RMS difference is [[ ]] However, in these comparisons the isotopic concentrations were taken from TGBLA and used in MCNP and, therefore, errors in depletion were not included. The results in this figure show that on a code-to-code basis that, for the lower void fractions, the RMS difference at low exposures and at high void fraction exceeds the [[ ]] RMS value, with maximum differences of about 2.5%. For the burnups above the gadolinium burnout and for lower void fractions, the differences are consistent with the [[ ]] value. Note that the results for the 90% void fraction presented in Figure 2.8.7-23 does not include potential errors caused by the quadratic fit and extrapolation used in GE's

neutronic methods. The results presented for TGBLA-6 are also for the corrected version of the code, as discussed below.

The comparison of TGBLA-6 and CASMO-4 RMS power distribution differences indicates in some situations that the RMS difference is larger than the [[ ]] value determined previously. Therefore, this supports the need for increasing the pin power peaking uncertainty to [[ ]] as was done for the SLMCPR.

Plutonium Isotopic Comparisons

The plutonium compositions were compared between the two codes as an indication of the prediction of the neutron spectrum and as a further investigation of potential impacts on the void reactivity coefficient and SDM. The comparison of the Pu-239, Pu-240 and Pu-241 isotopic compositions (Figures 2.8.7-25 to 27) show very good agreement, particularly at the lower void fractions. Notably, Pu-239, the primary fissile plutonium species present, has very close agreement between the two codes.

In its review of GE's neutronics methods, the NRC staff had concern over the accurate prediction of gadolinium burnout in high void conditions, and in general the performance of TGBLA depletion capabilities at high void fractions. In response to NRC staff RAI 3-1a in Reference 72, GE described an error discovered in the TGBLA-6 code in the [[

]] In response to NRC staff RAI SRXB-A-67(e) in Reference 35, GE provided a comparison for the eigenvalue tracking results that indicate that the impact at the core level is less than [[ ]] The impact on SDM was similar, but with the error generally having an insignificant impact on SDM prediction, which is the difference of two eigenvalues. The results included in the comparisons are based on TGBLA-6 with this error corrected.

Void Reactivity Comparisons

The void reactivity coefficient is an important parameter for predicting the transient response and stability of the reactor core. Comparisons of TGBLA-6 and CASMO-4 values of the void reactivity coefficient were compared in the RAI 66 response and are provided in Figure 2.8.7-24. The comparisons show that the agreement between the two codes is within the 15% assumed in the ODYN perturbation analysis. More significant is the large difference in the value of the void reactivity coefficient for the two different void histories used in the calculations (40% and 70%). The TGBLA/PANACEA neutronics methodology computes such instantaneous effects only for the 40% void history case. This assumption can lead to large errors in the evaluation of the void reactivity coefficient at high void fractions and high exposures. An evaluation of the impact of these errors was provided earlier.

Cross Sections and Lattice Physics Parameters

In NRC staff RAI SRXB-A-66, GE and Entergy were asked to provide comparisons of the cross section data between TGBLA06 and CASMO-4. The response to the RAI in Reference 33 stated that it was difficult to perform comparisons of cross section data because of differences in methods resulting in different definitions as well as differing energy group boundaries. In Reference 69 an assessment was performed in which three group cross sections computed with TGBLA were compared to values computed by MCNP. The purpose of these comparisons was to determine the potential error in the cross sections at high void fractions caused by the quadratic cross section fitting with respect to instantaneous void fraction. Similar calculations were performed at ORNL to confirm the results obtained by GE. It was found that the quadratic fitting process resulted in errors in some of the cross section and lattice physics parameters at high void fraction (particularly the flux ratios and migration area, which are parameters used in PANACEA). The impact of the fitting errors on the void reactivity coefficient however was subsequently considered, as discussed above. Note that while some of the cross sections do exhibit errors in the extrapolation process, it was found that Kinf was a parameter that did not exhibit significant extrapolation errors.

Conclusions

The code-to-code comparisons provide reasonable assurance that the TGBLA-6 neutronic methods are reasonable for EPU conditions for VYNPS based on the information and analysis provided by the licensee. The differences in the results from different codes indicate that the primary source of concern is in not considering the impact of the void history on parameters involving instantaneous changes in void fraction (such as void reactivity coefficient). Also considered is the impact of quadratic fitting and extrapolation to higher void fractions on cross sections and lattice physics parameters. The impact was evaluated for key parameters, such as void reactivity coefficient and Kinf.

The errors and uncertainties for plants with other fuel designs and operating conditions may be different than those included in the VYNPS evaluations. As discussed in Reference 70, GE plans to perform gamma scan measurements to confirm that the assumptions used in the neutronic methods for current GE fuel designs are still appropriate.

2.8.7.2.2.3.8 Bypass Voiding Impact on the Neutronic Methods

For a bundle with a relatively high void content (near 90% void fraction), the source of thermal neutrons that drive the fission process are predominately from outside the fuel region because of the limited moderation occurring within the channel. The bypass region, which typically contains water in the liquid phase, has a strong influence on the power distribution. However, increases in the core power density and higher in-channel void conditions increase the likelihood that there will be voiding in the bypass region and the water rod. Therefore, an evaluation was performed to determine if voiding in the bypass region and the water rod has an impact on the power distribution within the bundle.

GE has indicated that bypass voiding is unlikely, and it assumes that no voiding occurs in the bypass region. An analysis, however, was performed by GE to assess the potential error in the case of 5% and 10% bypass voiding that is not predicted by its analysis codes (e.g., PANACEA). Their analysis investigated the impact on Kinf and found a reduction in Kinf with bypass voiding for the bundle affected. By neglecting the bypass voiding, the Kinf and therefore the power in the higher void regions is over-predicted, which is conservative. However, the power in the other regions of the core will be under-predicted. This increase in power in the other regions of the core will be distributed and therefore will generally not result in a significant change in the power in those regions.

Since the GE analysis did not include evaluations of the potential impact of bypass and water rod voiding on the pin power distributions, the NRC staff independently investigated the impact of bypass and water rod voiding on the lattice physics data. The calculations were performed at ORNL, using MCNP to determine the impact of 10% and 30% voiding in the bypass region to represent the maximum possible voiding that could occur in operational and transient conditions. The calculation was performed at zero exposure. The results correspond to a high enrichment, highly gadolinium loaded GE-14 lattice with zero exposure, which represents the case with the most severe power peaking. A comparison of the Kinf and power distribution for bypass voiding of 0%, 10%, and 30% was performed with the results being presented in Table 2.8.7-3.

Table 2.8.7-3
Impact of Bypass Voiding on Lattice Kinf and Pin Power Distribution

Bypass Voiding	Lattice Kinf	RMS of Difference of Relative Power Distribution from Unity	Peak Pin Power from 0% Voiding
0%	0.9658	0.33	1.41
10%	0.9592	0.32	1.40
30%	0.9427	0.29	1.37

The results indicate that the power generation in the bundle is shifted away from the bypass region to the interior region of the bundle as voiding occurs. This is clearly demonstrated by plotting a difference of the power distribution between the 0% and 30% voiding cases, as is shown in Figure 2.8.7-28. The largest reduction in the pin power occurs in the corner pins, which have two faces adjacent to the bypass region. This results in a reduction of the power peaking factor since the peak pin is located near the bypass region. The reduction in the RMS of the pin power difference from the unity results presented in Table 2.8.7-3 show that the power profile becomes more uniform ("flatter") as the bypass region voids (an RMS of zero would indicate a flat power profile, while a large value indicates a high-degree of peaking). The results also show that the relative power of the peak pin also decreases (the peak does remain in the same pin for all bypass void conditions). The reduction in the lattice Kinf value indicates

that the overall bundle power will also decrease as the bypass region voids, as also indicated by the GE analysis.

Therefore, based on this evaluation, bypass voiding results in a reduction of the power generation in the fuel bundle, as determined by the reduction in the lattice Kinf. The pin power distribution changes such that power production is shifted from the bundle periphery to the interior region. This generally results in a reduction in the power generation in the peak pin in the bundle and a more uniform power distribution within the bundle. The power generation in the interior pins increase, but these increased values are generally less than the peak bundle value. The overall result is a reduction in the peak pin LHGR in comparison to the assumption that no bypass voiding occurs.

The impacts of the bypass and water rod voiding on the radial power distribution (within bundle and across bundles) affect the assumptions in the thermal limits calculations (e.g., R-factor and SLMCPR calculation) during steady-state and transient conditions. However, at natural recirculation conditions where bypass voiding may occur, the bundle power levels are expected not to be high enough to pose thermal limits concerns, without the presence of instability. Section 2.8.7.4 of this SE evaluates the impact of bypass voiding on the reliability of the neutron monitoring system to provide instability protection in the event that inadvertent plant maneuvers or transients place the reactor at the high power/low flow natural recirculation condition.

In conclusion, the NRC staff performed independent confirmatory analyses to assess the potential impact of bypass and water rod voiding on the pin power peaking and Kinf. Based on the staff's confirmatory analyses and review of GE's evaluation of the impact of bypass voiding, the staff finds that there is reasonable assurance that the lattice physics calculations assuming no bypass and water rod voiding would not significantly affect the predictions of the core conditions at natural recirculation conditions.

2.8.7.3 Applicability of the Thermal-Hydraulic Correlations

In general, correlations are developed from test data, that cover specific ranges of thermal-hydraulic conditions. An independent set of test data is used to validate the performance of the correlations and establish the correlation uncertainties. The key parameters that define the correlation (e.g, thermal-hydraulic and geometric parameters) that the test data is based on establish the "range of applicability." Changes in these key parameters could affect the accuracy of the correlation, requiring further evaluation of the performance of the correlation under the condition it is being applied. The NRC approval of licensing methodology requires application of the correlations within the ranges it was developed, validated, and approved. Any changes in the correlation's key "dependence" parameters require further test data to establish the correlation's accuracy to the conditions it is being applied to, as is the case for the critical power ratio (CPR) correlation (GEXL). For new fuel design, involving changes in the fuel thermal-mechanical design, GE uses new test data to model the features of the new fuel design and to develop a modified GEXL correlation applicable to the new design for the thermal-hydraulic conditions to which it would be applied.

The review in this section entails confirming that for the changes in core thermal-hydraulic conditions of bundles (e.g, higher bundle power/flow ratios) and new fuel designs (10x10 fuel designs, with two large water rods and 14 part-length rods), the correlations are being used within the approved applicability ranges for normal steady-state and transient conditions.

Figures 2.8.7-2, 2.8.7-3, and 2.8.7-4 show the VYNPS maximum powered bundles thermal-hydraulic conditions with cycle exposure relative to existing EPU plants and a high density operating plant. Note that plant C is a foreign plant that is uprated and is operated with a different operating strategy.

GEXL Correlation

A new GEXL correlation is developed for each new fuel design. In NRC staff RAI SRXB-A-50, the NRC staff asked Entergy to confirm that during transient and accident conditions, the GEXL critical power correlation correlations (GE-14) would fall within the NRC-approved applicability ranges. GEXL is a steady-state correlation, but had also been approved for transient conditions.

The licensee stated in Reference 31 that the critical power data were obtained for bundle mass fluxes ranging from [[ ]] inlet subcooling [[ ]] and pressures from [[ ]] The data cover the flow ranges from less than natural circulation to well beyond rated flow. Since, during the test, the power levels are increased up to the critical power for each flow, the data include void fractions up to [[ ]] Boiling transition would occur for void fractions of approximately 95%. These parameter ranges also cover the expected ranges for LOCA and transient events. For LOCA calculations, the GEXL correlation is used for the calculation of the early boiling transition during the flow coast down immediately following the break. This typically occurs when the flow has dropped to 30-50% of the initial value. This is well within the application range for the GEXL correlation.

Figure 2.8.7-29 shows the GE-14 application range and the expected ranges for typical operational transients. This figure demonstrates that critical power correlation test ranges cover the EPU conditions, and GEXL is being applied within the NRC-approved process as described in GESTAR II. The NRC staff finds this acceptable.

2.8.7.4 Impact of Bypass Voiding on the Reliability of Neutron Monitoring System

Stability Option I-D

The VYNPS stability solution is discussed in Section 2.8.3 of this SE. The following discussion focuses on the impact of potential bypass voiding on the reliability of the neutron monitoring system and on the effectiveness of the Option I-D stability option. Instability is considered an AOO and the SLMCPR must be protected from being violated during the event.

A two recirculation pump trip (RPT) would place the reactor in the high power/low flow operating regions susceptible to instability. For the current core designs and operating strategies, the high initial in-channel void fraction in the maximum powered bundle could result in bypass voiding potentially greater than 30%. Bypass voiding would affect the response of neutron monitoring system (TIPs and LPRMs) relied upon for both core monitoring during the event and to scram the core in the onset of power oscillation. The Option I-D instability protection consists of two features (1) prevention; and (2) detection and suppression. An administratively controlled buffer and exclusion region prevents instability by defining and restricting operation in the regions susceptible to power oscillation (i.e., prevention). In the event a transient results in a power oscillation, a flow-biased APRM scram occurs before the oscillation magnitudes increase to a level that would threaten SLMCPR protection (i.e., detection and suppression).

The licensee's response to NRC staff RAI SRXB-A-55 in Reference 31 addressed the impact of bypass voiding on the reliability of the instrumentation and the effectiveness of the Option I-D flow-biased scram. Voiding in the bypass region where the neutron monitoring system is located reduces the detector response, although the same power is being generated in the adjacent fuel bundle. The lower moderation due to bypass voiding decreases the incident thermal flux on the detector, thus reducing the instrument reading for the same neutron flux generated in the adjacent fuel. The impact on the gamma TIPs would be less since, for a given bypass void level, the gamma flux is not perturbed as much as the thermal neutron flux. The bypass voiding could also increase the noise in the neutron flux signal. Bypass voiding could potentially reduce the cooling of the instrumentation and increase the LPRM temperature. The following four items summarize the licensee's evaluation provided in Reference 31.

1. Impact of Bypass Voiding on LPRMs and TIPs

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   The impact of the bypass voids on the APRMs can be determined from the combination of 4 LPRMs elevations (A, B, C, and D) and each of the LPRM strings in the APRM. Using this approach, the licensee sought to determine the impact of the reduced LPRM detection on the APRM scram capability.

2. Impact of High Bypass Voiding on VYNPS Option I-D Stability Solution.

   The analytical limit for the APRM flow-biased flux scram setpoint should provide assurance that the scram would occur before the magnitude of the power oscillation gets large enough that the change in the MCPR results in operation of the plant below the SLMCPR. High bypass voiding can potentially reduce the APRM reading, delaying the scram. This is non-conservative, because the oscillation magnitude would grow and the margin to the SLMCPR would be reduced.

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3. Bases for the VYNPS Stability Setpoint Analytical Limit Validity

   The licensee evaluated the impact a [[ ]] of the average reactor power due to bypass voiding may have on the scram analytical limit. The flow-biased APRM flux trip analytical limit at natural recirculation is 53.7% of rated power. Based on best estimate calculation, the power level at the rated rod line at natural recirculation is 49.4% of rated power. To account for [[ ]] of the average reactor power, the licensee used [[ ]] to demonstrate that the current conservatism in the analytical limit would compensate for the [[ ]] in the reactor power at natural recirculation due to the bypass voiding.

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4. Impact of Noise Due to Bypass Voiding

The licensee's RAI response stated that increased voiding in the bypass region could potentially increase the LPRM noise, because of the bubbles going by the LPRM instrumentation assembly in the water gaps. In general, a normal APRM noise of approximately 2% is present, due to flow-induced vibration of the LPRM assembly in the water gap and other thermal-hydraulic phenomena inside the channel. The extra noise caused by the bypass bubbles increases the overall APRM noise at off-rated conditions. The licensee concluded that the impact of the noise on the APRM scram setpoint is negligible because the setpoint is based on the normal (non-void) noise at rated conditions and this bounds the increased noise at the off-rated conditions. The reason for this is that the normal noise decreases at the off-rated conditions, which compensates for the increase in the noise due to the bypass voiding. The setpoint is derived from the analytical limit by considering noise and other instrument errors.

The licensee evaluated the impact of bypass voiding on the neutron monitoring system and its impact on the Option I-D detect and suppress capability. The APRM's reading, which averages LPRM readings, led to a [[ ]] in the core average void at natural recirculation. In determining the decrease in the flux at the LPRM, the calculation assumed [[ ]] The licensee's response in Reference 31 to NRC staff RAI SRXB-A-55 stated that [[ ]] However, the NRC staff expects in-channel voids in the high 80% range to 90% would have been more appropriate and characteristic of the in-channel voids, especially for the hot channels. Therefore, the calculation may not have maximized the potential bypass voiding. As discussed previously, high bypass voids can potentially reduce the APRM reading, and so the margin to scram would increase and this could be non-conservative with respect to stability mitigation.

The NRC staff reviewed the licensee's assessment of the potential impact resulting from increased voiding in the bypass channel on the plant response to potential instability events. Of particular staff interest was whether the plant scram response would be affected by operation at a higher void fraction. The licensee performed a sensitivity assessment of the plant response to operation at a higher resident void condition and concluded that for the current cycle OLMCPR, the plant would have retained margin to the safety limit MCPR so that the safety limit would not be violated. The fact that VYNPS Cycle 24 is based on a conservative DIVOM curve with increased slope provides the necessary margin between the operating limit at natural recirculation [[ ]] compared to the actual SLMCPR of 1.07.

The NRC staff review for this area identified a few areas where the staff will be discussing the currently approved licensing methodology for Option I-D with GE as part of its ongoing review of GE methods on a generic basis. Should the staff require additional modification of the GE methods, the staff will identify those additional requirements generically for all operating plants using Option I-D at that time.

2.8.7.5 Conclusion Regarding Methods Evaluation

The NRC staff evaluated the licensee's proposed Alternative Approach in which the licensee sought to demonstrate that there is adequate margin to the fuel design limit and key parameters that affect the safety analyses supporting the proposed VYNPS EPU. The licensee's Alternative Approach proposed applying additional uncertainties or demonstrating that margins to the fuel design are adequate to compensate for potential inaccuracies. Based on the review of each fuel design limit or margin as presented above, the NRC staff accepts Entergy's proposed Alternative Approach for VYNPS.

The NRC staff also confirmed that the thermal-hydraulic codes utilized for VYNPS were applied within the applicable ranges. The staff's conclusions are discussed in the specific topic areas of the SE stated above.

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Figure 2.8.7-1
Peak LHGR vs. Cycle Exposure
VYNPS, EPU Plants and a High Density 5% Uprate Plant
(Reference 25, Attachment 3, Page 9)

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Figure 2.8.7-2
Maximum Bundle Power vs. Cycle Exposure
VYNPS, EPU Plants and a High Density 5% Uprate Plant
(Reference 25, Attachment 3, Page 4)

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Figure 2.8.7-3
Exit Void Fraction of Maximum Power Bundle vs. Cycle Exposure
VYNPS, EPU Plants and a High Density 5% Uprate Plant
(Reference 25, Attachment 3, Page 6)

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Figure 2.8.7-4
Maximum Power/Flow Ratio vs. Cycle Exposure
VYNPS, EPU Plants and a High Density 5% Uprate Plant
(Reference 25, Attachment 3, Page 5)

[[

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Figure 2.8.7-5
VYNPS Actual Operating Kw/ft vs. Exposure
(Reference 35, Attachment 4)

Notes:

1. The figure shows projected actual operating history for VYNPS Cycle 25 for bundle JLC505, rod K4.

2. The JLC505 node is the highest projected bundle nodal exposure in VYNPS Cycle 25. Rod K4 is the highest exposure rod node in JLC505.

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Figure 2.8.7-6
Cold Critical Eigenvalue Results for Several Plants and Cycles
(Reference 72)

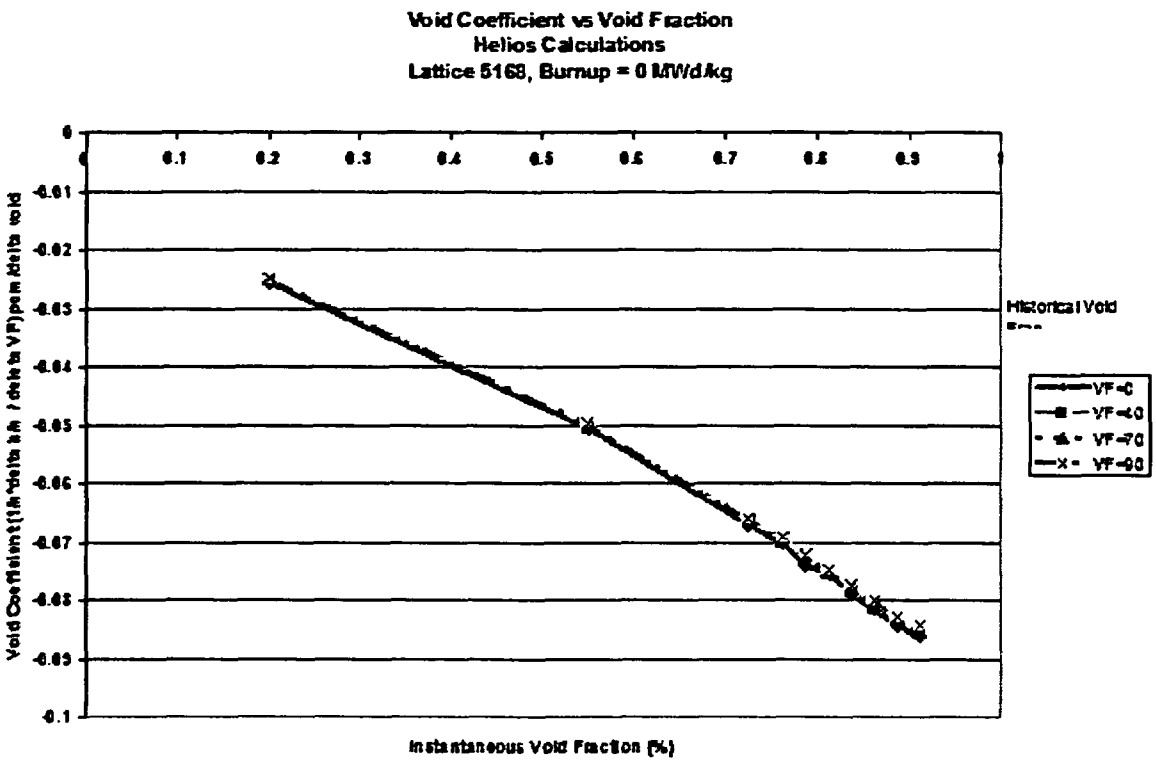

Figure 2.8.7-7
Void Reactivity Coefficient vs. Instantaneous Void Fraction at an Exposure of 0 GWd/t

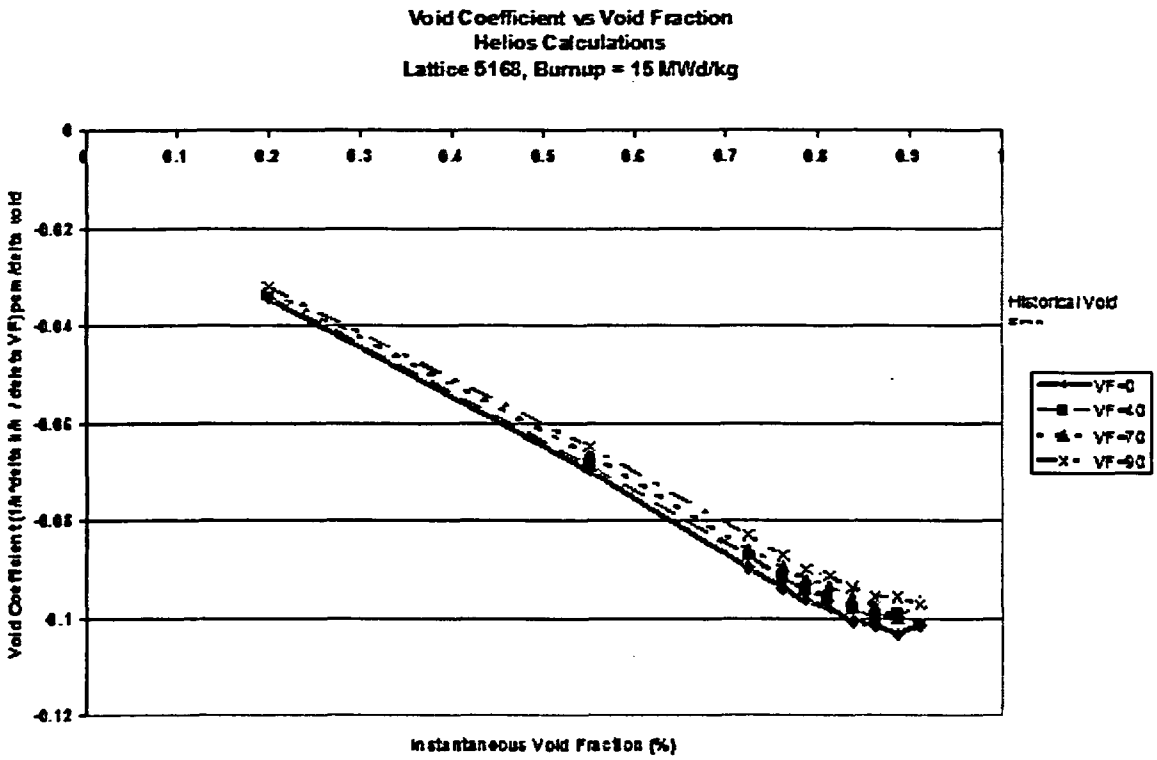

Figure 2.8.7-8
Void Reactivity Coefficient vs. Instantaneous Void Fraction at an Exposure of 15 GWd/t

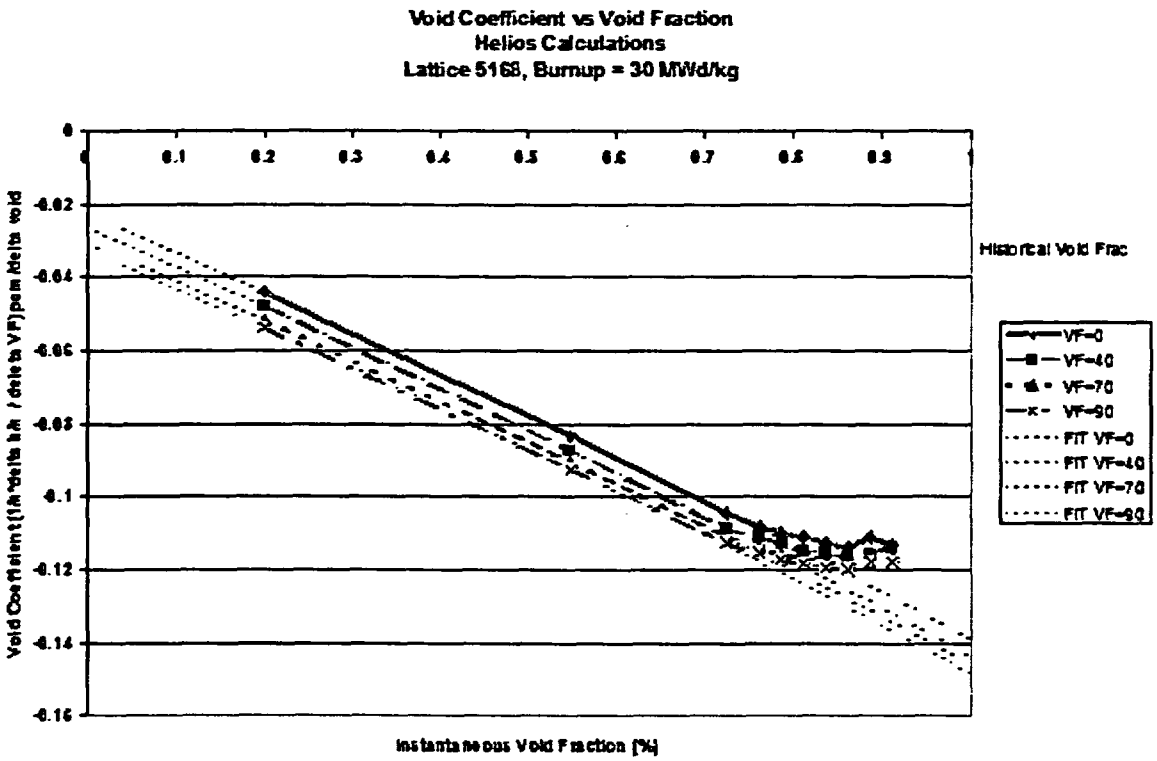

Figure 2.8.7-9
Void Reactivity Coefficient vs. Instantaneous Void Fraction at an Exposure of 30 GWd/t

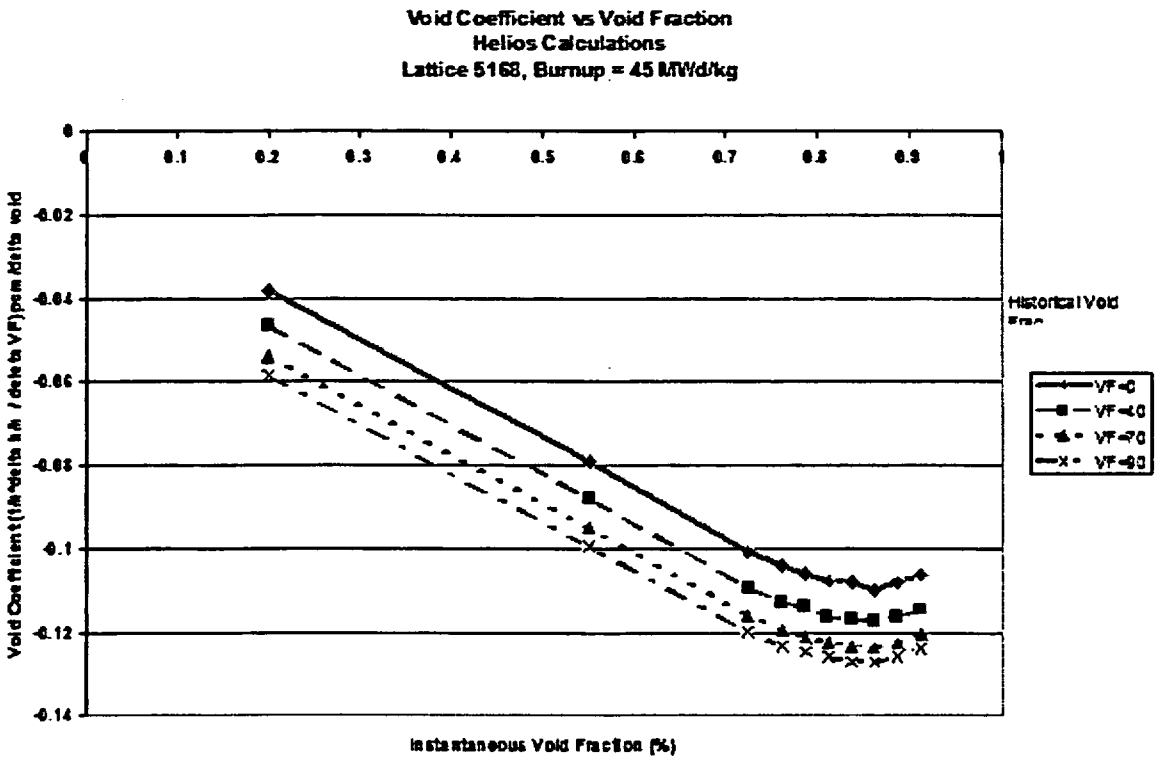

Figure 2.8.7-10
Void Reactivity Coefficient vs. Instantaneous Void Fraction at an Exposure of 45 GWd/t
(fit lines indicate result of linear fit of instantaneous values at 0, 40, and 70% void fraction)

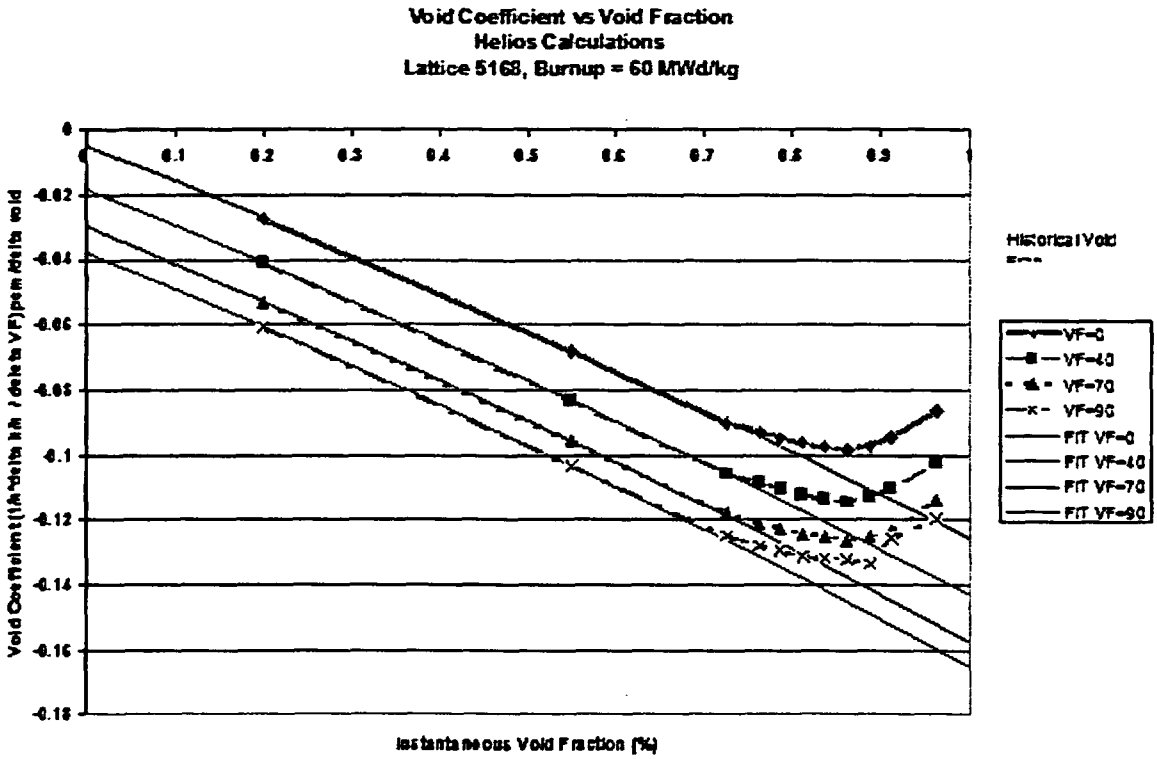

Figure 2.8.7-11
Void Reactivity Coefficient vs. Instantaneous Void Fraction at an Exposure of 60 GWd/t
(fit lines indicate result of linear fit of instantaneous values at 0, 40, and 70% void fraction)

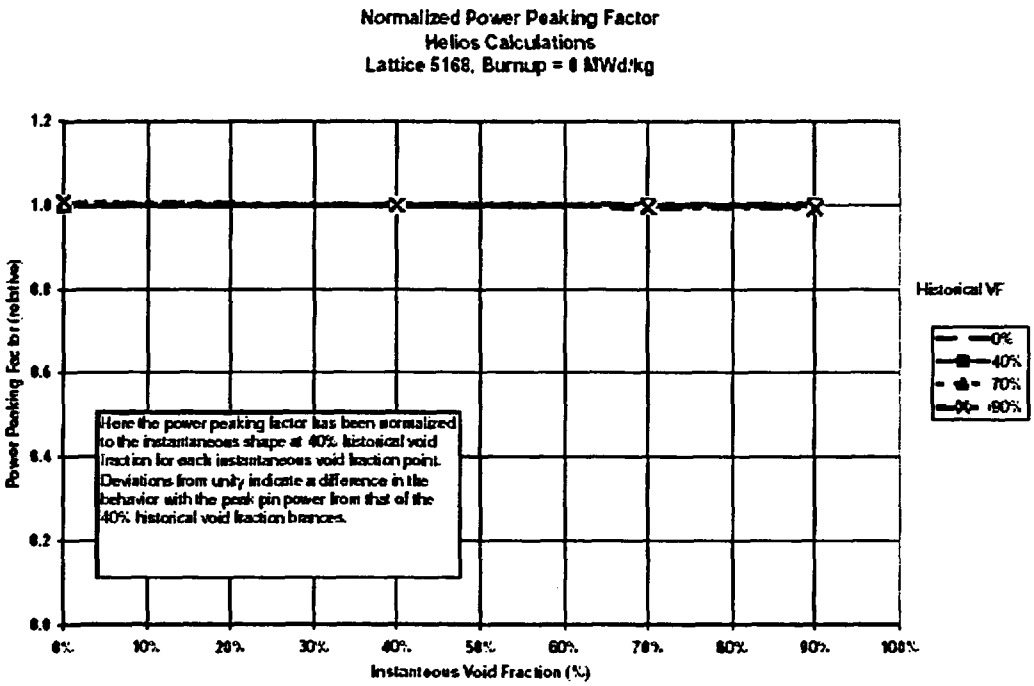

Figure 2.8.7-12
Branch Cases: Power Peaking Factors at Different Historical Voids
Normalized against 40% Historical Power Peaking vs. Exposure

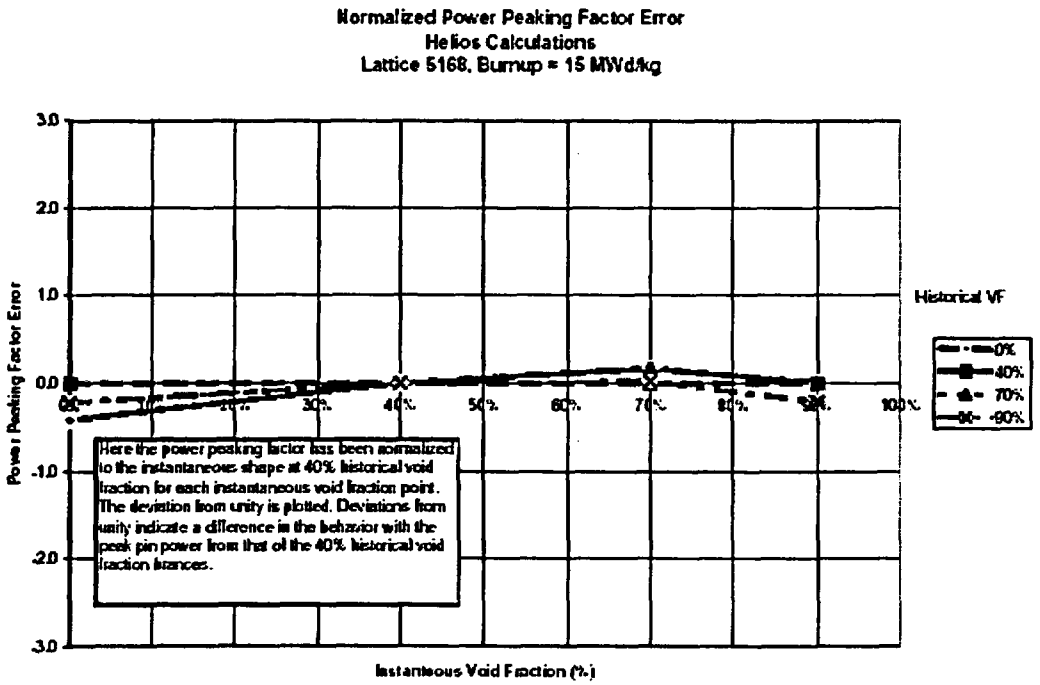

Figure 2.8.7-13
Branch Cases: Power Peaking Factors at Different Historical Voids
Normalized against 40% Historical Power Peaking vs. Exposure

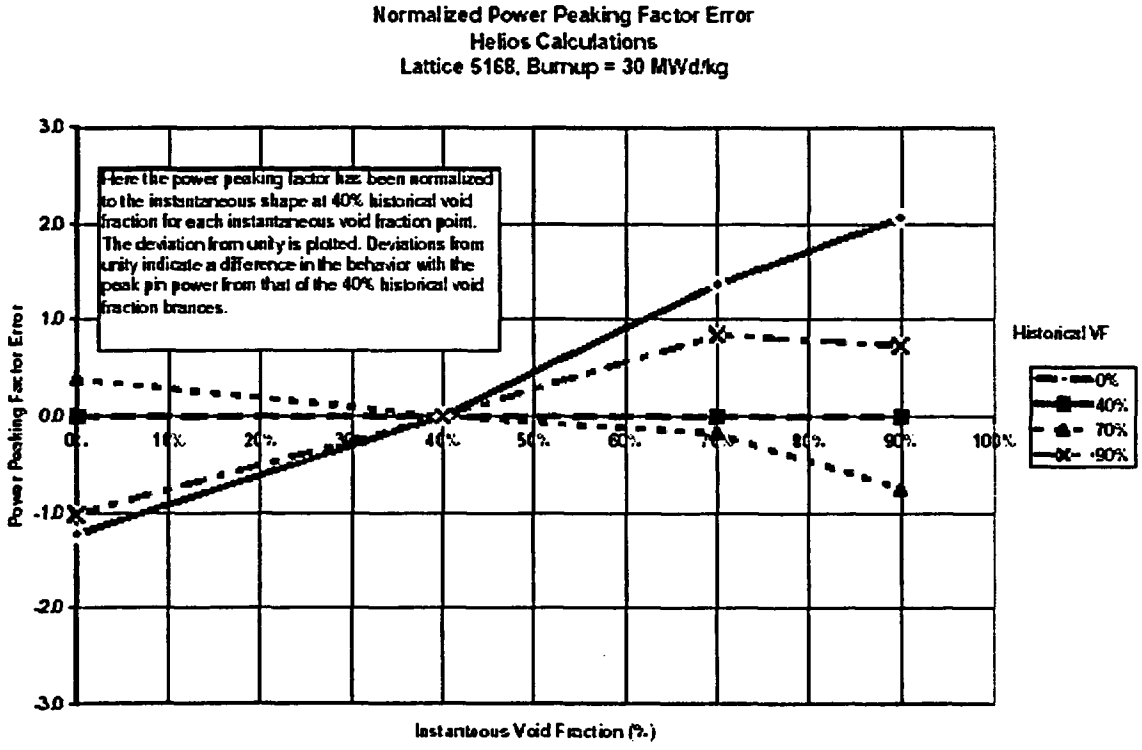

Figure 2.8.7-14
Branch Cases: Power Peaking Factors at Different Historical Voids
Normalized against 40% Historical Power Peaking vs. Exposure

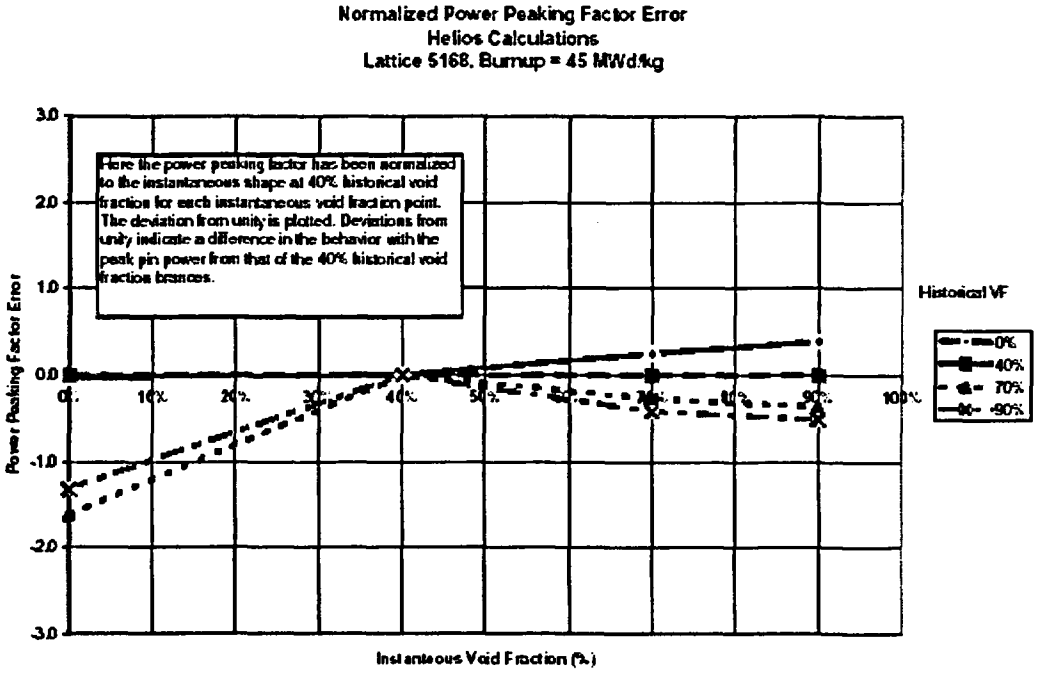

Figure 2.8.7-15
Branch Cases: Power Peaking Factors at Different Historical Voids
Normalized against 40% Historical Power Peaking vs. Exposure

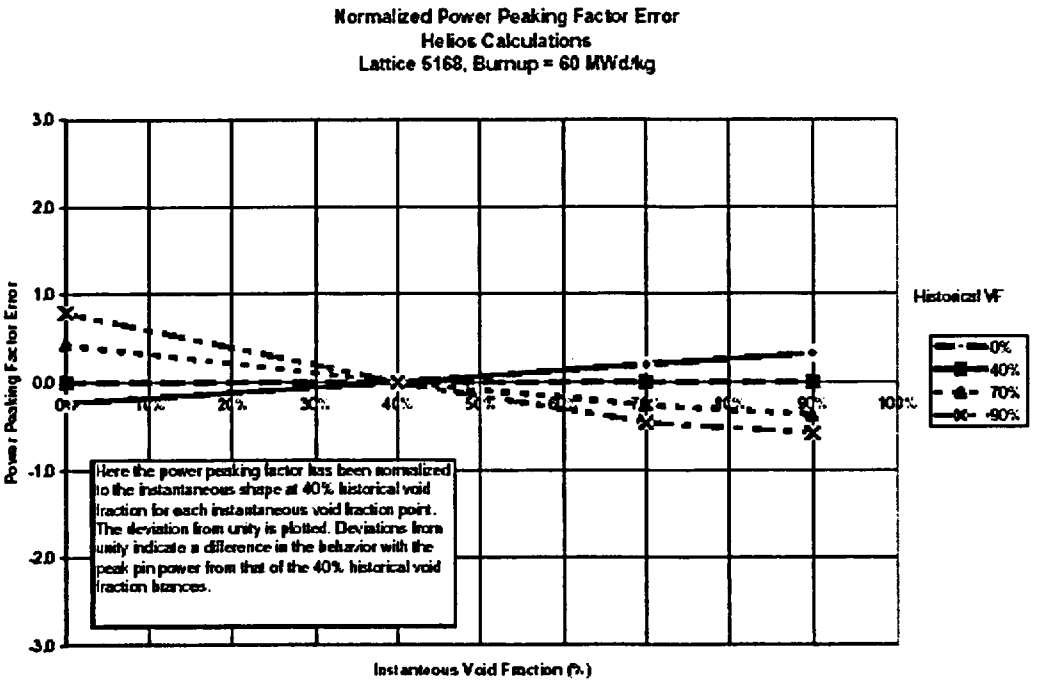

Figure 2.8.7-16
Branch Cases: Power Peaking Factors at Different Historical Voids
Normalized against 40% Historical Power Peaking vs. Exposure

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Figure 2.8.7-17
Void Coefficient Comparison Between TGBLA06 and MCNP
for Lattice 7009 at ≥70% VF (13 and 25 GWd/ST average)
(Reference 36, Attachment 1, Page 14)

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Figure 2.8.7-18
Void Coefficient Ratio MCNP / TGBLA06 for Lattice 7009 at ≥25 GWd/ST
(Reference 36, Attachment 1, Page 15)

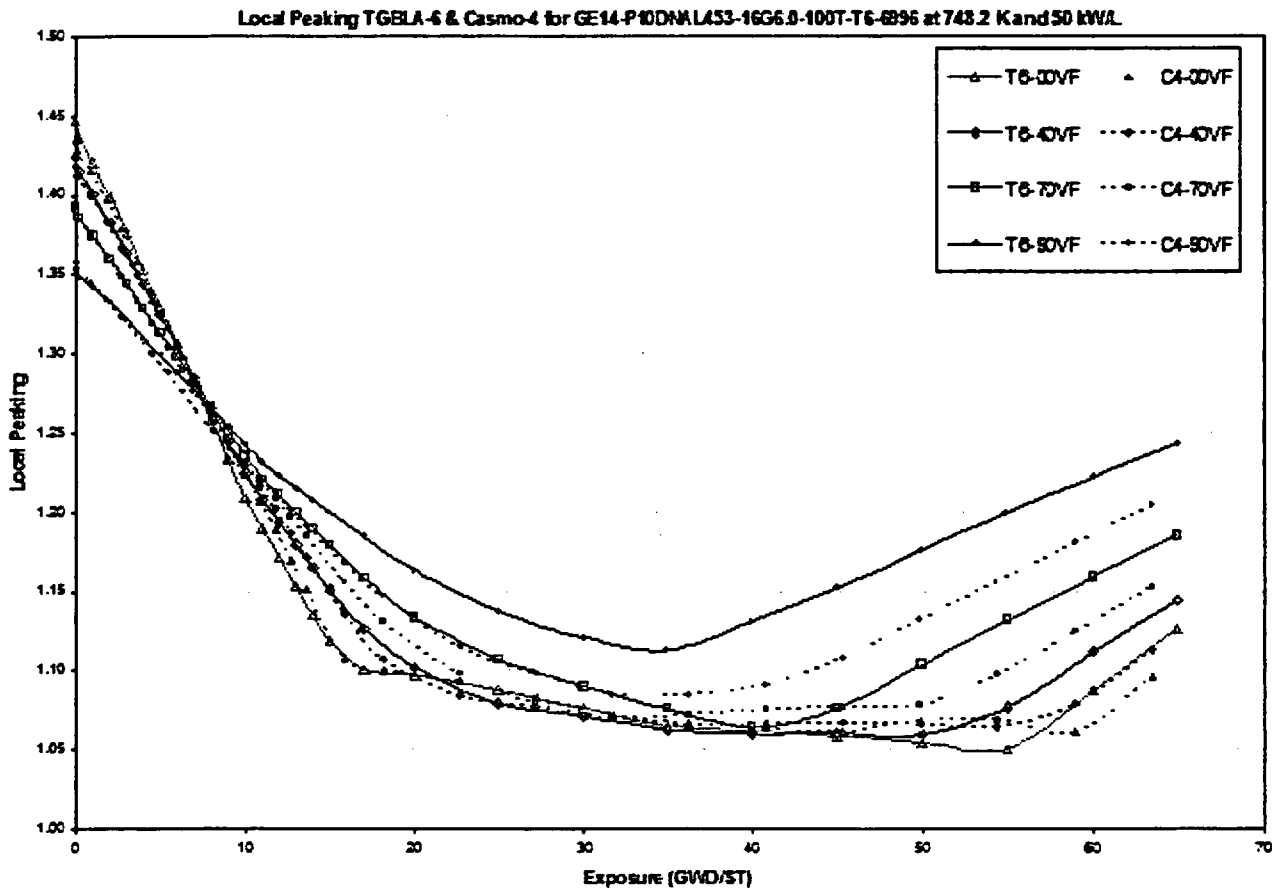

Figure 2.8.7-19
Comparison of TGBLA-6 and CASMO-4 Kinf Values for Lattice 6996

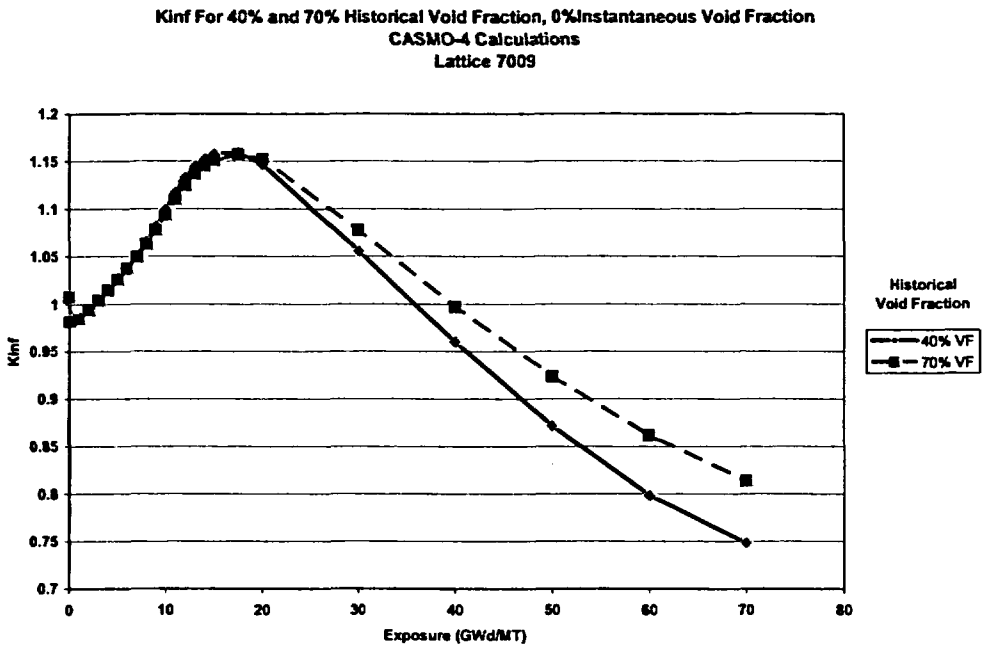

Figure 2.8.7-20
CASMO-4 Calculated Kinf Values
for Historical Void Fraction of 40% and 70% for Instantaneous Void Fraction of 0%.

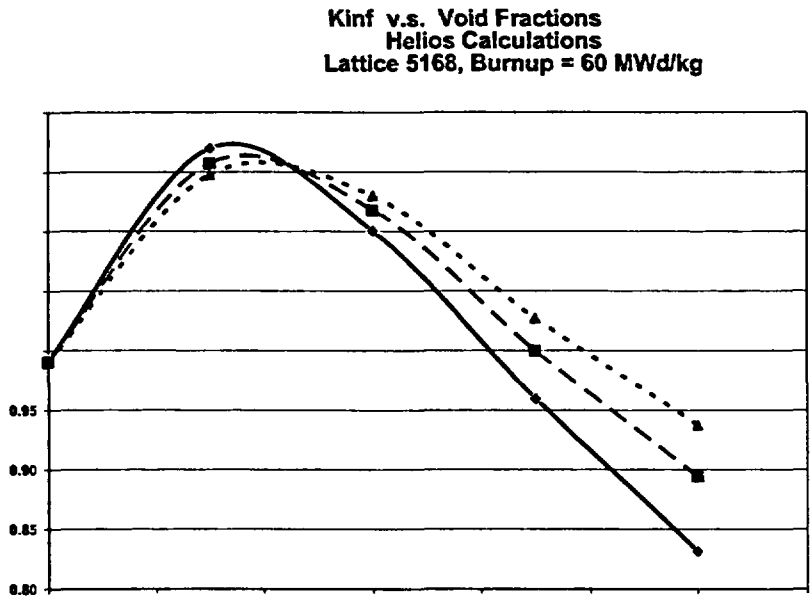

Figure 2.8.7-21
HELIOS Calculated Kinf Values
for Historical Void Fractions of 40%, 70%, and 90% and an Instantaneous Void Fraction of 0%.

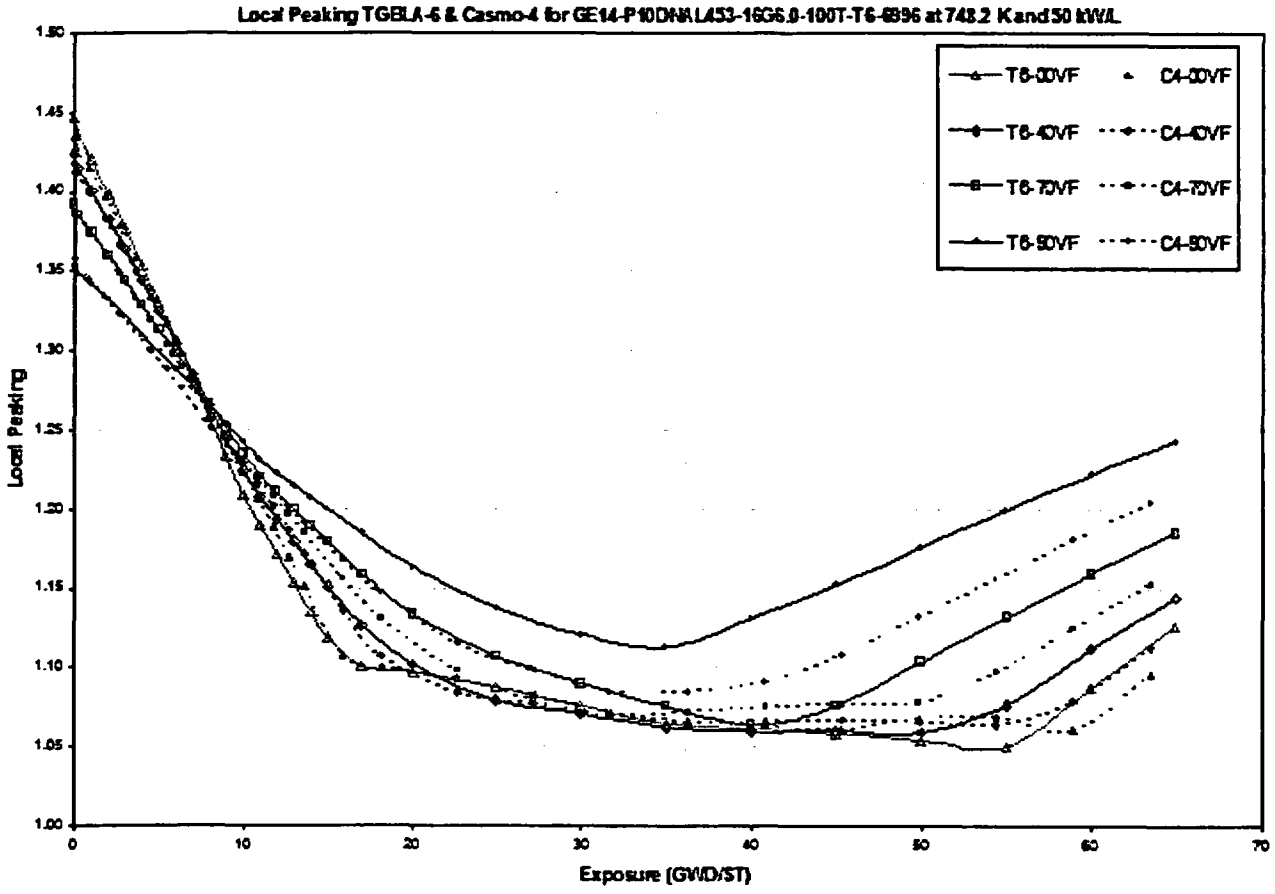

Figure 2.8.7-22
Comparison of TGBLA-6 and CASMO-4 Local Peaking for Lattices 6996

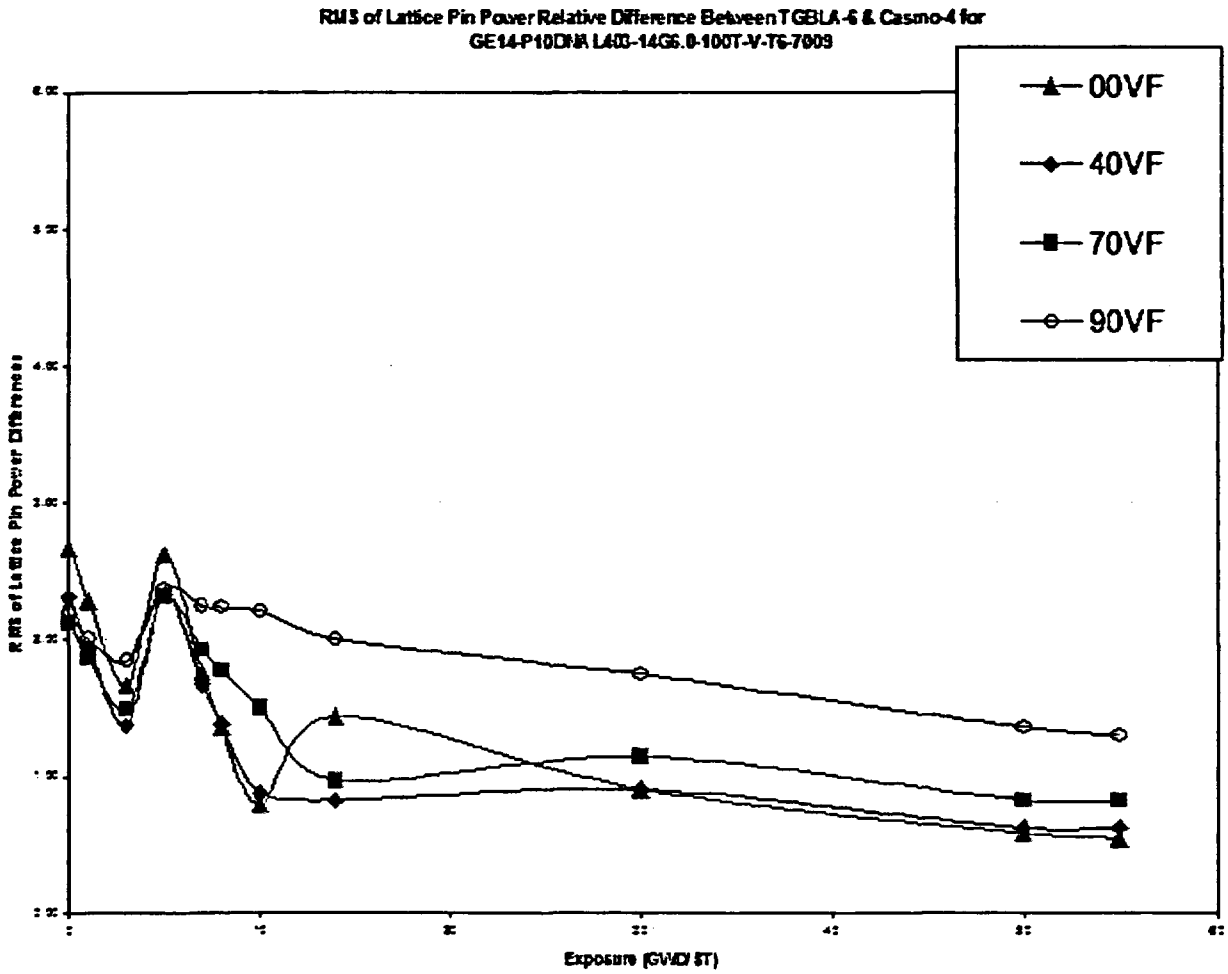

Figure 2.8.7-23
TGBLA-6 / CASMO-4 RMS of Lattice Pin Power Peaking

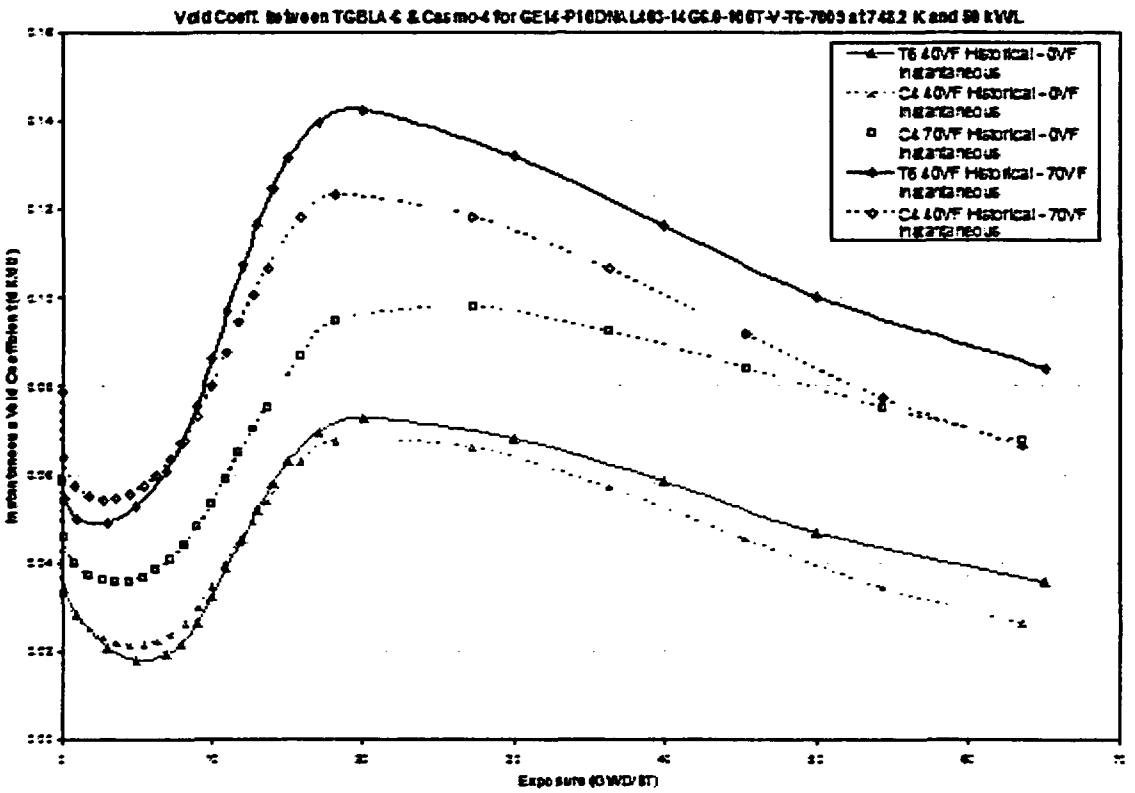

Figure 2.8.7-24
Comparison of TGBLA-6 and CASMO-4 Coolant Void Reactivity Coefficients

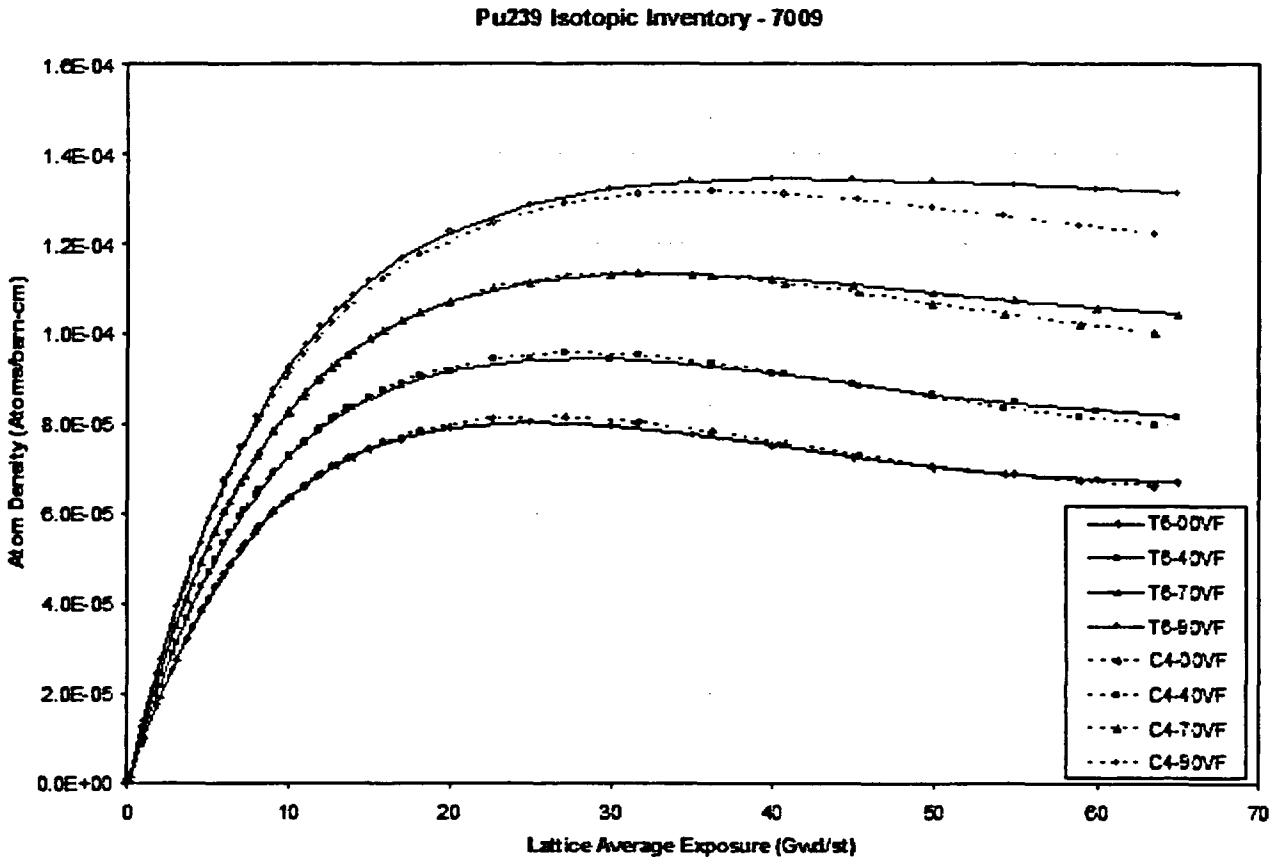

Figure 2.8.7-25
Comparison of TBGLA-6 and CASMO-4 Pu-239 Composition for Lattice 6996

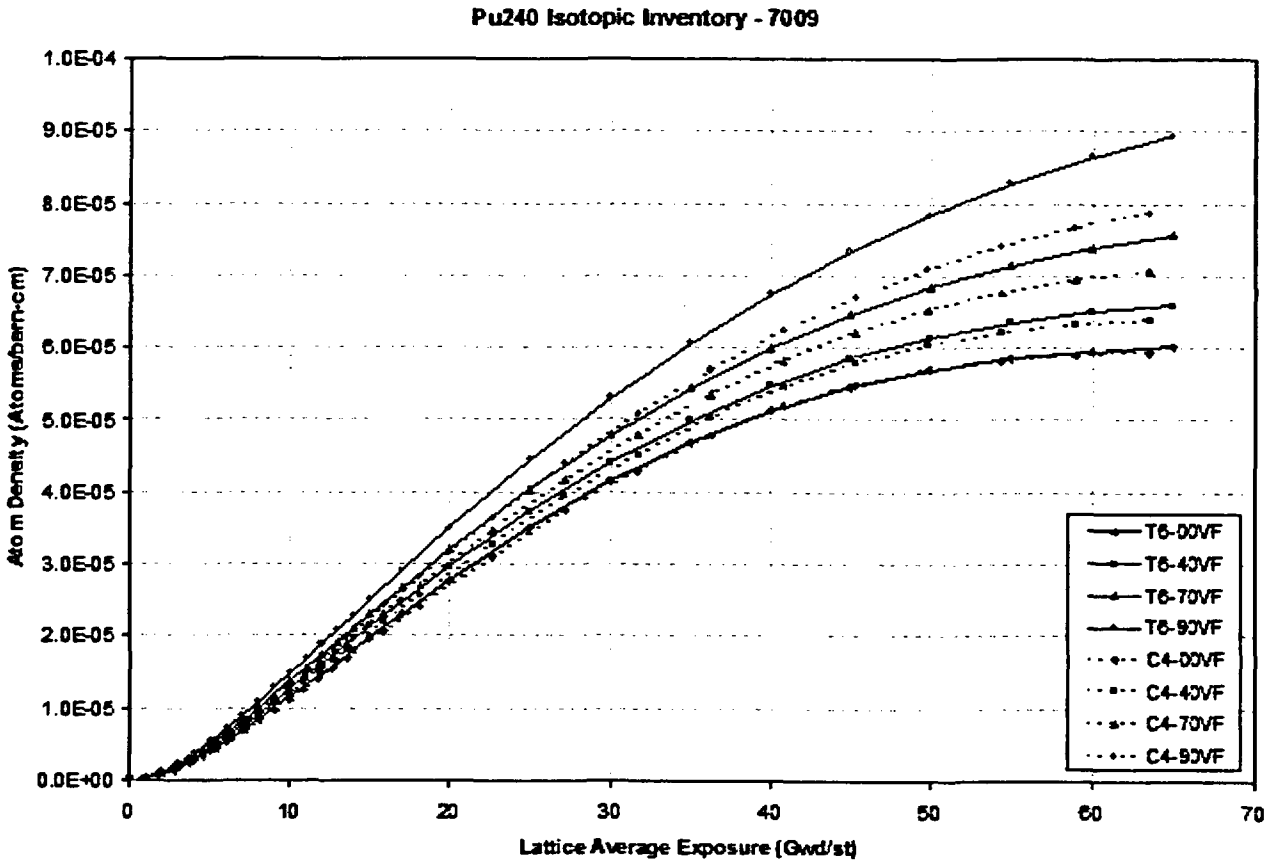

Figure 2.8.7-26
Comparison of TBGLA-6 and CASMO-4 Pu-240 Composition for Lattice 6996

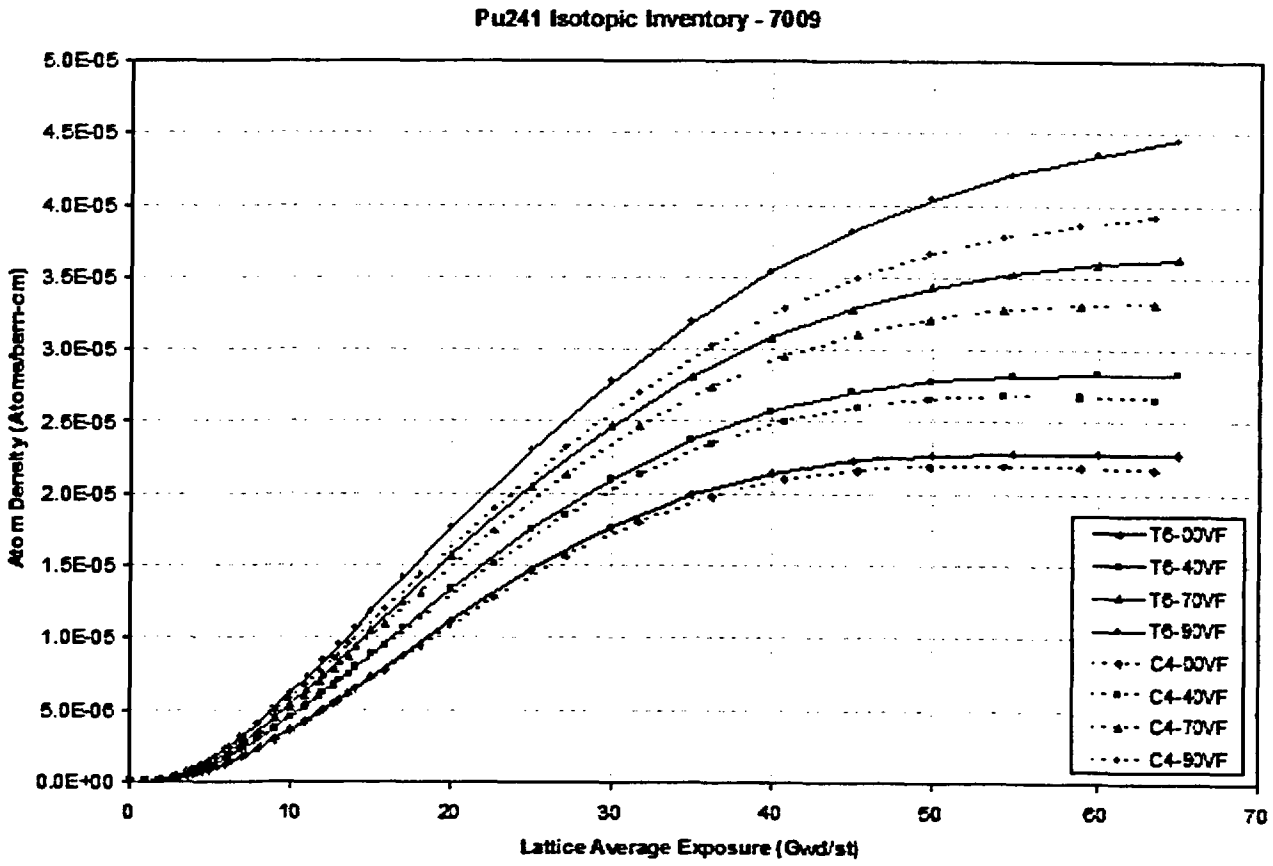

Figure 2.8.7-27
Comparison of TBGLA-6 and CASMO-4 Pu-241 Composition for Lattice 6996

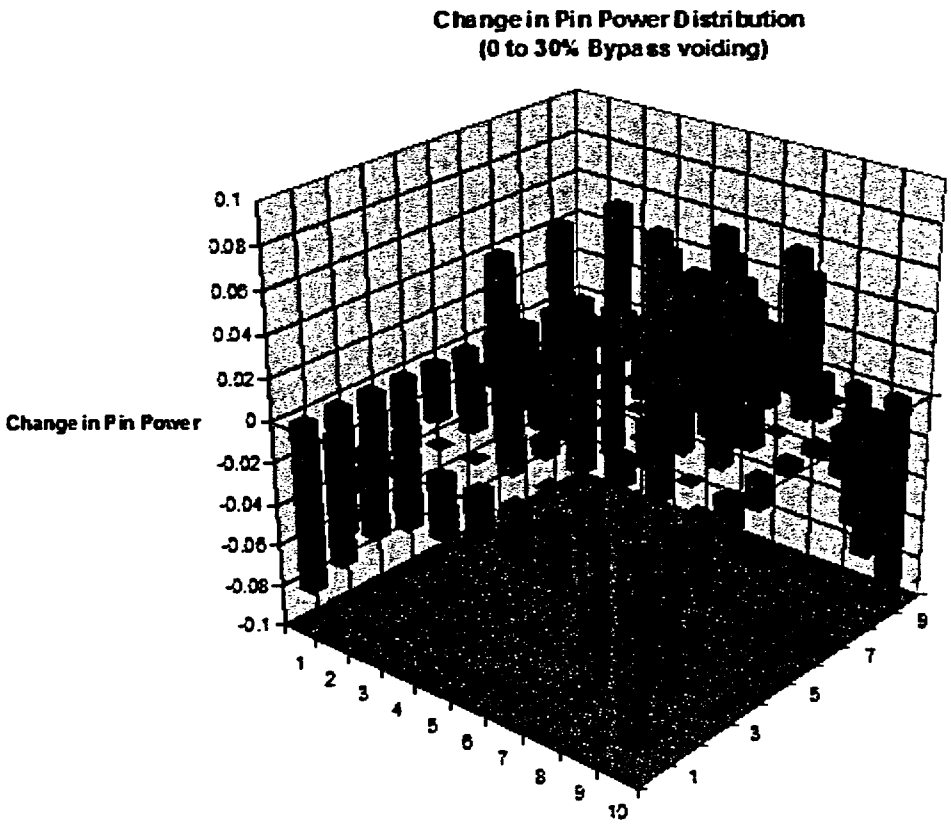

Figure 2.8.7-28
Change in Pin Power Distribution Between 0% and 30% Bypass and Water Rod Voiding

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Figure 2.8.7-29
GEXL-14 Test Range and Expected Ranges for Typical Operational Transients

2.9 Source Terms and Radiological Consequences Analyses

2.9.1 Source Terms for Radwaste Systems Analyses

Regulatory Evaluation

The NRC staff reviewed the radioactive source term associated with EPUs to ensure the adequacy of the sources of radioactivity used by the licensee as input to calculations to verify that the radioactive waste management systems have adequate capacity for the treatment of radioactive liquid and gaseous wastes. The NRC staff's review included the parameters used to determine (1) the concentration of each radionuclide in the reactor coolant, (2) the fraction of fission product activity released to the reactor coolant, (3) concentrations of all radionuclides other than fission products in the reactor coolant, (4) leakage rates and associated fluid activity of all potentially radioactive water and steam systems, and (5) potential sources of radioactive materials in effluents that are not considered in the plant's UFSAR related to liquid waste management systems and gaseous waste management systems. The NRC's acceptance criteria for source terms are based on (1) 10 CFR Part 20, insofar as it establishes requirements for radioactivity in liquid and gaseous effluents released to unrestricted areas; (2) 10 CFR Part 50, Appendix I, insofar as it establishes numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" criterion; and (3) draft GDC-70, insofar as it requires that the plant design include means to control the release of radioactive effluents. Specific review criteria are contained in SRP Section 11.1.

Technical Evaluation

In support of the subject license amendment request, the licensee provided analyses of the impact of the proposed EPU on the radiological consequences of DBAs in a separate license amendment request submittal which proposed a full-scope implementation of an alternative source term (AST) pursuant to 10 CFR 50.67. The NRC staff's evaluation of the licensee's calculated EPU radiological source term was performed as part of the review of the AST license amendment request. The AST amendment was approved on March 29, 2005 (Reference 57).

In Section 8 of the PUSAR, the licensee discussed the impact of operation at the proposed EPU power level on the source term for radioactive waste management systems. The licensee used either plant-specific evaluations or verified the applicability of the generic CLTR evaluations to VYNPS. As discussed in Section 2.5.5 of this SE, the NRC staff found that, for the proposed EPU, radioactive waste management systems would continue to control the release of radioactive materials consistent with the VYNPS licensing basis. As discussed in Section 2.10 of this SE, the NRC staff found that, for the proposed EPU, any increases in doses would remain ALARA and that the requirements in 10 CFR Part 20 would continue to be met.

Conclusion

The NRC staff has reviewed the radioactive source term associated with the proposed EPU and concludes that the proposed parameters and resultant composition and quantity of

radionuclides are appropriate for the evaluation of the radioactive waste management systems. The NRC staff further concludes that the proposed radioactive source term is appropriate for use in evaluating whether the requirements of 10 CFR Part 20, 10 CFR Part 50, Appendix I, and draft GDC-70 are met. Therefore, the NRC staff finds the proposed EPU acceptable with respect to source terms.

2.9.2 Radiological Consequences Analyses Using Alternative Source Terms

Regulatory Evaluation

The NRC staff evaluation included review of the DBA radiological consequences analyses. The radiological consequences analyses reviewed are the LOCA, fuel-handling accident (FHA), control rod drop accident (CRDA), and main steam line break (MSLB). The NRC staff's review for each accident analysis included (1) the sequence of events; and (2) models, assumptions, and values of parameter inputs used by the licensee for the calculation of the total effective dose equivalent (TEDE). The NRC's acceptance criteria for radiological consequences analyses using an alternative source term are based on 10 CFR 50.67, insofar as it sets standards for radiological consequences of a postulated accident. Specific review criteria are contained in SRP Section 15.0.1.

Technical Evaluation

In a separate amendment request, the licensee proposed a full-scope implementation of an alternative source term (AST) for VYNPS pursuant to 10 CFR 50.67. The AST amendment request assumed operation at the proposed EPU power level. The AST amendment was approved on March 29, 2005 (Reference 57). The NRC staff's evaluation of the licensee's calculated EPU radiological source term was performed as part of the review of the AST license amendment request. Within the AST review, as documented in the associated SE dated March 29, 2005, the NRC staff determined that the licensee has shown that the proposed changes, including uprated power, are acceptable with respect to the radiological consequences of all applicable DBAs. The licensee's dose analyses show that the dose criteria of 10 CFR 50.67, as further clarified in SRP 15.0.1, are met for the EPU. As a result of the AST license amendment review, the staff found the licensee's dose analysis methodology, assumptions and inputs to be acceptable. As part of the AST review, the staff also performed independent dose analyses which confirmed the licensee's dose results.

Conclusion

As part of the evaluation of the full-scope implementation of an AST at VYNPS, the NRC staff evaluated the licensee's revised accident analyses performed in support of the proposed EPU and concluded that the licensee has adequately accounted for the effects of the proposed EPU. In its review of the proposed full-scope implementation of an AST at VYNPS, the NRC staff further concluded that the plant site and the dose-mitigating ESFs remain acceptable with respect to the radiological consequences of postulated DBAs since the calculated TEDE at the exclusion area boundary (EAB), at the low population zone (LPZ) outer boundary, and in the

control room, meet the exposure guideline values specified in 10 CFR 50.67, as well as applicable acceptance criteria denoted in SRP Section 15.0.1. Therefore, based on the issuance of the full-scope implementation AST license amendment and its accompanying SE, the NRC staff finds the licensee's proposed EPU acceptable with respect to the radiological consequences of DBAs.

2.10 Health Physics

2.10.1 Occupational and Public Radiation Doses

Regulatory Evaluation

The NRC staff conducted its review in this area to ascertain what overall effects the proposed EPU will have on both occupational and public radiation doses and to determine whether the licensee has taken the necessary steps to ensure that any dose increases will be maintained within applicable regulatory limits and as low as is reasonably achievable (ALARA). The NRC staff's review included an evaluation of any increases in radiation sources and how this may affect plant area dose rates, plant radiation zones, and plant area accessibility. The NRC staff evaluated how doses for personnel needed to access plant vital areas following an accident are affected. The NRC staff considered the effects of the proposed EPU on nitrogen-16 (N-16) levels in the plant and any effects this increase may have on radiation doses outside the plant and at the site boundary from skyshine. The NRC staff also considered the effects of the proposed EPU on plant effluent levels and any effect this increase may have on radiation doses at the site boundary. The NRC's acceptance criteria for occupational and public radiation doses are based on 10 CFR Part 20, 10 CFR 50.67, and draft GDC-11. Specific review criteria are contained in SRP Sections 12.2, 12.3,12.4, and 12.5, and other guidance provided in Matrix 10 of RS-001.

Technical Evaluation

Source Terms

In general, the production of radiation and radioactive material (either fission or activation products) in the reactor core are directly dependent on the neutron flux and power level of the reactor. Therefore, as a first order approximation, a 20% increase in power level is expected to result in a proportional increase in the direct (i.e., from the reactor fuel) and indirect (i.e., from the reactor coolant) radiation source terms. However, due to the physical and chemical properties of the different radioactive materials that reside in the reactor coolant, and the various processes that transport them to locations in the plant outside the reactor, several radiation sources encountered in the balance of plant are not expected to change in direct proportion to the increased reactor power. The most significant of these are:

1. The concentration of noble gas and other volatile fission products in the main steam line will not change. The increased production rate (20%) of these materials is offset by the corresponding increase in steam flow (20%). Although the concentration of these materials

in the steam line remains constant, the increased steam flow results in a 20% increase in the rate these materials are introduced into the main condenser and offgas systems.

2. For the very short lived activities, most significantly N-16, the decreased transit (and decay time) in the main steam line, and the increased mass flow of the steam results in a larger increase in these activities in the major turbine building components. For N-16, with its 7.13 second half-life, the licensee estimates a 26% increase in activity in the turbine building.

3. The concentrations of non-volatile fission products, actinides, and corrosion and wear products in the reactor coolant are expected to increase proportionally with the power increase. However, the increased steam flow is expected to result in an increased moisture carryover in the steam, resulting in an increased transport of these activities to the balance of the plant. The licensee has calculated that the 20% increase in steam flow will double the moisture carry-over (from 0.04% to 0.08%) resulting in an overall increase in the condensate system by a factor of 2.4. The radiation from these non-volatile radioactive materials provides only a small contribution to the dose rates around balance of plant systems during normal power operations.

Radiation Protection Design Features

1. Occupational and onsite radiation exposures.

The radiation sources in the core are expected to increase in proportion to the increase in power. This increase, however, is bounded by the existing safety margins of the plant design. Due to the design of the shielding and containment surrounding the reactor vessel, and since the reactor vessel is inaccessible to plant personnel during operation, a 20% increase in the radiation sources in the reactor core will have no effect on occupational worker personnel doses during power operations. Similarly, the radiation shielding provided in the balance of plant (i.e., around rad-waste systems, main steam lines, the main turbine, etc.) is conservatively sized such that the increased source terms discussed above are not expected to significantly increase the dose rates in the normally occupied areas of the plant. The existing radiation zoning design (e.g., the maximum designed dose rates for each area of the plant) will not change as a result of the increased dose rates associated with the EPU.

Operating at a 20% higher power level will result in an increased core inventory of radioactive material that is available for release during postulated accident conditions. The plant shielding design must be sufficient to provide control room habitability, per Draft GDC 11, and operator access to vital areas of the plant, per NUREG-0737, item II.B.2, during the accident. As part of a recent change to the VYNPS design basis, the licensee recalculated the radiological consequences of the postulated design basis accidents using the AST in accordance with the provisions in 10 CFR 50.67. The AST, which was approved in VYNPS Amendment No. 223, issued on March 29, 2005 (Reference 57), provides more realistic assumptions, than the previous VYNPS design basis source term, on the timing and mechanisms of radioactive material release from the core during postulated accident conditions. The licensee's reevaluation of the DBAs included an evaluation of control room habitability, and post accident

vital areas access, at the proposed EPU power level of 1912 MWt. The NRC staff reviewed this design basis change and concludes that licensee continues to meet the applicable requirements.

Therefore, following implementation of the EPU, VYNPS will continue to meet its design basis in terms of radiation shielding, in accordance with the criteria in SRP Section 12.4, draft GDC 11, and NUREG-0737, item II.B.2.

2. Public and offsite radiation exposures.

There are two factors associated with the EPU that may impact public and offsite radiation exposures during plant operations. These are the possible increases in gaseous and liquid effluents released from the site, and the increase in direct radiation exposure from radioactive plant components and solid wastes stored onsite. As described above, the proposed EPU will result in a 20% increase in gaseous effluents released from the plant during operations. This increase is a minor contribution to the radiation exposure of the public. The nominal annual public dose from plant gaseous effluents for the VYNPS station is about 1 mrem. A 20% increase in this nominal dose is still well within the design criteria of 10 CFR Part 50, Appendix I.

The proposed EPU will also result in increased generation of liquid and solid radioactive waste. The increased condensate feed flow associated with the EPU results in faster loading of the condensate demineralizers. Similarly, the higher feed flow introduces more impurities into the reactor resulting in faster loading of the reactor water cleanup (RWCU) system demineralizers. Therefore, the demineralizers in both of these systems will require more frequent backwashing to maintain them. The licensee has estimated that these more frequent backwashes will increase the volume of liquid waste that will need processing by 1.2%, and an increase in processed solid radioactive waste by 17.8%. These increases are well within the processing capacity of the VYNPS radwaste system and are not expected to noticeably increase the liquid effluents or solid radioactive waste released from the plant. Therefore, these increases will have a negligible impact on occupational or public radiation exposure.

The most significant increase in offsite doses, from the proposed EPU will be due to increased N-16 skyshine and the direct exposure to radiation from miscellaneous radioactive waste stored on site. Based on measurements, the licensee has determined that the west boundary of the facility has the highest direct offsite radiation dose, nominally 15 mrem per year. The licensee has estimated that almost 90% of this dose, 13.4 mrem per year, is due to N-16 skyshine from the turbine building components. Skyshine is a physical phenomenon where gamma radiation that is released skyward during radioactive decay interacts with air molecules and, in this case, is scattered back down to the ground where it can expose members of the public. Since there is significantly less radiation shielding above the steam components in the turbine building, than there is to the sides of these components, skyshine from N-16 gamma radiation is a significant contributor to offsite dose rates. As discussed above, the licensee has estimated that plant operations at the EPU power level will increase the N-16 activity in the turbine building by 26%. Therefore, the gamma dose rate from N-16 skyshine at the west site boundary will likely

increase to a nominal value of 16.9 mrem per year. Increases in the solid radioactive waste resulting from this EPU, which are stored on site, can also increase the direct radiation dose rate offsite. However, the licensee has committed to administratively control the contribution to offsite dose rates from these miscellaneous radioactive wastes. The maximum dose rate contribution, for the highest offsite location (west boundary), from radioactive waste stored onsite will be 1.74 mrem per year. Therefore, the projected maximum offsite dose rate from direct radiation exposure following this EPU is estimated to be about 18.6 mrem per year. This annual dose is within the applicable 40 CFR Part 190 annual limit of 25 mrem to an actual member of the public, as referenced by 10 CFR 20.1301(e).

As indicated in Attachment 3 of the licensee's application dated September 10, 2003 (Reference 1), the licensee plans to perform radiation surveys as part of the EPU power ascension testing. The surveys will be conducted at approximately 100%, 105%, 110%, 115%, and 120% of CLTP. In fall 2005, as part of the NRC's baseline inspection process, NRC Region I staff, with support from NRC Headquarters staff, initiated an inspection of the direct dose calculation methodology described in the VYNPS Offsite Dose Calculation Manual (ODCM). The calculation methodology described in Section 6.11.1 of the ODCM is used by the licensee to ensure compliance with the offsite dose requirement in 10 CFR 20.1301(e). As part of the ODCM methodology inspection effort, the NRC staff plans to review the results of the licensee's power ascension radiation surveys to confirm that the dose to a member of the public continues to meet the annual limit under EPU conditions.

Operational Radiation Protection Programs

The increased production of non-volatile fission products, actinides, and corrosion and wear products in the reactor coolant may result in proportionally higher plate-out of these materials on the surfaces of, and low flow areas in, reactor systems. The corresponding increase in dose rates associated with these deposited materials will be an additional source of occupational exposure during the repair and maintenance of these systems. However, the current ALARA program practices at VYNPS (i.e., work planning, source term minimization, etc.), coupled with existing radiation exposure procedural controls, will be able to compensate for the anticipated increases in dose rates associated with the EPU. Therefore, the increased radiation sources resulting from the proposed EPU, as discussed above, will not adversely impact the licensee's ability to maintain occupational and public radiation doses resulting from plant operation within the applicable limits in 10 CFR Part 20 and ALARA.

Conclusion

The NRC staff has reviewed the licensee's assessment of the effects of the proposed EPU on radiation source terms and plant radiation levels. The NRC staff concludes that the licensee has taken the necessary steps to ensure that any increases in radiation doses will be maintained ALARA. The NRC staff further concludes that the proposed EPU meets the requirements of 10 CFR Part 20 and draft GDC-11. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to radiation protection and ensuring that occupational radiation exposures will be maintained ALARA.

2.11 Human Performance

2.11.1 Human Factors

Regulatory Evaluation

The area of human factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC staff's human factors evaluation was conducted to ensure that operator performance is not adversely affected as a result of system changes made to implement the proposed EPU. The NRC staff's review covered changes to operator actions, human-system interfaces, and procedures and training needed for the proposed EPU. The NRC's acceptance criteria for human factors are based on draft GDC-11, 10 CFR 50.120, 10 CFR Part 55, and the guidance in GL 82-33. Specific review criteria are contained in SRP Sections 13.2.1, 13.2.2, 13.5.2.1, and 18.0.

Technical Evaluation

Changes in Emergency and Abnormal Operating Procedures

The licensee indicated that the Emergency Operating Procedures (EOPs)/Severe Accident Management Guidelines (SAMGs) should remain unchanged in most aspects, with slight modifications required for some parameter thresholds and graphs which depend on the power and decay heat levels. These modifications would require changes in some values in the EOPs and the supporting documentation, but the adjustments would not affect the accident mitigation philosophy. Additionally, any change in scenario timings would be minor and would not significantly change the Human Error Probabilities (HEPs) in the risk assessments. The licensee will review the EOPs for any required changes, implementing those changes, and providing training to operators on the procedures.

For the Abnormal Operating Procedures (AOPs), the licensee indicated that some operator actions may be influenced by plant modifications required for supporting the increase in rated thermal power. The increased power level may require modifications to the AOPs and the supporting documentation. The licensee will review the AOPs to identify any effects of the EPU, including modifications to equipment and changes in setpoints to implement any changes to the AOPs, equipment, and setpoints necessary as a result of those effects, and to provide training to operators on the AOPs, equipment modifications, and setpoint changes.

Because no new procedures would be required, necessary changes to EOPs/SAMGs/AOPs, equipment and setpoints will be implemented, and training to address these changes will be provided, the NRC staff finds the licensee's proposed actions in this area to be acceptable.

Changes to Operator Actions Sensitive to EPU

The licensee stated that operator responses to transients, accidents, and special events would be minimally affected by EPU conditions. Operator actions for plant safety, after applicable

automatic responses initiate, would not change as result of the EPU. The licensee's submittal described an operational enhancement that would provide for automatic recirculation runback given a single reactor feedwater pump (RFP) trip under EPU conditions, and this enhancement can be regarded essentially as the automation of an operator action.

The licensee also explained that there would be small reductions in time available for some operator actions due to the increase in decay heat for the EPU. Based on a screening process using risk assessment, the licensee identified a list of actions for explicit consideration. These actions were listed with current and EPU available times for completion. The licensee calculated the HEP for each action using industry standard techniques that included estimation of cognitive and manipulation times. The licensee stated that the shorter time limits would still be sufficient for operators to complete the tasks. To support the time estimates, the licensee performed interviews with multiple cognizant individuals, including a Senior Reactor Operator (SRO), a trainer, and an EOP developer. The time estimates were then entered into a HEP equation with standard deviation values based on an industry study of operator response times to over 100 different human actions. The licensee stated that differences in abilities of crews were taken into consideration in estimating the completion times and by using the HEP equation mentioned. Therefore, the licensee considered performance of all VYNPS operating crews to be bounded by the HEP calculations. Additionally, the licensee provided specific details on four of the most time-limited actions, which were all related to ATWS scenarios. For these actions, the licensee provided the current as well as EPU times available, along with estimated cognitive and manipulation times. For all four, the combined cognitive and manipulation estimated times were below the time limits under EPU.

As discussed in SE Section 1.6, an engineering inspection was performed at VYNPS, from August 9 through September 3, 2004, as documented in an inspection report dated December 2, 2004 (Reference 55). One of the inspection findings related to whether the licensee had appropriately accounted for the effects of the proposed EPU on the available time for operator actions related to a 10 CFR Part 50, Appendix R fire event. Specifically, the engineering inspection team found that the timeline for operator actions to place RCIC in service during an Appendix R scenario had been impacted due to procedure changes and that the licensee had not incorporated these changes into the VYNPS Safe Shutdown Capability Analysis (SCCA). However, the team found that at the current power level, during an Appendix R scenario, the operators have sufficient time to place RCIC in service from the alternate shutdown panels prior to reactor water level reaching the top of active fuel. At the proposed EPU power level, the team concluded that the margin was reduced such that the ability to place RCIC in service from the alternate shutdown panels prior to reactor water level reaching the top of active fuel was questionable. The specific details from the inspection report are as follows:

> The Vermont Yankee SSCA relies on the reactor core isolation cooling (RCIC) system to be placed in service from the alternate shutdown panels prior to reactor water level reaching the top of active fuel following a loss of feedwater flow. In December 1999, the Vermont Yankee SSCA documented that, for the present day 100 percent power level, it would take 25.3 minutes for reactor water level to reach the top of active fuel following a

loss of feedwater and that it would take approximately 15 minutes to place the RCIC system in service from the alternate shutdown panels. The Vermont Yankee SSCA concluded adequate margin (approximately 10 minutes) existed to ensure that the RCIC is placed in service prior to reactor water level reaching the top of active fuel.

In June 2001 the Operations Department conducted an additional review of the time it would take to place RCIC in service from the alternate shutdown panels. The Operations Department determined that, using the version of the procedure in effect in June 2001, it would take 19.3 minutes to place RCIC in service from the alternate shutdown panels.

During the inspection, using the version of the procedure in effect during the inspection period, the team performed a field walkdown with licensed operators to validate that RCIC could be placed into service from the alternate shutdown panels within 19.3 minutes. The team noted that since June 2001, the licensee had added steps in the procedure to comply with Electrical Safety Standards. Based on the team's validation, the total time to place RCIC in service from the alternate shutdown panels was determined to be approximately 21 minutes. The team concluded that this time was still within the 25.3 minute limit stated in the Vermont Yankee SSCA.

Additionally, the team found that the licensee had not revised the December 1999 Vermont Yankee SSCA to reflect the June 2001 time estimate or present day version of the procedure to place RCIC in service from the alternate shutdown panels. The team also determined that the licensee's engineering organization was unaware that the time to complete the task had increased from approximately 15 to 21 minutes and had effectively reduced the time margin available for event mitigation from about 10 minutes to 4 minutes at the current full power level. As a consequence, the engineering organization had not revised the Vermont Yankee SSCA.

The team reviewed the impact the licensee's proposed EPU would have on this issue. Based on an EPU power level, the licensee calculated it would take 21.3 minutes for reactor water level to reach the top of active fuel following a loss of feedwater. Therefore, the team concluded that for the proposed EPU, the ability to place the RCIC in service from the alternate shutdown panels (21 minutes) prior to reactor water level reaching the top of active fuel (21.3 minutes) is questionable.

As a corrective action, in response to the NRC staff's inspection finding, the licensee revised the procedure governing the required operator actions, completed training of the VYNPS licensed operators on the revised procedure, and performed timed walkthroughs of the actions required in the procedure with all six VYNPS operating crews. The results of the walkthroughs, as documented in the licensee's letter dated December 8, 2004 (Reference 23), were as follows:

Operating Crew	Time to RCIC Initiation (minutes:seconds)
A	14:38
B	13:26
C	12:26
D	15:09
E	13:18
F	12:17
Average	13:32

The licensee concluded that, based on the results of this demonstration, the assumption in the SCCA that the RCIC system can be made operable in approximately 15 minutes is confirmed. As discussed above, and as shown in PUSAR Table 6-5, the time to core uncovery calculated for EPU conditions is 21.3 minutes. Therefore, the NRC staff concludes that sufficient margin exists to allow operator action to manually start the RCIC system during an Appendix R event.

Based on the above discussion (i.e., automatic recirculation pump runback, ATWS scenarios, Appendix R event), the NRC staff concludes that there is reasonable assurance that the licensee has appropriately accounted for the effects of the proposed EPU on the available time for operator actions.

Changes to Control Room Controls, Displays and Alarms

The licensee analyzed potential system changes as result of the EPU and indicated that the following control room instrumentation would require modification: main steam line flow indicators would be replaced with digital units, feedwater (FW) flow indicators would be replaced with digital units, the main steam (MS) flow/FW flow recorder would be rescaled and use new chart paper, and the condensate flow recorder would be re-scaled. The licensee stated that none of these control room display changes would affect the Human Reliability Analysis results. Additionally, all modifications would be implemented in accordance with the VYNPS design modification process, which requires both human factors review and impact review by operations and training personnel. Training would also be provided on these changes.

The NRC staff finds that the licensee has adequately considered the equipment changes resulting from the EPU that affect operator ability to perform required functions.

Changes on the Safety Parameter Display System (SPDS)

The licensee will review the analog and digital inputs to the SPDS to determine the effects of the EPU, including any changes to monitored points, calculations, and alarm setpoints. The licensee will also examine changes to the EOP curves and limits for required updates on the SPDS. The licensee will complete all changes to the SPDS prior to EPU implementation, and any changes made will be included as part of the operator training program for the EPU. Based on these actions, the staff finds the proposed changes to the SPDS to be acceptable.

Changes to the Operator Training Program and the Control Room Simulator

The licensee indicated that recommendations for operator training and simulator changes and a final determination of operator training needs will be made following a review of EPU modifications and key procedure changes. This process would be consistent with the VYNPS training program for selection of modifications for operator training. Specifically, the training will cover areas such as plant modifications, procedure changes, startup test procedures, as well as changes to parameters, setpoints, scales, and systems. In addition, the licensee will revise existing lesson plans to reflect the EPU changes. The licensee will develop lesson plans and complete operator classroom training in the training phase prior to EPU implementation

With regard to the simulator, the licensee will provide simulator training prior to operation of the unit at EPU conditions. The simulator training, like the classroom training, will be completed during the last training phase prior to EPU implementation. The simulator training will include normal operating procedure actions required to achieve the EPU power level, power ascension testing, and comparisons of plant conditions between current and EPU power levels. Other topics that will be included are RFP trip and recirculation pump runback, and selected transients and accidents. The licensee will install changes on the simulator prior to EPU implementation. The installation will include hardware changes for the new digital bar graph indicators for the MS and FW flow, new scale and chart paper for the MS/FW flow recorder, and software updates for changes in modeling due to EPU. Any setpoint changes will also be reflected on the simulator.

The licensee indicated that acceptance testing of the simulator will be conducted to benchmark its performance and will be implemented in accordance with ANSI/ANS 3.5-1998. The performance of the simulator will be validated against the expected EPU response and then against operating data collected during EPU implementation and start-up testing. Based on the results of the validation, the licensee will then make any necessary adjustments to the simulator model.

The NRC staff is satisfied, based on the above licensee actions, that the licensee has proposed an acceptable plan for developing and implementing a satisfactory training program, including simulator training, and to make the necessary modifications to the simulator, for the proposed EPU.

Conclusion

The NRC staff has reviewed the changes to operator actions, human-system interfaces, procedures, and training required for the proposed EPU and concludes that the licensee has (1) appropriately accounted for the effects of the proposed EPU on the available time for operator actions and (2) taken appropriate actions to ensure that operator performance is not adversely affected by the proposed EPU. The NRC staff further concludes there is reasonable assurance that the licensee will continue to meet the requirements of draft GDC-11, 10 CFR 50.120, and 10 CFR Part 55 following implementation of the proposed EPU. Therefore, the NRC staff finds the licensee's proposed EPU acceptable with respect to the human factors aspects of the required system changes.

2.12 Power Ascension and Testing Plan

2.12.1 Approach to EPU Power Level and Test Plan

Regulatory Evaluation

The purpose of the EPU test program is to demonstrate that SSCs will perform satisfactorily in service at the proposed EPU power level. The test program also provides additional assurance that the plant will continue to operate in accordance with design criteria at EPU conditions. The NRC staff's review included an evaluation of: (1) plans for the initial approach to the proposed maximum licensed thermal power level, including verification of adequate plant performance, (2) transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased maximum licensed thermal power level, and (3) the test program's conformance with applicable regulations. The NRC's acceptance criteria for the proposed EPU test program are based on 10 CFR Part 50, Appendix B, Criterion XI, which requires establishment of a test program to demonstrate that SSCs will perform satisfactorily in service. Specific review criteria are contained in SRP Section 14.2.1.

Technical Evaluation

SRP 14.2.1 Section III.A
Comparison of Proposed EPU Test Program to the Initial Plant Test Program

SRP 14.2.1 Section III.A, specifies the guidance and acceptance criteria which the licensee should use to compare the proposed EPU testing program to the original power-ascension test program performed during initial plant licensing. The scope of this comparison should include: (1) all initial power-ascension tests performed at a power level of equal to or greater than 80% of the original licensed thermal power level; and (2) initial test program tests performed at lower power levels if the EPU would invalidate the test results. The licensee shall either repeat initial power-ascension tests within the scope of this comparison or adequately justify proposed test deviations. The following specific criteria should be identified in the EPU test program:

- all power-ascension tests initially performed at a power level of equal to or greater than 80% of the original licensed thermal power level;

- all initial test program tests performed at power levels lower than 80% of the original licensed thermal power level that would be invalidated by the EPU; and,

- differences between the proposed EPU power-ascension test program and the portions of the initial test program identified by the previous criteria.

The NRC staff reviewed Section 13.5 of the VYNPS UFSAR which described the startup and power test program performed during initial plant operation to demonstrate that the station was capable of operating safely and satisfactorily. Additional information reviewed by the staff included supplemental information provided in response to staff RAIs, applicable sections of the TSs, and the UFSAR. The following EPU test plan information was also reviewed:

- UFSAR, Section 13.5, "Startup and Power Test Program," provides an overview of the initial power ascension test program from initial fuel loading through 100% power.

- VYNPS Startup Test Report, submitted to the Atomic Energy Commission (AEC) on May 2, 1974 (Reference 58), contains the results of the Startup Test Program performed through 75% power. Results of the physics test, thermal-hydraulic and system dynamic performance tests were presented.

- Attachment 3 of Entergy letter BVY 03-80, dated September 10, 2003 (Reference 1), describes the modifications and tests necessary to support the EPU, provides a Power Ascension Test Plan matrix that specified expected EPU testing at different power levels, and also provides a comparison of initial startup testing and planned EPU testing.

- Attachment 4 of Entergy letter BVY 03-80, dated September 10, 2003 (Reference 1), provides the Power Uprate Safety Analysis Report (PUSAR) which summarizes the results of the safety analyses and evaluations performed specifically for the VYNPS EPU. A non-proprietary version of the PUSAR is provided as Attachment 6 to Reference 1.

- Attachment 7 of Entergy letter BVY 03-80, dated September 10, 2003 (Reference 1), describes the basis for determining the exclusion of transient tests (e.g., main steam line isolation valve closure) for the EPU. This attachment references NRC-approved GE Nuclear Energy, "Constant Pressure Power Uprate," Licensing Topical Report NEDC-33004P-A, dated July 2003 (Reference 51).

- Attachment 1 of Entergy letter BVY 03-98, dated October 28, 2003 (Reference 4), provides an update to Attachment 3 of the September 10, 2003, submittal which addressed Entergy's EPU testing and modification plans. The attachment also provides a discussion of the aggregate impact of the EPU modifications to the dynamic plant response.

- Attachment 2 of Entergy letter BVY 03-98, dated October 28, 2003 (Reference 4), provides an update to Attachment 7 of the September 10, 2003, submittal which provided the justification for an exception to performance of large transient testing.

- Attachment 3 of Entergy letter BVY 03-98, dated October 28, 2003 (Reference 4), provides an update to Attachment 1 of the October 1, 2003, submittal which is a review matrix that cross-references the criteria of NRC review standard RS-001 for EPUs with the information in the PUSAR and the NRC-approved generic topical report for a CPPU.

- Attachment 2 of Entergy letter BVY 04-008, dated January 31, 2004 (Reference 6), provides additional information in response to an NRC RAI.

- Attachment 1 of Entergy letter BVY 04-009, dated January 31, 2004 (Reference 5), provides additional information requested by the NRC staff in a letter dated December 15, 2003 (Reference 59). The NRC letter provided the status of the staff's acceptance review of the EPU application and identified areas where additional details were needed to consider the application acceptable.

- Attachment 2 of Entergy letter BVY 04-058, dated July 2, 2004 (Reference 9), provides additional information in response to an NRC RAI.

- Entergy letter BVY 04-109, dated October 7, 2004 (Reference 20), provides additional information regarding comparisons between the initial startup testing performed and the planned EPU testing. The letter also provides historical testing information and additional justification for the proposed EPU power ascension test program.

- Entergy letter BVY 05-084, dated September 14, 2005 (Reference 34), provides a proposed license condition and associated steam dryer power ascension testing plan that would be implemented during initial power ascension under EPU conditions. Power ascension testing associated with the steam dryer is evaluated in SE Section 2.2.6.

- Entergy letter BVY 05-017, dated February 24, 2005 (Reference 24), provides additional information in response to an NRC staff RAI regarding additional justification for the proposed elimination of large transient testing upon implementation of the EPU.

- Entergy letter BVY 05-046, dated April 22, 2005 (Reference 29), provides additional information in response to an NRC staff RAI related to the licensee's power ascension and testing program.

- Entergy letter BVY 05-072, dated August 1, 2005 (Reference 31), provides additional information in response to an NRC staff RAI related to condensate and feedwater system testing.

- Entergy letter BVY 05-083, dated September 10, 2005 (Reference 33), provides additional information in response to an NRC staff RAI related to condensate and feedwater system testing.

The NRC staff found that all tests described in the initial startup test program were listed in Attachment 2, Table 1 of Reference 20 and were compared to the proposed EPU testing program. In addition, a licensee evaluation of the initial test program found no examples of tests performed at lower than 80% of the original licensed thermal power level that would be invalidated by the EPU. The staff agreed with the licensee determination in that regard.

The licensee's EPU power ascension test plan does not include performing large transient tests (e.g., main steam isolation valve closure and generator load rejection). The justification for not performing such tests was presented by the licensee in Attachment 2 of Reference 20. This issue is further discussed below as part of the evaluation of SRP 14.2.1, Section III.C.

The following testing will be performed during the power ascension steps of the EPU:

- Testing will be performed in accordance with the TS Surveillance Requirements on instrumentation that is re-calibrated for EPU conditions. Overlap between the IRM and APRM will be assured.

- Steady-state data will be taken at points from 90% up to 100% of the pre-EPU rated thermal power (RTP) so that system performance parameters can be projected for EPU power before the pre-EPU RTP is exceeded.

- EPU power increases above 100% pre-EPU RTP will be made along an established flow control/rod line in increments of equal to or less than 5% of pre-EPU RTP. Steady-state operating data, including fuel thermal margin, will be taken and evaluated at each step. Routine measurements of reactor and system pressures, flows and vibration will be evaluated from each measurement point, prior to the next power increment. Radiation measurements will be made at selected power levels to ensure the protection of personnel.

- Control system tests will be performed for the reactor feedwater/reactor water level controls and pressure controls. These operational tests will be made at the appropriate plant conditions for that test at each of the power increments, to show acceptable adjustments and operational capability.

- Testing will be done to confirm the power level near the turbine first-stage scram bypass setpoint.

The NRC staff concludes through comparison of the documents referenced above and a review of the initial startup and test program described in Section 13.5 of the UFSAR, that the proposed EPU test program adequately identified: (1) all initial power ascension tests performed at a power level of equal to or greater than 80% of the original licensed thermal

power level, and (2) all initial test program tests performed at power levels lower than 80% of the original licensed thermal power level that would be invalidated by the EPU.

SRP 14.2.1 Section III.B
Post Modification Testing Requirements for Functions Important to Safety Impacted by EPU-Related Plant Modifications

SRP 14.2.1, Section III.B, specifies the guidance and acceptance criteria which the licensee should use to assess the aggregate impact of EPU plant modifications, setpoint adjustments, and parameter changes that could adversely impact the dynamic response of the plant to anticipated operational occurrences (AOOs). AOOs include those conditions of normal operation that are expected to occur one or more times during the life of the plant and include events such as loss of all offsite power, tripping of the main turbine generator set, and loss of power to all reactor coolant pumps. The EPU test program should adequately demonstrate the performance of SSCs important to safety that meet all of the following criteria: (1) the performance of the SSC is impacted by EPU-related modifications; (2) the SSC is used to mitigate an AOO described in the plant-specific design basis; and, (3) involves the integrated response of multiple SSCs. The following should be identified in the EPU test program as it pertains to the above paragraph:

- plant modifications and setpoint adjustments necessary to support operation at EPU conditions, and

- changes in plant operating parameters (such as reactor coolant temperature, pressure, T_{ave}, reactor pressure, flow, etc.) resulting from operation at EPU conditions.

The NRC staff reviewed the planned EPU modifications described in Attachment 3 of Reference 1. The attachment describes the modifications and tests necessary to support the EPU. It also provides a Power Ascension Test Plan matrix that specifies expected EPU testing at different power levels, and a comparison of the initial startup testing performed in 1972 and the planned EPU testing. The licensee stated that it has evaluated the modifications necessary to support the EPU and determined that they do not constitute a material alteration to the plant, as discussed in 10 CFR 50.92. The following modifications and post-modification test descriptions were identified by the licensee and reviewed by the NRC staff:

- Main turbine modifications to include new control valve settings; modifications to turbine control and overspeed setpoint for EPU conditions; and replacement of the low pressure turbine 8th stage diaphragms. High pressure turbine testing to include tests for overspeed, control and stop valves, and as-found and as-left performance testing.

- Main turbine cross-around relief valve discharge piping to be modified to accommodate higher capacity relief valves. Relief valves will be bench tested prior to installation.

- Main generator system will be upgraded for EPU conditions. Manufacturer to perform applicable electrical testing of windings. The cooling hydrogen system will be modified to include upgraded coolers and performance monitoring testing performed.

- Isolation phase bus duct cooling system will be modified to remove bus duct heat under EPU conditions and performance monitoring testing performed.

- Replacement of several high pressure feedwater heaters and testing to include demonstration of thermal performance, pressure, radiography, and magnetic particle.

- Steam dryer modifications needed to maintain dryer structural integrity at EPU conditions. Performance monitoring performed for dryer cover plate integrity includes checks for RPV water level and main steam line flow indication, steam dome pressure, and moisture carryover.

- Specific NSSS/balance-of-plant instruments upgraded for EPU conditions with associated testing for instrument rescaling, calibration and function.

- Install/remove flow induced vibration (FIV) instrumentation; collect and analyze FIV background and EPU data.

- Modifications to the main condenser to reduce the effects of FIV; perform tube leak testing.

- Modifications made to NSSS and torus piping supports; perform non-destructive examination on welds (i.e., liquid penetrant, magnetic particle).

- RHR service water system pump motor oil coolers to be modified to increase water flow to the coolers; perform inspections of piping (i.e., visual, particle, ultrasonic flow, and in-service inspection).

- Reactor recirculation (RR) system modifications which permit continued reactor power operation by pumps running back to a preset demand if the reactor is operating at or greater than a predetermined power level and one feedwater pump trips. Testing to be performed with breakers in "test position" and RR system not operating.

- Condensate demineralizer filtered bypass strainer will be installed to permit one demineralizer to be removed under EPU conditions. Once installed, flow rates will be monitored under various EPU conditions.

- EQ upgrades to include re-routing of electrical feed to SRV monitor to new breaker. Voltage and meggar testing to be performed.

- Install more efficient cooling tower fan blades and upgraded higher performance drive motors.

- Core spray and RHR pump seals were evaluated for possible replacement. As discussed in SE Section 2.2.4.2, the seals were requalified for EPU conditions and did not need to be replaced. Leak check testing to be performed at pump-rated conditions.

- Feedwater system pump modifications to include the addition of two sequential levels of low suction pressure trips at various time delays to ensure only one pump trips at a time. Normal modification testing, with breakers in "test" position, to be performed.

The licensee stated that evaluations of the actual test results may identify the need for additional tests or the revision of the tests planned and therefore, the final test plan may be revised. The NRC staff also reviewed the EPU modification aggregate impact analysis, submitted by the licensee in Reference 4, which concluded that there is no adverse impact to the dynamic response of the plant to anticipated initiating events as a result of the proposed plant modifications.

The NRC staff concludes, based on review of each identified modification, the associated post-maintenance test, and the basis for determining the appropriate test, that the EPU test program will adequately demonstrate the performance of SSCs important to safety and included those SSCs: (1) impacted by EPU-related modifications; (2) used to mitigate an AOO described in the plant design basis; and (3) supported a function that relied on integrated operation of multiple systems and components. Additionally, the staff concludes that the proposed test program adequately identified plant modifications necessary to support operation at the EPU power level, and that there were no unacceptable system interactions because of proposed modifications to the plant.

SRP 14.2.1 Section III.C
Use of Evaluation To Justify Elimination of Power-Ascension Tests

Draft SRP 14.2.1, Section III.C, specifies the guidance and acceptance criteria that the licensee should use to provide justification for a test program that does not include all of the power-ascension testing that would normally be considered for inclusion in the EPU test program pursuant to the review criteria of SRP 14.2.1, Sections III.A and III.B. The proposed EPU test program shall be sufficient to demonstrate that SSCs will perform satisfactorily in service. The following factors should be considered, as applicable, when justifying elimination of power-ascension tests:

- previous operating experience;

- introduction of new thermal-hydraulic phenomena or identified system interactions;

- facility conformance to limitations associated with analytical analysis methods;

- plant staff familiarization with facility operation and trial use of operating and emergency operating procedures;

- margin reduction in safety analysis results for AOOs;
- guidance contained in vendor topical reports; and
- risk implications.

The NRC staff reviewed the licensee's justification, in Attachment 2 of Reference 20, for not re-performing certain original startup tests. The attachment provides summaries from historical startup testing records and further justifies not performing certain startup tests during EPU power ascension testing. This information supplemented the bases for the proposed testing program provided in Reference 4. The EPU power ascension test plan does not include all of the power ascension testing that would typically be performed during initial startup of a new plant. The following factors were applied by the licensee in determining which tests may be excluded from EPU power ascension testing:

- Previous operating experience has demonstrated acceptable performance of SSCs under a variety of steady state and transient conditions.
- The effects of the VYNPS EPU are in conformance with the criteria of the NRC-approved GE CPPU Licensing Topical Report NEDC-33004P-A (Reference 51). Because the EPU is a constant pressure power uprate, the effects on SSCs due to changes in thermal-hydraulic phenomena are limited.
- Most of the plant modifications associated with the EPU were installed and tested during the spring 2004 refueling outage and subsequent restart. Therefore, modified plant equipment has been in service since that time and plant staff familiarization with changes in plant operation as a result of the modifications has occurred.

The following is a brief justification provided by the licensee with respect to the startup tests that will not be re-performed as part of the EPU power ascension program:

- STP-11, LPRM Calibration. The test is not required to be re-performed since calibration of LPRMs, which is maintained by TSs, is not affected by the EPU.
- STP-13, Process Computer. The test is not required to be re-performed since operation of the process computer is not affected by the EPU. Plant procedures maintain the accuracy of the process computer.
- STP-20, Steam Production. The test is not required to be re-performed since it was only applicable for initial plant startup to demonstrate warranted capabilities.
- STP-21, Response to Control Rod Motion. The test is not required to be re-performed since operation at EPU power increases the upper end of the power operating domain, which does not significantly or directly affect the manner of operating or response of the reactor at lower power levels.

- STP-25, Main Steam Isolation Valves (MSIVs). In accordance with VYNPS TS 4.7.D, each MSIV is tested at least once per quarter by tripping each valve and verifying the closure time. As discussed in Attachment 7 of Reference 1, one of the licensee's justifications for not performing large transient testing is that the initial startup test involving simultaneous closure of all MSIVs would result in an unnecessary and undesirable transient cycle on the primary system which will not likely reveal unforeseen equipment issues related to operation at EPU conditions.

- STP-27, Turbine Trip, and STP-28, Generator Trip. These large transient tests were evaluated by the licensee for exception from EPU power ascension testing in accordance with Attachment 7 of Reference 1. A discussion of the NRC staff's review of the licensee's justification is provided below.

- STP-29, Recirculation Flow Control. Section 3.6 of the VYNPS PUSAR documents that the plant-specific system evaluation of the reactor recirculation system performance at CPPU power determined that adequate core flow can be maintained without requiring any changes to the recirculation system and with only a small increase in pump speed for the same core flow. Because the response to flow changes will be similar to that demonstrated during initial startup testing, this test is not required.

- STP-30, Recirculation System. For a one or two pump trip test at 100% power, Section 3.6 of the PUSAR indicates a CPPU that increases voids in the core during normal EPU operations requires a slight increase in recirculation drive flow to achieve the same core flow. Section 3.6 documents that the plant-specific evaluation of the reactor recirculation system performance at CPPU power determines that adequate core flow can be maintained without requiring any changes to the system or pumps and with only a small increase in their speed for the same core flow. The response to a one or two pump trip will be similar to that of original startup testing, therefore the test is not required.

- STP X-5 (90), Vibration Testing. This test obtains vibration measurements on various reactor pressure vessel internals to demonstrate the mechanical integrity of the system under conditions of FIV and to check the validity of the analytical vibration model. The licensee stated in a previous submittal associated with the steam dryer and other plant systems and components (Reference 16) that the analysis of the vessel internals at the EPU power level was performed to ensure that the design continues to comply with the existing structural requirements. Section 3.4.2 of the PUSAR states that calculations indicate that vibrations of all safety-related reactor internal components under EPU conditions are within GE acceptance criteria.

As mentioned previously in the discussion of startup tests STP-27 and STP-28, the NRC staff also reviewed Attachment 7, "Justification for Exception to Large Transient Testing," contained in Reference 1. The licensee cited industry experience at ten other domestic BWRs (EPUs up to 120% OLTP) in which the EPU demonstrated that plant performance was adequately predicted under EPU conditions. The licensee stated that one such plant, Hatch Units 1 and 2, was granted an EPU by the NRC without the requirement to perform large transient testing and

that the VYNPS and Hatch are both BWR/4 designs with Mark I containments. Hatch Unit 2 experienced an unplanned event that resulted in a generator load reject from 98% of uprated power in the summer of 1999. As noted in Southern Nuclear Operating Company's licensee event report (LER) 1999-005, no anomalies were seen in the plant's response to this event. In addition, Hatch Unit 1 has experienced a turbine trip and a generator load reject event subsequent to its uprate, as reported in LERs 2000-004 and 2001-002. Again, the behavior of the primary safety systems was as expected indicating that the analytical models being used are capable of modeling plant behavior at EPU conditions.

The licensee also provided information regarding transient testing for the Leibstadt (i.e., KKL) plant which was performed during the period from 1995 to 2000. Uprate testing was performed at 3327 MWt (i.e., 110.5% OLTP) in 1998, 3420 MWt (i.e., 113.5% OLTP) in 1999, and 3515 MWt in 2000. Testing for major transients involved turbine trips at 110.5% OLTP and 113.5% OLTP and a generator load rejection test at 104.2% OLTP. The testing demonstrated the performance of the equipment that was modified in preparation for the higher power levels. These transient tests also provided additional confidence that the uprate analyses consistently reflected the behavior of the plant. Another factor used by the licensee to evaluate the need to conduct large transient testing for the EPU were actual plant transients experienced at the VYNPS. Generator load rejections from 100% current licensed thermal power, as discussed in VYNPS LERs 91-005, 91-009, and 91-014, produced no significant anomalies in the plant's response to these events. Additionally, the licensee indicated that transient experience for a wide range of power levels at operating BWRs has shown a close correlation of the plant transient data to the predicted response.

The NRC staff also reviewed the licensee's technical justification for not performing a loss of turbine generator and offsite power test, which was originally performed at approximately 20% of CLTP. The licensee stated that under emergency operations/distribution (emergency diesel generator) conditions, the AC power supply and distribution components are considered adequate and their evaluation assures an adequate AC power supply to safety-related systems. The TSs and approved plant procedures govern the testing of the safety-related AC distribution system, including loss of offsite power tests.

The power ascension test program is relied upon as a quality check to: (a) confirm that analyses and any modifications and adjustments that are necessary for proposed EPUs have been properly implemented, and (b) benchmark the analyses against the actual integrated performance of the plant thereby assuring conservative results. This is consistent with 10 CFR Part 50, Appendix B, which states that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate calculational methods, or by the performance of a suitable testing program; and requires that design changes be subject to design control measures commensurate with those applied to the original plant design (which includes power ascension testing).

SRP 14.2.1 specifies that the EPU test program should include steady-state and transient performance testing sufficient to demonstrate that SSCs will perform satisfactorily at the requested power level and that EPU-related modifications have been properly implemented.

The SRP provides guidance to the staff in assessing the adequacy of the licensee's evaluation of the aggregate impact of EPU plant modifications, setpoint adjustments, and parameter changes that could adversely impact the dynamic response of the plant to anticipated operational occurrences.

The NRC staff's review is intended to ensure that the performance of plant equipment important to safety that could be affected by integrated plant operation or transient conditions is adequately demonstrated prior to extended operation at the requested EPU power level. Licensees may propose a test program that does not include all of the power-ascension testing that would normally be included in accordance with the guidance provided in the SRP provided each proposed test exception is adequately justified. If a licensee proposes to omit a specified transient test from the EPU testing program based on favorable operating experience, the applicability of the operating experience to the specific plant must be demonstrated. Plant design details (such as configuration, modifications, and relative changes in setpoints and parameters), equipment specifications, operating power level, test specifications and methods, operating and emergency operating procedures; and adverse operating experience from previous EPUs must be considered and addressed.

Entergy's test program primarily includes steady-state testing with some minor load changes, and no large-scale transient testing is proposed. In a letter dated December 21, 2004 (Reference 60), the NRC staff requested that Entergy provide additional information (including performance of transient testing that will be included in the power ascension test program) that explains in detail how the proposed EPU test program, in conjunction with the original VYNPS test results and applicable industry experience, adequately demonstrates how the plant will respond during postulated transient conditions following implementation of the proposed EPU given the revised operating conditions that will exist and plant changes that are being made. In letters dated July 27, and September 7, 2005 (Reference 60 and 61), the NRC staff requested that the licensee provide additional information regarding the need for condensate and feedwater system transient testing. The results of the staff's review of this issue and the need for a license condition is discussed in SE Section 2.5.4.4.

Based on its review of the information provided by the licensee, as described above, the NRC staff concludes that in justifying test eliminations or deviations, other than the condensate and feedwater system testing discussed in SE Section 2.5.4.4, the licensee adequately addressed factors which included previous industry operating experience at recently uprated BWRs, plant response to actual turbine and generator trip tests at other plants, and experience gained from actual plant transients experienced in 1991 at the VYNPS. From the EPU experience referenced by the licensee, it can be concluded that large transients, either planned or unplanned, have not provided any significant new information about transient modeling or actual plant response. As such, the staff concludes that there is reasonable assurance that the VYNPS SSCs will perform satisfactorily in service under EPU conditions. The staff also noted that the licensee followed the NRC staff approved GE topical report guidance which was developed for the VYNPS EPU licensing application.

SRP 14.2.1 Section III.D
Evaluate the Adequacy of Proposed Transient Testing Plans

SRP 14.2.1, Section III.D, specifies that a licensee's EPU amendment request should include plans for the initial approach to the increased EPU power level and testing that will be used to verify that the reactor plant operates within the values of EPU design parameters. The test plan should assure that the test objectives, test methods, and the acceptance criteria are acceptable and consistent with the design basis for the facility. The predicted testing responses and acceptance criteria should not be developed from values or plant conditions used for conservative evaluations of postulated accidents. During testing, safety-related SSCs relied upon during operation shall be verified to be operable in accordance with existing TS and Quality Assurance program requirements. The following should be identified in the EPU test program:

- the method in which the initial approach to the uprated EPU power level is performed in an incremental manner including steady-state power hold points to evaluate plant performance above the original full-power level;

- appropriate testing and acceptance criteria to ensure that the plant responds within design predictions including development of predicted responses using real or expected values of items such as beginning-of-life core reactivity coefficients, flow rates, pressures, temperatures, response times of equipment, and the actual status of the plant;

- contingency plans if the predicted plant response is not obtained; and

- a test schedule and sequence to minimize the time untested SSCs important to safety are relied upon during operation above the original licensed full-power level.

The NRC staff reviewed Reference 4 which addresses the licensee's EPU testing and modification plans. The staff found that the licensee adequately addressed EPU operating experience for similar designed plants (Hatch Units 1 and 2) in determining the current proposed test plan for the VYNPS.

The VYNPS Startup Test Report (Reference 58), submitted to the AEC on May 2, 1974, discussed the results of the initial VYNPS Startup Test Program. The report included the results of the physics tests, thermal-hydraulic and system dynamic performance tests. The complete Startup Test Program is divided into several phases. Phase 3, "Power Ascension Tests," included low power through high power ascension testing and fuel warranty run performed at power levels of 15%-100%. However, because of fuel hydriding effects in the reactor core which caused unexpectedly high stack gas release rates, testing was suspended by VYNPS at 75% power. The report stated that the results of testing above 75% power would be submitted as an addendum to the report at a later time. However, the results of testing above 75% power were not formally reported to the AEC or NRC.

As noted in the May 2, 1974 report, the VYNPS Joint Test Group, which had responsibility for initial startup testing, was dissolved in July 1973. The licensee stated that subsequent tests were conducted either in accordance with startup test or plant procedures and that those tests that constitute routine surveillances continue to be performed in accordance with plant operating procedures. Startup testing at 100% power was subsequently performed and the full power warranty run was completed in February 1975. Because VYNPS did not submit a final Startup Test Report to document initial testing at 100% power, the staff requested that the licensee submit sufficient documentation to demonstrate that such high power ascension testing had occurred.

In Attachment 2 to Reference 20, the licensee provided additional information regarding the startup test program. The submittal provided historical information regarding initial plant testing conducted at greater than or equal to 80% OLTP, but not planned for the EPU, consistent with Section III.A.1 of SRP 14.2.1. In Table 1, the licensee provided a comparison of the power ascension tests that were scheduled to be performed during initial startup testing at power levels greater than or equal to 80% of OLTP level versus those planned for EPU testing. The table supplemented and expanded on the information previously provided in the application regarding the eleven startup tests (out of a total of eighteen) that will not be entirely re-performed as part of the EPU power ascension testing program. For those tests where no EPU power ascension test is planned, the affected SSC may still be subject to periodic testing in accordance with plant TSs or plant procedures.

The licensee will conduct seven of the original eighteen startup tests at the time of implementation of the EPU. The tests will be conducted in accordance with the NRC-approved generic EPU guidelines of GE Licensing Topical Report NEDC-32424P-A (Reference 63) to demonstrate the capability of plant systems to perform their design functions under uprated conditions. The tests will be done in accordance with a site-specific test procedure developed by the licensee and will follow established controls and procedures that have been revised to reflect the uprated conditions. The tests consist essentially of steady-state testing between 90% and 100% of currently licensed thermal power. Several sets of data will be obtained between 100% and 120% current power with no greater than a 5% power increment between data sets.

Section 10.4 of the PUSAR provides additional information relative to power uprate testing and describes a standard set of tests that have been established for the initial power ascension steps. These tests, which supplement the normal TS testing requirements, are discussed above as part of the evaluation for SRP 14.2.1 Section III.A.

The NRC staff concludes that the proposed test plan will adequately assure that the test objectives, test methods, and test acceptance criteria are consistent with the design basis for the facility. Additionally, the staff concludes that the test schedule would be performed in an incremental manner, with appropriate hold points for evaluation, and contingency plans would be utilized if the predicted plant response is not obtained.

BOP Systems Testing Review

In light of the considerations that are discussed in SRP 14.2.1 for power uprates, the NRC staff requested additional information regarding the VYNPS EPU power ascension testing for BOP systems. In response to the staff's request, the licensee provided supplemental information in References 24, 29, 31, and 33. The staff reviewed the information that was provided and found that the licensee adequately addressed the considerations discussed in SRP 14.2.1 with the exception of testing of the condensate and feedwater system as discussed below.

The NRC staff's review of a licensee's power ascension and testing plan for BOP systems focuses primarily on two areas. One area deals with the capability of the turbine bypass control system to discharge steam to the main condenser as assumed in the turbine generator load reject and turbine trip transient analyses. Because the licensee is not proposing to credit additional steam bypass capacity beyond what was previously assumed, transient testing for the purpose of demonstrating the capacity of the turbine bypass control system is not required.

The other area of the NRC staff's review focuses on transient testing that may be needed as a consequence of BOP modifications that are necessary for implementing a proposed EPU. In this regard and as discussed in SE Section 2.5.4.4, the staff questioned the adequacy of the licensee's transient testing for the condensate and feedwater system modifications that are being made. Contrary to the licensee's position, the staff concluded that transient testing of the condensate and feedwater system is required in order to confirm acceptable performance during EPU operation. Therefore, a license condition that requires condensate and feedwater system transient testing has been established by the staff for this purpose. With respect to large transient testing, in June 2004, VYNPS experienced a generator load reject event at 100% of CLTP (i.e., about 80% of the proposed EPU power level), with many of the EPU modifications implemented. The licensee found no significant anomalies in plant response to this incident (Reference 24, Attachment 1, response to RAI SPLB-A-10). Therefore, the staff found that this transient satisfied the objectives of a large transient test in that the transient demonstrated that the steam and power conversion systems, as modified for EPU operation, performed as designed.

Based on a review of the information that was provided, the NRC staff has determined that, with (a) the limited scope of EPU modifications for BOP systems, (b) no introduction of new credible thermal-hydraulic phenomena, and (c) past plant experience, combined with (d) the required demonstration of acceptable plant performance during the power ascension test program, including the condensate and feedwater system testing that is required by a license condition as discussed in Section 2.5.4.4, the licensee's proposed test program provides reasonable assurance that BOP systems will function as designed.

Conclusion

The NRC staff has reviewed the EPU test program, including plans for the initial approach to the proposed maximum licensed thermal power level, transient testing necessary to demonstrate that plant equipment will perform satisfactorily at the proposed increased

maximum licensed thermal power level, and the test program's conformance with applicable regulations. The staff concludes that the proposed EPU test program, including the testing required by the license condition discussed in SE Section 2.5.4.4, provides reasonable assurance that the plant will operate in accordance with design criteria and that SSCs affected by the proposed EPU, or modified to support the proposed EPU, will perform satisfactorily in service. Further, the staff finds that there is reasonable assurance that the EPU testing program satisfies the requirements of 10 CFR Part 50, Appendix B, Criterion XI. Therefore, the NRC staff finds the proposed EPU test program acceptable.

2.13 Risk Evaluation

2.13.1 Risk Evaluation of Extended Power Uprate

Regulatory Evaluation

A risk evaluation is conducted to: (1) demonstrate that the risks associated with the proposed EPU are acceptable and (2) determine if "special circumstances" are created by the proposed EPU. As described in Appendix D of SRP Chapter 19, special circumstances are any issues that would potentially rebut the presumption of adequate protection provided by the licensee meeting the currently specified regulatory requirements. The NRC staff's review covers the impact of the proposed EPU on core damage frequency (CDF) and large early release frequency (LERF) for the plant due to changes in the risks associated with internal events, external events, and shutdown operations. In addition, the NRC staff's review covers the quality of the risk analyses used by the licensee to support the application for the proposed EPU. This includes a review of licensee actions to address issues or weaknesses that may have been raised in previous NRC staff reviews of the licensee's individual plant examinations (IPEs) and individual plant examinations of external events (IPEEEs), or by an industry peer review. The NRC's risk acceptability guidelines are contained in RG 1.174. Specific review guidance is contained in Matrix 13 of RS-001 and its attachments.

Technical Evaluation

The NRC staff reviewed the risk evaluation submitted by the licensee (Reference 1), as supplemented by responses to the staff's request for additional information (Reference 6). In general, the licensee's risk evaluation compared the risks of the pre-EPU to the post-EPU plant design and operation. A combination of quantitative and qualitative methods was used to assess the risk impacts of the proposed EPU. The following sections provide the staff's technical evaluation of the risk information provided by the licensee.

Level 1 Internal Events Risk Evaluation

The licensee maintains a Level 1 probabilistic safety assessment (PSA) of the VYNPS that estimates the CDF due to internal initiating events (including internal floods). The risk impacts of the proposed EPU due to internal initiating events were assessed by reviewing the changes in plant design and operations resulting from the proposed EPU, mapping these changes onto

appropriate PSA elements, modifying affected PSA elements as needed to capture the risk impacts of the proposed EPU, and requantifying the PSA to determine the CDF of the post-EPU plant.

Changes in Plant Design and Operations

The proposed EPU is a CPPU that increases the RTP without changing the RPV operating pressure and temperature. In order to assess the impact of the proposed EPU on plant risk, the licensee considered the following changes in plant design and operations:

- General
  - Increase in RTP from 1593 to 1912 megawatts thermal
  - Increase in steam, feedwater, and condensate flow rates by approximately 20%
  - Use of 3-out-of-3 reactor feedwater pumps (RFPs) during power operations
  - Additional spring safety valve (SSV), needed to implement the Average Power Range Monitor/Rod Block Monitor/Technical Specifications Maximum Extended Load Line Limit Analysis (ARTS/MELLLA)
  - Transition to GE-14 fuel, which was initiated in Cycle-23
- Hardware Modifications
  - Mechanical
    - Replacement of the high pressure (HP) turbine rotor and modification of the turbine controls
    - Replacement of the main generator hydrogen cooling system heat exchangers
    - Balance of plant and nuclear steam supply system pipe support modifications
    - Modification to residual heat removal service water motor cooling piping
    - Replacement of HP feedwater heaters in both trains
    - Modifications to the isolated phase bus cooling system to provide additional cooling capacity
    - Increased condensate demineralizer flow capacity
    - Stake main condenser tubes
    - RPV steam dryer modification for structural integrity
  - Electrical
    - Rewind of main turbine generator
    - Upgrade generator disconnect switch
    - Addition of recirculation pump runback logic
    - Various instrumentation and controls component upgrades and replacements, such as the feedwater level control

- Adjustments to the VYNPS emergency operating procedures (EOPs) and severe accident management guidelines (SAMGs), to be consistent with CPPU operating conditions. The licensee stated that in almost all respects, the EOPs/SAMGs are expected to remain unchanged because they are symptom-based; however, certain parameter thresholds and graphs are dependent upon power and decay heat levels and will require slight modifications. EOP variables that play a role in the PSA and which may require adjustment for the EPU include:
  - Boron Injection Initiation Temperature
  - Heat Capacity Temperature Limit
  - Pressure Suppression Pressure Limit

- Setpoint Changes
  - Turbine overspeed
  - Turbine first-stage pressure steam scram bypass
  - Main steam line (MSL) high flow isolation

Initiating Event Frequencies

The VYNPS PSA addresses transients, loss of offsite power (LOOP), LOCAs, support system failures, internal floods, and external events.

The licensee stated that the proposed EPU is only expected to affect the frequency of the turbine trip (TT) initiating event. The frequency of this initiating event is affected because all three of the RFPs are required for power operation for the post-EPU condition. The licensee estimated that the TT frequency would increase by about 4% as a result of the proposed EPU. The NRC staff finds this change reasonable and concurs with the licensee's approach for adjusting the TT initiating event frequency.

The licensee stated that the frequency of total loss-of-feedwater (LOFW) is not expected to change as a result of the proposed EPU because failure of the RFPs is a negligible contributor to the overall frequency of this initiating event (total LOFW is dominated by other issues such as feedwater regulating valve closure). As part of the proposed EPU plant modifications, a reactor recirculation system runback modification will be installed to avoid a plant trip on loss of a condensate pump or RFP. The licensee stated that malfunction of the reactor recirculation system runback circuitry cannot cause a total loss-of-feedwater.

As a result of the proposed EPU, the plant's turbine bypass capacity will be reduced from 105% to 85% of rated steam flow. The licensee stated that the reduced capacity has no impact on the frequencies of transient initiating events because VYNPS does not use the large turbine bypass capacity to prevent a reactor trip given a load rejection event when reactor power is above approximately 30% of CLTP.

The licensee evaluated the impact of the proposed EPU on the LOOP frequency, and determined that there would be no impact. To confirm this conclusion, the NRC staff compared

the licensee's estimated LOOP frequency to data recently collected and analyzed by the NRC's Office of Nuclear Regulatory Research (RES) in response to the August 2003 Northeast Blackout. The VYNPS PSA model contains a single LOOP initiating event, whose frequency is the sum of contributions from plant-centered events, grid-related events, and weather-related events. The data obtained by RES indicates that the overall nationwide average LOOP frequency for the period after 1997 has decreased as compared to the period prior to 1997, although the contributions have changed. Specifically, the contribution from plant-centered LOOPs has decreased, the contribution from grid-related LOOPs has increased, and the contribution from weather-related LOOPs has remained unchanged. The NRC staff has also conducted a traditional engineering (deterministic) evaluation of the impact of the proposed EPU on the stability of the grid surrounding the VYNPS site. The staff's evaluation, shown in SE Section 2.3.2, concluded that, once the planned modifications are completed (including rewinding the main generator and addition of a 60 MVAR capacitor bank), the proposed EPU will not impact grid stability. The licensee noted that VYNPS currently has certain operational configurational conditions that require power reductions to maintain grid stability. The same, or similar, conditions and operations will exist once the proposed EPU has been implemented. The NRC staff concludes that the LOOP frequency used by the licensee to evaluate the risk impacts of the proposed EPU is reasonable because it is higher than the overall LOOP frequency recently estimated by RES, the staff's deterministic evaluation concluded that the proposed EPU will have no impact on grid stability, and the licensee will continue to implement actions intended to maintain grid stability.

No changes to RPV operating pressure, inspection frequencies, or primary water chemistry are planned in support of the proposed EPU. As such, no effect on LOCA frequencies due to the proposed EPU was postulated.

The licensee's EPU application indicates that FIV may cause an inadvertent SRV opening or a stuck-open SRV. In response to the NRC staff's request for additional information, the licensee concluded that the proposed EPU would not increase the frequency of an inadvertent SRV (termed "inadvertent open relief valve (IORV)" in the PSA model) based on the following information:

- Pilot leakage has been a common problem in Target Rock three-stage relief valves and has, in the industry, resulted in inadvertent valve openings and blowdown. The BWR community has concluded that inadequate simmer margin is the leading cause of pilot leakage. However, the proposed EPU will maintain the simmer margin and, therefore, there will be no change in the pilot valve seating force.

- Changes in FIV caused by the proposed EPU may affect the pilot disc's ability to maintain alignment (e.g., cocking/tilting) with the pilot seat or have a resonance effect on the pilot's pre-load and setpoint adjustment spring's natural frequency (reducing effective seating force). These vibrational effects, depending upon extent and magnitude, could then lead to an increased propensity for pilot valve leakage and thus, over time, an inadvertent valve opening. However, the solenoid assembly is not mounted directly on the air actuator but

rather remotely on a pipe support. Therefore, no FIV will be transmitted to the solenoid assembly.

- The VYNPS SRVs have not experienced any maintenance problems. During air actuator refurbishments each outage, no wearing or vibration-induced indications have been found.

- Operators would be alerted to a leaking pilot valve since they routinely monitor SRV tail pipe temperatures and an alarm is received on high SRV tail pipe temperature. VYNPS procedures provide guidance to the operators for a leaking SRV and an inadvertent opening of an SRV. Specifically, Operations Procedure OT 3121 is entered upon receiving indications of a leaking SRV. There is sufficient time for the condition to be evaluated by engineering and for operations to conduct a controlled plant shutdown, if necessary. This would preclude an inadvertent opening of the SRV.

No significant changes to support systems (e.g., instrument air, service water) are planned in support of the proposed EPU. As such, no effect on support system initiating event frequencies due to the proposed EPU were postulated.

No changes to pipe inspection scopes or frequencies are planned in support of the proposed EPU. As such, no effect on internal flooding initiator frequencies due to the proposed were postulated.

The frequency of external event initiators (e.g., seismic events, extreme winds, fires) is not linked to reactor power or operation. As such, no effect on external event initiator frequencies due to the proposed EPU were postulated.

The NRC staff concurs with the licensee's assessment of the impact of the proposed EPU on initiating event frequencies, and concludes that they should not be noticeably impacted by the proposed EPU as long as the operating ranges or limits of equipment are not exceeded. In addition, the staff notes that any changes in the initiating event frequencies following implementation of the proposed EPU would be identified and tracked under the licensee's existing performance monitoring programs and processes.

Component Failure Rates

The licensee stated that the hardware changes in support of the proposed EPU may be characterized as either replacement of components with enhanced similar components or upgrades of existing components. As a result, the component failure rates used in the post-EPU PSA model are the same as those used in the pre-EPU PSA mode, with one exception.

The licensee evaluated the impact of the proposed EPU on the probability of a stuck-open relief valve (SORV). The SRV setpoints will not be changed as a result of proposed EPU. Given the power increase of the proposed EPU, it may be postulated that the probability of an SORV given a transient initiator would increase due to an increase in the number of SRV cycles. The

licensee considered three approaches to reflect the impact of the proposed EPU on the SORV probability:

- The upper bound approach would be to increase the SORV probability by a factor equal to the increase in reactor power (i.e., by a factor of 1.2 since the proposed EPU increases CLTP by 20%). This approach assumes that the SORV probability is linearly related to the number of SRV cycles, and that the number of cycles is linearly related to the reactor power increase.

- A less conservative approach is to assume that the SORV probability is linearly related to the number of SRV cycles. However, the number of cycles is not necessarily directly related to the reactor power increase. In this case, the postulated increase in SRV cycles due to the proposed EPU would be determined by thermal hydraulic calculations (e.g., material access authorization program (MAAP) runs).

- The lower bound approach would be to assume that the SORV probability is dominated by the initial cycle and that subsequent cycles have a much lower failure rate. In this approach, the pre-EPU SORV probability could be assumed to be insignificantly changed by a postulated increase in the number of SRV cycles.

The licensee applied the second approach to modify the SORV probability for the proposed EPU. The increase in the number of SRV cycles during accident response was estimated by comparing the results of MAAP runs for isolation transient scenarios performed in support of the post-EPU risk assessment. These analyses indicated that for the post-EPU plant, the number of SRV cycles in the first couple of hours of the accident progression increases by no more than 15%. Accordingly, the licensee increased the SORV probability by 15% in the post-EPU PSA model. The NRC staff agrees with the adjustment made by the licensee, and notes that the risk insights from the post-EPU PSA would not be expected to substantially change had the more conservative (first) approach been used.

The NRC staff finds that it is reasonable to conclude that equipment reliability will not change, as long as the operating ranges or limits of the equipment are not exceeded. For equipment that is operated within its operating ranges or limits, the staff notes that the licensee's component monitoring programs, as identified above, should detect significant degradation in performance and the staff expects these programs to maintain the current reliability of the equipment.

Accident Sequence Delineation

The success criteria for the VYNPS PSA are derived based on realistic evaluations of system capability over the 24-hour mission time of the PSA analysis. The licensee stated that approximately 60 Level I MAAP runs and 6 Level 2 MAAP runs were performed in support of the VYNPS post-EPU internal events Level 1 PSA.

In order to reflect the risk impacts of the proposed EPU, the licensee changed the success criteria for the number of SRVs/SSVs required to control initial RPV overpressure during an isolation anticipated transient without scram (ATWS) scenario. The pre-EPU (i.e., without implementing ARTS/MELLLA) plant configuration includes four SRVs and two SSVs. The post-EPU plant configuration (which includes ARTS/MELLLA) has an additional SSV in order to provide additional relief capacity for the limiting ATWS transient.

As noted in the discussion of initiating events, the plant's turbine bypass capacity will be reduced from 105% to 85% of rated steam flow as a result of the proposed EPU. The licensee stated that the reduced bypass capacity does not impact ATWS sequences as modeled in the VYNPS PSA. Specifically:

- ATWS sequences with successful recirculation pump trip (RPT): following a successful RPT during an ATWS scenario, the power level is well below the 85% turbine bypass capacity, just as it is in the pre-EPU condition. Thus, no modeling changes to the PSA were necessary.

- ATWS sequences with failure of RPT: ATWS scenarios with failure of RPT are modeled as leading directly to core damage. This is a typical and reasonable industry PSA approach. There is no change in reactor trip frequency and, therefore, no modeling changes to the PSA are necessary.

The licensee has requested credit for containment accident pressure to provide adequate NPSH to the ECCS following a design-basis LOCA event and an ATWS event. Without adequate NPSH, the ECCS pumps will cavitate, perhaps substantially reducing their flow rate and discharge pressure and resulting in their inability to perform their required functions. Information provided by the licensee, as discussed in SE Section 2.6.5, indicates that the need to credit containment accident pressure arises from the conservative nature of the assumptions made in the traditional, deterministic analyses of these events. As a result, the licensee did not explicitly address the impact of the proposed containment accident pressure credit on risk in its original EPU application because of its belief that no containment accident pressure credit would be required if the NPSH analysis had been based on realistic assumptions typical of those used to formulate PSA success criteria.

From a risk perspective, the proposed containment accident pressure credit introduces a dependency between the systems used to prevent accidents and the containment, which is used to mitigate the effects of accidents. Specifically, a loss of containment integrity could prevent the containment from pressurizing and providing adequate NPSH to the ECCS pumps. As previously discussed, the VYNPS EPU application is not a risk-informed application submitted according to RG 1.174. The NRC staff's traditional engineering (deterministic) evaluation, shown in SE Section 2.6.5, concluded that the licensee's proposal to credit containment accident pressure to provide adequate NPSH to the ECCS pumps is acceptable. However, in order to better understand the risk implications of the proposed containment accident pressure credit and in accordance with the process established in SRP Chapter 19, Appendix D concerning the use of risk information in the review of non-risk informed license

amendment requests, the staff asked the licensee to provide a risk evaluation of the proposed containment accident pressure credit that addressed the five key principles of risk-informed decisionmaking contained in RG 1.174. The NRC staff also conducted a scoping risk evaluation to help confirm the licensee's risk evaluation.

The proposed containment accident pressure credit introduces uncertainty about the success criteria used to construct the PSA model. As discussed in SE Section 2.6.5, the available evidence strongly suggests that no containment accident pressure credit is required when realistic initial conditions and parameters are used to determine the available NPSH to the ECCS pumps. However, these initial conditions and parameters contain both aleatory uncertainties (e.g., changes in service water temperature caused by seasonal variations) and epistemic uncertainties (e.g., uncertainty in determining frictional head losses for a given piping configuration). In order to assess the impact of these uncertainties on risk, the licensee performed a sensitivity analysis (Reference 39, 40, and 44) of the proposed containment accident pressure credit. Specifically, the licensee modified its PSA model by assuming that the proposed containment accident pressure credit is needed and compared the results of this modified PSA model to the results of the post-EPU risk evaluation, which assumes that containment accident pressure credit is not required. The NRC staff used an identical approach in conducting its scoping risk evaluation. Section 2.2.5.5 of RG 1.174 states that sensitivity analyses may be used to assess the impact of modeling uncertainties.

The licensee's sensitivity analysis was based on the PSA model for internal events (including internal flooding). In the sensitivity analysis, it was assumed that a core-damage accident would occur if all of the following events occur:

- An accident occurs that discharges reactor coolant into the containment, adding heat to the suppression pool. The accident may discharge reactor coolant to the suppression pool either directly (e.g., a LOCA) or indirectly (e.g., a transient followed by a subsequent safety relief valve (SRV) opening).

- The low-pressure safety injection (LPSI) or core spray (CS) pumps are needed to provide reactor inventory control or decay heat removal.

- Containment integrity is lost so that the containment accident pressure is not sufficient to provide adequate NPSH to the LPSI and CS pumps.

- The operator does not initiate alternative injection sources to provide core cooling.

The licensee's sensitivity analysis considered two specific containment failure modes: pre-existing, undetected leaks through the containment and failure of the primary containment isolation system. The probability of containment leaks was estimated using EPRI Report TR-1009325 (Reference 48). This information source, which used expert elicitation to develop containment leakage probabilities, is currently under NRC staff review as a technical basis for extending Type A integrated leak rate test (ILRT) intervals up to 15 years.

The licensee's sensitivity analysis indicated that the change in CDF associated with crediting containment accident pressure to provide adequate NPSH to the ECCS pumps was approximately 6×10^{-7}/year. The change in LERF was estimated as approximately 5×10^{-8}/year.

The licensee's sensitivity analysis does not include external events such as earthquakes and internal fires. As reported in the IPEEE of VYNPS, the high confidence of low probability of failure (HCLPF) values for the reactor coolant system piping, reactor vessel supports, SRVs, and the containment are in excess of 0.3g. This indicates that the likelihood of a seismic event causing a simultaneous LOCA and failure of the containment is very small. Concerning the risk from internal fires, the IPEEE stated that fire damage to SRV cable in the reactor building and adjacent cable vaults would cause the affected SRV to fail in the de-energized, closed position. Further, the likelihood of a fire-induced hot short causing a spurious opening of an SRV was judged to be remote due to the design features of the automatic depressurization system (ADS) inhibit switch and the rated fire barrier protection. The NRC staff concludes that the licensee's determination that external event risks need not be included in the licensee's quantitative sensitivity analysis of the proposed containment accident pressure credit is reasonable and acceptable, and is consistent with RG 1.1.74 which allows the use of qualitative arguments.

The NRC staff modified the VYNPS Standardized Plant Analysis Risk (SPAR) model in order to conduct the scoping risk evaluation. SPAR models are developed by RES, and are used to support the staff's significance determination process (SDP) and accident sequence precursor (ASP) reviews. The VYNPS SPAR model was benchmarked against the plant-specific PSA maintained by the licensee in May 2003. New accident sequences were created by modifying the event tree models for all initiating events except interfacing system LOCAs. Specifically, the following event tree models were modified:

- Large-LOCA
- Medium-LOCA
- Small-LOCA
- IORV
- General transient
- Loss of condenser heat sink
- Loss of main feedwater
- LOOP
- Station blackout (SBO)
- Loss of service water
- Loss of instrument air
- Loss of medium voltage alternating current (AC) bus 3
- Loss of medium voltage AC bus 4
- Loss of direct current (DC) bus DC-1
- Loss of DC bus DC-2

The scoping risk evaluation assumed that a core-damage accident would occur if all of the following events occur:

- An accident occurs that discharges reactor coolant into the containment, adding heat to the suppression pool. The accident may discharge reactor coolant to the suppression pool either directly (e.g., a LOCA) or indirectly (e.g., a transient followed by a subsequent SRV opening).

- The LPSI or CS pumps are needed to provide reactor inventory control or decay heat removal.

- Containment integrity is lost so that the containment accident pressure is not sufficient to provide adequate NPSH to the LPSI and CS pumps.

- The operator does not initiate suppression pool cooling within 4 hours after the accident occurs.

Modifications to the interfacing system LOCA event trees were not needed because the containment is not pressurized following these types of events (leakage from the reactor coolant system is outside of the containment). Therefore, no heat is added to the suppression pool from interfacing system LOCAs and there is no need to ensure containment integrity to provide adequate NPSH to the ECCS pumps.

The NRC staff observes that a loss of containment integrity either prior to the accident (e.g., due to a pre-existing and undetected containment leak) or immediately after the accident (e.g., due to failure of the primary containment isolation system) will not cause an immediate failure of the LPSI and CS pumps because it takes time for the discharge of reactor coolant to sufficiently heat the suppression pool to the point where these pumps will cavitate. The licensee provided (Reference 24) a MAAP calculation that indicates the operator will have about 4 hours from the start of a large-LOCA to initiate suppression pool cooling and avoid pump cavitation.

The NRC staff's scoping risk evaluation considered three specific containment failure modes, whose probabilities were estimated as follows:

- Pre-existing, undetected leaks through the containment: the probability of this failure mode was obtained from the licensee's evaluation of the risk impacts of extending, on a one-time basis, the ILRT to 15 years (Reference 65). The NRC staff approved this ILRT extension on August 31, 2005 (Reference 66).

- Failure of the primary containment isolation system: the probability of this failure mode was obtained from the licensee's evaluation of the risk impacts of extending, on a one-time basis, the ILRT to 15 years as noted above.

- Failure of MSIVs to close on demand: the probability of this failure mode, including common-cause failures, was estimated using data obtained from RES.

The NRC staff's scoping risk evaluation indicated that the change in CDF associated with crediting containment accident pressure to provide adequate NPSH to the ECCS pumps was

approximately 6 x 10^{-8}/year, which is also equal to the change in LERF. The conditional containment failure probability (CCFP) increased by approximately two percentage points.

In order to put the results of the licensee's sensitivity analysis and the NRC staff's scoping risk evaluation into perspective, the staff assessed major differences between the assumptions made. This assessment showed that the success criteria used by the staff are different than the ones used by the licensee:

- The licensee assumed that the LPCI and CS pumps would fail immediately upon loss of containment integrity. In contrast, the staff's scoping risk evaluation credited initiation of suppression pool cooling within four hours after containment integrity is lost.

- The licensee gave credit for alternative injection sources whereas the staff did not. For medium LOCAs, the licensee considered use of condensate, CRD, and condensate transfer. For transients and small LOCAs, feedwater, HPCI, and RCIC were considered in addition to the alternative injection systems considered for medium LOCAs. For large LOCAs, the only alternative injection source considered was the fire water system via an intertie between the service water system and RHR Loop A.

The probability of pre-existing, undetected containment leakage used by the licensee was about two orders of magnitude lower than the value used by the NRC staff. The licensee's value was based on EPRI TR-1009325, which provides the probability of a leak as a function of the leak's size. In contrast, the staff's estimate was based on Type A ILRT test data that had been analyzed using Bayesian methods that do not explicitly consider the size of the leak when assigning its probability.

The differences in success criteria assumptions and containment leakage probabilities are competing effects, making it difficult to reconcile the different numerical results of the licensee's sensitivity analysis and the NRC staff's scoping risk evaluation. However, neither analysis fully credited all available means for preventing core damage following a LOCA and loss of containment integrity, and therefore they both provide a conservative estimate of the increase in CDF if the proposed containment accident pressure credit is assumed to affect the PSA success criteria. The staff notes that the results of the licensee's sensitivity analysis are higher than the results of the staff's scoping risk evaluation. It should be noted that even if the change in CDF determined by the licensee's sensitivity analysis is added to the change in CDF due to other EPU-related causes, the proposed EPU meets the numerical risk acceptance guidelines in RG 1.174.

The NRC staff concludes that crediting containment accident pressure to provide adequate NPSH to the ECCS pumps does not create "special circumstances" that rebut the presumption of adequate protection provided by the licensee meeting the currently specified regulatory requirements, because comparison of the results of the scoping risk evaluation to the numerical risk acceptance guidelines in RG 1.174 indicates that implementing the credit results in a very small risk increase. The staff notes that the above conclusion is specific to VYNPS, and is not a generic conclusion that can be applied to other nuclear power plants.

With respect to accident sequence modeling (including event tree and systems analysis), the licensee stated that the proposed EPU does not change the plant configuration and operation in a manner such that new accident sequences or changes to existing accident scenario progressions result. The NRC staff observes that this conclusion is reasonable, given the changes to accident sequence success criteria identified by the licensee. The staff believes that the licensee could have utilized the PSA to assess the risk impacts of crediting containment accident pressure to provide NPSH to the ECCS pumps, and notes that doing so would have necessitated changes to the PSA logic model. Had this been a risk-informed application under RG 1.174, the staff would have pursued this matter further with the licensee in order to ensure that the post-EPU PSA was used to assess the risk impacts of the proposed containment accident pressure credit. However, as previously discussed, the staff has concluded that crediting containment accident pressure at VYNPS to provide NPSH to the ECCS pumps does not create "special circumstances" that rebut the presumption of adequate protection provided by the licensee meeting the currently specified regulatory requirements. Therefore, the NRC staff concludes that the accident sequence modeling used by the licensee in the post-EPU PSA is technically adequate to support the EPU application. The staff expects the licensee to make appropriate changes to the PSA model as required to address the risk impact of crediting containment accident pressure before submitting any future risk-informed license amendment requests under RG 1.174.

Operator Actions and LOOP Recovery

The VYNPS risk profile is dependent on the operating crew's actions for successful accident mitigation. The success of these actions is in turn dependent on a number of performance shaping factors. The performance shaping factor that is principally influenced by the proposed EPU is the time available to detect, diagnose, and perform required actions. The higher power level results in reduced times available for some actions. To quantify the potential effect of this performance-shaping factor, deterministic thermal-hydraulic calculations using the MAAP computer code were used. The licensee also examined the impact of the proposed EPU on the man-machine interface performance shaping factor. Changes to be made to the control room displays for the proposed EPU are:

- MSL flow indicators replaced with digital units
- Feedwater (FW) flow indicators replaced with digital units
- Steam/FW flow recorder re-scaled
- Condensate flow recorder re-scaled

The licensee stated that none of these control room display changes affect the quantification of human error probabilities (HEPs) in the VYNPS PSA.

Not all operator actions in the VYNPS PSA have a significant effect on the results. The licensee performed a screening analysis to identify those operator actions that have a significant effect on the PSA results. The licensee's screening process was performed against the following criteria:

- Fussell-Vesely (with respect to CDF) importance measure ≥ 5E-3
- Risk achievement worth (with respect to CDF) importance measure ≥ 2.0
- Fussell-Vesely (with respect to LERF) importance measure ≥ 5E-3
- Risk achievement worth (with respect to LERF) importance measure ≥ 2.0
- Time critical (≤ 30 minutes available) action

Of the 59 post-initiator actions in the VYNPS PSA, 18 screened out and 41 were retained for explicit re-assessment in the VYNPS post-EPU risk assessment. The NRC staff reviewed the licensee's screening process, including the operator actions that were screened out, and concludes that the licensee's process to identify significant operator actions affected by the proposed EPU is reasonable.

The licensee recalculated the HEPs for the operator action identified by the screening process using the same human reliability analysis methods used in the VYNPS PSA. The NRC staff concludes that the changes made to the HEPs reasonably reflect the reductions in the times available for the operators to perform the necessary actions under post-EPU conditions and they are based on methodologies previously accepted by the staff in various risk-informed license amendment requests.

The licensee considered the dependencies among operator actions for the post-EPU PSA. The approach used to judge the level of dependence between operator actions is based on dependency level categories and conditional probabilities developed in the "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," NUREG/CR-1278. Based on the NUREG/CR-1278 methodology, time, function, and spatial attributes were judged to be the most important considerations when determining the level of dependence between operator actions within an accident sequence. These attributes were used to develop qualitative criteria (rules) that were used to judge the level of dependence (complete dependency (CD), high dependency, medium dependency, low dependency, and zero dependency (ZD)) between the operator actions. After the level of dependence between the various HEPs was determined using these rules, quantitative values associated with the level of dependence was assigned and used in a quantitative sensitivity assessment. Based on this systematic framework for analysis of human action dependency, it was concluded that many HEPs are already modeled as CD in the VYNPS PSA model. Likewise, many of the HEPs were judged to have zero influence (zero dependency) on other HEPs in the same sequence. Only a few were judged to have some level of dependence other than ZD or CD that was not already captured in the VYNPS PSA model. The NRC staff concludes that the licensee's treatment of operator action dependencies is reasonable and it is based on methodologies previously accepted by the staff in various risk-informed license amendment requests.

The licensee assessed how the relative significance of the operator actions to the VYNPS risk profile changed as a result of the proposed EPU. The risk importance measures of the operator actions changed slightly in the post-EPU PSA, but did not change their relative significance to the VYNPS risk profile. As such, the licensee concluded that the proposed EPU

does not introduce new risk-significant operator actions. The NRC staff finds this conclusion to be reasonable.

The VYNPS PSA model credits recovery of offsite power for SBO sequences. Offsite power (OSP) recovery is assumed to be needed before the DC-1 and DC-2 batteries deplete. The licensee used NUREG/CR-5496 to obtain OSP non-recovery probabilities. The NRC staff observes that this source of data is somewhat dated. To confirm the OSP non-recovery probabilities used by the licensee, the staff compared them to LOOP duration data recently collected and analyzed by RES in response to the August 2003 Northeast Blackout. The staff also observes that the proposed EPU has no obvious cause-and-effect relationship to OSP non-recovery probabilities. The staff concludes that the licensee's assessment of OSP non-recovery probabilities is reasonable because it produces values that are similar to those recently determined by RES.

Level 1 Internal Events Results

The licensee stated that the proposed EPU increases the CDF by 3×10^{-7}/year (an increase of 4.2% over the pre-EPU CDF of 8×10^{-6}/year). The increase is due to the change in the TT frequency and the reduced times for certain operator actions. There is no visible impact on the ATWS-related CDF due to addition of a new SSV (required to implement ARTS/MELLLA).

The NRC staff finds that the licensee's evaluation of the impact of the proposed EPU on internal initiating event risk is reasonable and it is based on methodologies previously accepted by the staff for use in IPEs and EPU risk evaluations. Since the CDF risk metrics satisfy the risk acceptance guidelines in RG 1.174, the staff concludes that the change in internal initiating event risk due to the proposed EPU is very small and that there are no issues concerning internal initiating events that rebut the presumption of adequate protection provided by the licensee meeting the currently specified regulatory requirements.

Level 1 Internal Fire Risk Evaluation

The VYNPS plant risk due to internal fires was evaluated in 1998 as part of the VYNPS IPEEE using the fire induced vulnerability evaluation (FIVE) methodology developed by the Electric Power Research Institute (EPRI). The intent of the IPEEE program was to identify plant vulnerabilities pertaining to severe accidents, and the NRC staff has accepted the FIVE methodology as acceptable for that purpose. The VYNPS IPEEE internal fire analysis identified the most risk significant fire areas in the plant using a screening process and by calculating conservative core damage frequencies for fire scenarios. As such, the accident sequence frequencies calculated for the VYNPS fire PSA are not a best estimate calculation of plant fire risk and should not be combined with the internal events PSA results for comparison with RG 1.174 acceptance guidelines. RG 1.174 allows the use of qualitative methods for assessing the risk of proposed licensing basis changes.

The licensee stated that the impact of the proposed EPU on the different aspects of the internal fire qualitative risk assessment were assessed based on knowledge of the VYNPS Fire IPEEE

and the modifications for the proposed EPU (e.g., no significant changes to combustible loadings, no significant changes to fire protection systems). Based on this qualitative assessment, the licensee concluded that no unique impacts on internal fire risk would result from the proposed EPU.

The NRC staff finds that the licensee's evaluation of the impact of the proposed EPU on internal fire risk is reasonable and it is based on a methodology previously accepted by the staff for use in IPEEEs and EPU risk evaluations. Since the licensee did not identify any new plant vulnerabilities related to internal fires arising from the proposed EPU, the staff concludes that the change in internal fire risk due to the proposed EPU is very small and that there are no issues concerning internal fires that rebut the presumption of adequate protection provided by the licensee meeting the currently specified regulatory requirements.

Level 1 Seismic Risk Evaluation

The VYNPS plant risk due to seismic events was evaluated in 1998 as part of the VYNPS IPEEE using the seismic margins assessment (SMA) as provided in NUREG-1407 and EPRI NP-6041. The intent of the IPEEE program was to identify plant vulnerabilities pertaining to severe accidents, and the NRC staff has accepted the SMA methodology as acceptable for that purpose. The SMA is a deterministic evaluation process that does not calculate risk on a probabilistic basis. No CDF sequences were quantified as part of the seismic risk evaluation. RG 1.174 allows the use of qualitative methods for assessing the risk of proposed licensing basis changes.

Based on a review of the VYNPS IPEEE and the key general assumptions identified earlier in this assessment, the licensee concluded that the results and insights of the SMA are considered to be unaffected by the proposed EPU because there will little or no effect on the seismic qualifications of the systems, structures, and components (SSCs). Specifically, the proposed EPU increases the amount of thermal energy stored in the RPV, which causes an increase in the blowdown loads on the RPV and the containment following a seismically-induced LOCA. The licensee has indicated that the increase in the blowdown loads would be minor and, therefore, does not affect the determination of the HCLPF values of the RPV and containment. The staff observes that the HCLFP values for other SSCs are not affected by the proposed EPU because there will be no changes to equipment mountings or building structures, and replacement equipment will be installed using anchorages that are similar to the existing anchorages. Further, the plant modifications associated with the proposed EPU do not alter the definition of the primary and alternative safe shutdown paths defined in the SMA. Therefore, the inputs to the SMA are not affected by the proposed EPU.

The NRC staff finds that the licensee's evaluation of the impact of the proposed EPU on seismic risk is reasonable and it is based on a methodology previously accepted by the staff for use in IPEEEs and EPU risk evaluations. Since the licensee did not identify any new plant vulnerabilities related to seismic events arising from the proposed EPU, the staff concludes that the change in seismic risk due to the proposed EPU is very small and that there are no issues

concerning seismic events that rebut the presumption of adequate protection provided by the licensee meeting the currently specified regulatory requirements.

Level 1 Other External Events Risk Evaluation

In addition to internal fires and seismic events, the VYNPS IPEEE analyzed a variety of other external hazards:

- High winds/tornadoes
- External floods
- Transportation and nearby facility accidents
- Other external hazards

The VYNPS IPEEE analysis of high winds, tornadoes, external floods, transportation accidents, nearby facility accidents, and other external hazards was accomplished by reviewing the plant environs against regulatory requirements regarding these hazards. Based on this review, it was concluded that VYNPS meets the applicable NRC SRP requirements and therefore has an acceptably low risk with respect to these hazards.

The NRC staff finds that the licensee's evaluation of the impact of the proposed EPU on other external event risk is reasonable and it is based on a methodology previously accepted by the staff for use in IPEEEs and EPU risk evaluations. The staff concludes that there are no issues concerning other external events that rebut the presumption of adequate protection provided by the licensee meeting the currently specified regulatory requirements.

Level 2 Evaluation

Fission product inventory in the reactor core is higher as a result of the increase in power due to the proposed EPU. The increase in fission product inventory results in an increase (approximately 20%) in the total radionuclides available for release given a severe accident. However, this does not affect the definition or quantification of the LERF risk measure used in RG 1.174, which is the basis for the VYNPS Level 2 risk assessment.

The licensee assessed the impact of the proposed EPU on the Level 2 PSA. The assessment considered the following major issues:

- Level I PSA input
- Accident progression
- Operator actions
- Success criteria
- Containment capability
- Release

Approximately 6 Level 2 MAAP runs were performed in support of the proposed EPU risk assessment. The Level 2 MAAP runs were focused on the assessment of any significant

changes in release categories. No changes to the VYNPS PSA Level 2 success criteria, accident progression logic modeling, or the release binning categorization were judged necessary to reasonably represent the post-EPU plant. The slight changes in accident progression timing and decay heat load have only minor or negligible impacts on Level 2 PSA safety functions, such as containment isolation, ex-vessel debris coolability, and challenges to the ultimate containment strength.

The licensee stated that post-core damage (Level 2 PSA) operator actions were considered in the operator action screening process for the VYNPS post-EPU risk assessment. However, no Level 2 PSA action human error probabilities required re-calculation due to the proposed EPU. Either the Level 2 action did not meet the screening criteria or the action is a "recovery probability" (recovery probabilities would not be adjusted based on the timing changes of the proposed EPU).

In addition, the proposed EPU does not change the containment capability assessment. The changes to the plant from the proposed EPU have no impact on the definition of the containment loading profiles or the likelihood of containment isolation failure. The slightly higher decay heat levels associated with the proposed EPU will result in minor reductions in times to reach loading challenges; however, the time frames are long (many hours) and the accident timing reductions of 10-15% due to the proposed EPU have an insignificant (even non-quantifiable) impact on the Level 2 results. Regarding energetic phenomena occurring at or near the time of core slump or RPV breach, such accident progression scenarios are appropriately modeled in the VYNPS Level 2 PSA as leading directly to high magnitude releases.

The VYNPS Level 2 PSA release categorization scheme uses both release magnitude and timing. Release categories were assigned to the VYNPS pre-EPU PSA based on results of representative MAAP runs for many accident scenarios, and based on judgment and standard industry approaches for selected scenarios (e.g., see discussion above related to containment failures due to energetic phenomena).

The VYNPS release magnitude classification is based on the percentage, as a function of the initial end-of-cycle inventory in the core, of cesium iodide released to the environment; this approach is consistent with the majority of BWR PSAs. Changes to the release categories assigned to individual accident sequences in the VYNPS Level 2 PSA were not necessary; this was confirmed by MAAP runs. Typical post-core damage accident scenarios were run (e.g., transient with loss of all coolant injection, RPV breach, and subsequent primary containment failure due to shell melt-through) and the assigned release magnitude classifications for the scenarios did not change between the pre-EPU and post-EPU cases. While a thermal-hydraulic case may be uniquely devised such that it calculates a release magnitude that is just below the border of the Moderate and High release categories so that the post-EPU condition may then push it into the High category, such cases are not representative of the VYNPS PSA (in fact, the MAAP runs performed for the VYNPS CPPU risk assessment could not produce such a case without making unrealistic MAAP modeling assumptions).

The best estimate at-power internal events LERF increase due to the CPPU is a change in LERF of 1 x 10^{-7}/year (an increase of about 5% over the pre-EPU LERF of 2 x 10^{-6}/year). Given the minor change in Level I CDF results, minor changes in the Level 2 PSA release frequencies result. Such changes are directly attributable to the change in the TT initiating event frequency and the minor changes in short-term accident sequence timing and their effect on HEPs.

The NRC staff finds that the licensee's evaluation of the impact of the proposed EPU on LERF is reasonable and it is based on a methodology previously accepted by the staff for use in risk-informed submittals and EPU risk evaluations. Since the LERF risk metrics satisfy the risk acceptance guidelines in RG 1.174, the staff concludes that the change in LERF due to the proposed EPU is very small and that there are no issues concerning containment performance that rebut the presumption of adequate protection provided by the licensee meeting the currently specified regulatory requirements.

Shutdown Risk Evaluation

The effect of the proposed EPU on shutdown risk is similar to the effect on the at-power Level 1 PSA. Based on the insights of the at-power PSA effect assessment, the licensee identified the following areas for review appropriate to shutdown risk:

- Initiating events
- Success criteria
- Human reliability analysis

The licensee stated that it does not maintain a shutdown PSA. The following qualitative discussion, provided by the licensee, applies to the shutdown conditions of hot shutdown (Mode 3), cold shutdown (Mode 4), and refueling (Mode 5). The risk effect of the proposed EPU during the transitional periods such as at-power (Mode 1) to hot shutdown and startup (Mode 2) to at-power are judged to be subsumed by the at-power Level 1 PSA.

Shutdown initiating events include the following major categories:

1. Loss of reactor coolant system inventory

2. Inadvertent drain-down

    a. LOCAs
    b. Loss of decay heat removal (DHR), including LOOP

No new initiating events or increased potential for initiating events during shutdown (e.g., loss of DHR train) can be postulated due to the proposed EPU.

The effect of the proposed EPU on the success criteria during shutdown is similar to the Level 1 PSA. The increased power level decreases the time to core boildown. However, because the reactor is already shutdown, the boildown times are much longer compared to the at-power

PSA. The time needed to boildown to the top of active fuel is approximately 2 hours at 2 hours after shutdown (e.g., time of hot shutdown), and approximately 4-6 hours at 12-24 hours after shutdown (e.g., time of cold shutdown). The decrease in the boildown time for the EPU is small because of the lower decay heat level relative to at-power conditions.

The increased decay heat loads associated with the proposed EPU affects the time when low capacity DHR systems can be considered successful alternate DHR systems. The increased power delays the time after shutdown when low capacity DHR systems may be used as an alternative to shutdown cooling. However, shutdown risk is dominated during the early time frame soon after shutdown when the decay heat level is high and, in this time frame, low capacity DHR alternatives are already not viable DHR systems.

Other success criteria are marginally affected by the proposed EPU. There is a minor effect on shutdown RPV inventory makeup during loss of DHR scenarios in shutdown because of the low decay heat level. The heat load to the suppression pool during loss of DHR scenarios in shutdown (i.e., during shutdown phases with the RPV intact) is also lower because of the low decay heat level such that the margins for suppression pool cooling capacity are adequate for the proposed EPU.

The impact of the proposed EPU on the success criteria for blowdown loads, RPV overpressure margin, and SRV actuation is estimated to be negligible because of the low RPV pressure and low decay heat level during shutdown.

Similar to the at-power Level 1 PSA, the decreased boildown time due to the proposed EPU decreases the time available for operator actions and OSP recoveries. The risk significant operator actions and OSP recoveries during shutdown conditions include recovering a failed DHR system or initiating alternate DHR systems. Thermal-hydraulic calculations performed for the VYNPS water volumes during shutdown conditions show that the times available to perform loss of DHR response actions during shutdown is many hours. The reductions in these times due to the proposed EPU is shown in the range of 10% to 15% (depending on time after shutdown and water level configuration). Such small changes in already lengthy response times result in negligible changes in HEPs and in OSP non-recovery probabilities.

The licensee stated that it will continue to use a computerized risk monitor and site-specific matrices as tools for controlling outage risk. The NRC staff finds that the licensee's qualitative assessment of shutdown risks associated with the proposed EPU is reasonable because the licensee has demonstrated that:

- Suitably redundant and diverse plant response capability is maintained for significant initiators during shutdown modes, and

- Sufficient elements of the plant response capability are subject to programmatic activities to ensure suitable performance.

Therefore, the NRC staff concludes there are no issues concerning shutdown operations that rebut the presumption of adequate protection provided by the licensee meeting the currently specified regulatory requirements.

PSA Model Quality

The VYNPS PSA used to support the risk evaluation of the proposed EPU is an evolution of the IPE and IPEEE developed in response to GL 88-20. The IPE was submitted to the NRC staff on December 21,1993; the staff replied on January 3, 1996, that the licensee's IPE met the intent of GL 88-20. The IPEEE was submitted to the staff on June 30, 1998; the staff replied on March 22, 2001, that the IPEEE met the intent of GL 88-20. The licensee has stated that all of the staff's findings on the IPE and IPEEE have been addressed.

Since its submittal to the NRC staff, the PSA model was updated several times to maintain it consistent with the as-built, as-operated plant. In November 2000, an owners group peer review of the PSA was conducted. A total of 104 Findings and Observations (F&Os) were identified. Of these, there was 1 category "A" and 51 category "B" review findings. The licensee provided information to the staff indicating how each of the F&Os was resolved. The staff reviewed this information, and concludes that they have been adequately addressed for the purpose of assessing the risk impacts of the proposed EPU.

In May 2003, the NRC conducted a benchmarking exercise of its SDP Phase 2 model by comparing its results to the then current PSA model (VY02, Revision 3). Two changes were made to the model to reflect the results of the benchmarking exercise. The current version of the VYNPS PSA model is VY02, Revision 6, which was completed in July 2003.

The NRC staff finds that the licensee has met the intent of RG 1.174 (Sections 2.2.3 and 2.5), SRP Chapter 19 (Section III.2.2.4), and SRP Chapter 19.1, and that the VYNPS PSA has sufficient scope, level of detail, and technical adequacy to support the risk evaluation of the proposed EPU.

Conclusion

The NRC staff has reviewed the licensee's assessment of the risk implications associated with the implementation of the proposed EPU and concludes that the licensee has adequately modeled and/or addressed the potential impacts associated with implementation of the proposed EPU. The NRC staff further concludes that the results of the licensee's risk analysis indicate that the risks associated with the proposed EPU are acceptable and do not create the "special circumstances" described in Appendix D of SRP Chapter 19. Therefore, the NRC staff finds the risk implications of the proposed EPU acceptable.

3.0 FACILITY OPERATING LICENSE AND TECHNICAL SPECIFICATION CHANGES

To achieve the EPU, the licensee proposed the following changes to the Facility Operating License (FOL) and TSs for VYNPS.

3.1 FOL, Page 3, Section 3.A - Maximum Power Level

The maximum licensed reactor core power level would be changed from 1593 MWt to 1912 MWt. This change reflects the proposed 20% increase in the thermal power level for the plant and is consistent with the licensee's supporting safety analyses. Therefore, the NRC staff finds the proposed change acceptable.

3.2 TS Page 3, Definitions 1.0.P and 1.0.Q - Rated Neutron Flux and Rated Thermal Power

Definition 1.0.P, "Rated Neutron Flux," and 1.0.Q, "Rated Thermal Power," would be revised to reflect that rated neutron flux and rated thermal power correspond to a steady state power level of 1912 MWt rather than the CLTP level of 1593 MWt. These changes reflect the proposed 20% increase in the thermal power level for the plant and are consistent with the licensee's supporting safety analyses. Therefore, the NRC staff finds the proposed changes acceptable.

3.3 TS Page 6, TS 2.1.A.1.a - APRM Flux Scram Trip Setting (Run Mode)

The current allowable values (AVs) for the APRM flow-biased flux scram in TS 2.1.A.1.a are based on implementation of VYNPS Amendment No. 219, dated April 14, 2004 (Reference 53), which revised the TSs to reflect an expanded operating domain resulting from the implementation of the Average Power Range Monitor, Rod Block Monitor TSs/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA). The current AVs are as follows:

Two loop operation:
$S \leq 0.4W + 61.10\%$ for $0\% < W \leq 31.1\%$
$S \leq 1.28W + 33.31\%$ for $31.1\% < W \leq 54.0\%$
$S \leq 0.66W + 67.28\%$ for $54.0\% < W \leq 75.0\%$
With a maximum of 117.0% power for $W > 75\%$

Single loop operation:
$S \leq 0.4\,W + 58.09\%$ for $0\% < W \leq 39.1\%$
$S \leq 1.28\,W + 23.56\%$ for $39.1\% < W \leq 61.9\%$
$S \leq 0.66\,W + 62.10\%$ for $61.9\% < W \leq 83.0\%$
With a maximum of 117.0% power for $W > 83.0\%$

where:

S = setting in percent of rated thermal power (1593 MWt)

W = percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow

The proposed amendment would change the AVs to read as follows:

Two loop operation:
$S \leq 0.33W + 50.45\%$ for $0\% < W \leq 30.9\%$
$S \leq 1.07W + 27.23\%$ for $30.9\% < W \leq 66.7\%$
$S \leq 0.55W + 62.34\%$ for $66.7\% < W \leq 99.0\%$
With a maximum of 117.0% power for W > 99%

Single loop operation:
$S \leq 0.33\ W + 48.00\%$ for $0\% < W \leq 39.1\%$
$S \leq 1.07\ W + 19.01\%$ for $39.1\% < W \leq 61.7\%$
$S \leq 0.55\ W + 51.22\%$ for $61.7\% < W \leq 119.4\%$
With a maximum of 117.0% power for W > 119.4%

where:

S = setting in percent of rated thermal power (1912 MWt)

W = percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow

The methodology for determining the AVs for the APRM flow-biased flux scram was previously reviewed and approved by the NRC staff in Reference 53. As discussed in the TS Bases for TS 2.1.A.1.a, the AV is the limiting value that the trip setpoint may have when tested periodically. For VYNPS, the periodic testing is defined as the calibration. The actual scram trip is conservatively set in relation to the AV to ensure operability between periodic testing. The NRC staff finds the proposed changes acceptable since the AVs were developed based on the use of a previously-approved methodology.

3.4 TS Page 7, TS 1.1.B - Core Thermal Power Limit

TS 1.1.b currently provides the following safety limit (SL):

When the reactor pressure is ≤ 800 psia or core flow ≤ 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

The proposed amendment would change the SL from 25% to 23% of rated thermal power (RTP).

As discussed in Section 2.1 of the PUSAR:

> The percent power level above which fuel thermal margin monitoring is required may change with CPPU. The original plant operating licenses set this monitoring threshold at a typical value of 25% of Rated Thermal Power (RTP). [[
>
>]]
>
> For CPPU, as specified in the CLTR, the fuel thermal margin monitoring threshold is scaled down, if necessary, to ensure that the monitoring is initiated [[
>
>]], then the existing power threshold value must be lowered by a factor of $1.2/P_{25}$.
>
> For VYNPS, the CPPU fuel thermal monitoring threshold is established at 23% of CPPU RTP. A change in the fuel thermal monitoring threshold also requires a corresponding change to the Technical Specification reactor core safety limit for reduced power or low core flow.

Since the revised SL is based on [[
]], the analysis is bounding for VYNPS. Therefore, the NRC staff finds the proposed change acceptable.

3.5 <u>TS Page 7, TS 2.1.A.1.a - APRM Flux Scram Trip Setting</u>

Currently, TS 2.1.A.1.a states that:

> In the event of operation at 25% Rated Thermal Power the APRM gain shall be equal to or greater than 1.0.

The proposed amendment would change this TS to read:

> In the event of operation at 23% Rated Thermal Power the APRM gain shall be equal to or greater than 1.0.

The basis for the change from 25% RTP to 23% RTP is the same as discussed in SE Section 3.4. Therefore, the NRC staff finds the proposed change acceptable.

3.6 TS Page 10, TS 2.1.E - Turbine Stop Valve Scram Bypass

Currently, TS 2.1.E states that:

> Turbine stop valve scram shall, when operating at greater than 30% of Rated Thermal Power, be less than or equal to 10% valve closure from full open.

The proposed amendment would change this TS to read:

> Turbine stop valve scram shall, when operating at greater than 25% of Rated Thermal Power, be less than or equal to 10% valve closure from full open.

As discussed in PUSAR Section 5.3.2, the EPU results in an increased power level, and the high pressure turbine modifications result in a change to the relationship of turbine first-stage pressure to reactor power level. The turbine first-stage pressure setpoint is used to reduce scrams at low power levels where the turbine steam bypass system is effective for turbine trips and generator load rejections. In the safety analysis, this trip bypass only applies to events at low power levels that result in a turbine trip or load rejections. [[

]]

The various TSs which are based on the turbine first-stage pressure function currently specify an analytical limit of 30% of RTP. Using the approach discussed above, this value would be changed to 25% of RTP under EPU conditions (i.e., 30% x (100% + 120%) = 25%). Since the approach is consistent with the licensee's supporting safety analyses, the NRC staff finds the proposed change acceptable.

3.7 TS Page 10, TS 2.1.F - Turbine Control Valve Fast Closure Scram Bypass

Currently, TS 2.1.F states that:

> Turbine control valve fast closure scram shall, when operating at greater than 30% of Rated Thermal Power, trip upon actuation of the turbine control valve fast closure relay.

The proposed amendment would change this TS to read:

> Turbine control valve fast closure scram shall, when operating at greater than 25% of Rated Thermal Power, trip upon actuation of the turbine control valve fast closure relay.

Based on the discussion in SE Section 3.6, the NRC staff finds the proposed change acceptable.

3.8 TS Page 21, TS Table 3.1.1 - APRM High Flux (flow bias)

Table 3.1.1 provides the Reactor Protection System (RPS) scram instrument requirements. The proposed amendment would change the "Trip Settings" column of this table for RPS trip function 4, APRM High Flux (flow bias). The current and proposed trip settings for this function are actually AVs and are the same as the AVs for two loop operation and one loop operation for TS 2.1.A.1.a as discussed in SE Section 3.3. As such, the NRC staff finds the proposed changes acceptable since the AVs were developed based on the use of a previously-approved methodology.

3.9 TS Page 24, Table 3.1.1 Note 3d

TS Table 3.1.1 Note 3 provides the required actions when the number of instrument channels is less than the minimum required. Note 3, step d, currently requires that reactor power be reduced to less than 30% of rated within 8 hours. The proposed amendment would change the "30% of rated" to "25% of rated."

Based on the discussion in SE Section 3.6, the NRC staff finds the proposed change acceptable.

3.10 TS Page 24, Table 3.1.1 Note 10

TS Table 3.1.1 Note 10 pertains to the Trip Settings for the turbine control valve fast closure and turbine stop valve closure scram in Table 3.1.1. Note 10 currently states:

> Turbine stop valve closure and turbine control valve fast closure scram signals may be bypassed at ≤ 30% of reactor Rated Thermal Power.

The proposed amendment would change this note to read:

> Turbine stop valve closure and turbine control valve fast closure scram signals may be bypassed at ≤ 25% of reactor Rated Thermal Power.

Based on the discussion in SE Section 3.6, the NRC staff finds the proposed change acceptable.

3.11 TS Page 83, TS 3.3.B.3 - Rod Worth Minimizer

Currently, TS 3.3.B.3 states that:

> While the reactor is below 20% power, the Rod Worth Minimizer (RWM) shall be operated while moving control rods...

The proposed amendment would change this TS to read:

> While the reactor is below 17% power, the Rod Worth Minimizer (RWM) shall be operated while moving control rods...

As discussed in Attachment 1 to the application dated September 10, 2003 (Reference 1), the EPU RWM low power setpoint is established at the same absolute thermal power level as currently licensed for the control rod drop accident. Since the approach is consistent with the licensee's supporting safety analyses, the NRC staff finds the proposed change acceptable.

3.12 TS Page 92, TS 4.4.A.1 - Standby Liquid Control System Pump Discharge Pressure

Currently, TS 4.4.A.1 states that the Standby Liquid Control (SLC) system shall be verified operable by:

> Testing pumps and valves in accordance with Specification 4.6.E. A minimum flow rate of 35 gpm at 1320 psig shall be verified for each pump.

The proposed amendment would change this TS to read:

> Testing pumps and valves in accordance with Specification 4.6.E. A minimum flow rate of 35 gpm at ≥ 1325 psig shall be verified for each pump.

As shown in PUSAR Table 9-5, and discussed in PUSAR Section 6.5 and in Attachment 10 to the licensee's letter dated August 1, 2005 (Reference 31), the results of the licensee's ATWS analysis for VYNPS determined that the calculated peak reactor vessel bottom pressure increases from 1367 psig at CLTP conditions to 1490 psig at EPU conditions. However, this peak occurs very early in the transient event. During the time period when SLC is assumed to operate, the peak pressure at EPU conditions was determined to be 1292 psia (1277 psig). The minimum SLC system pump relief valve nominal setpoint for EPU is 1400 psig. As discussed in Reference 31, based on the 1325 psig discharge test pressure, there is a minimum of 75 psi margin. This margin provides allowance for SLC pump relief valve setpoint drift and SLC pump pressure pulsations. The GE recommendation for relief setpoint margin is 75 psi.

Based on the plant-specific ATWS analysis results for EPU conditions, the NRC staff finds that the proposed SLC pump test pressure of ≥ 1325 psig provides reasonable assurance that the pump can develop the necessary pressure for SLC injection and provides sufficient margin to prevent the respective SLC pump relief valve from lifting. Therefore, the NRC staff finds the proposed change acceptable.

3.13 TS Page 94, TS 3.4.C.3 - Standby Liquid Control System Operabilty Factors

Currently, TS 3.4.C.3 states that the combination of SLC system pump flow rate, boron concentration, and boron enrichment shall satisfy the following relationship for the SLC system to be operable:

$Q/86 \times M251/M \times C/13 \times E/19.8 \geq 1$

where:

C = the concentration of sodium penetaborate solution (weight percent) in the Standby Liquid Control tank

E = the boron-10 enrichment (atom percent) of the sodium penetaborate solution

Q = 35 gpm

M251/M = a constant (the ratio of mass of water in the reference plant compared to VY)

The proposed amendment would change the mathematical expression to read as follows:

$Q/86 \times M251/M \times C/13 \times E/19.8 \geq 1.29$

In addition the definition for the pump flow rate would be changed to read as follows:

$Q \geq 35$ gpm

As discussed in Attachment 1 to the application dated September 10, 2003 (Reference 1) and Attachment 6 to the licensee's letter dated September 10, 2005 (Reference 33), the licensee's analysis of the ATWS event at EPU conditions shows the need to increase the combined relationship of the SLC system pump flow rate, concentration, and enrichment to meet ATWS acceptance criteria. The EPU analysis was performed using the following SLC system nominal values:

- flow rate of 40.5 gpm;
- boron concentration of 10.42 weight percent; and
- boron-10 enrichment of 43%.

When these values are combined with the mass ratio (628,000 lbs/401,247 lbs), the result is slightly less than 1.29. To ensure the EPU ATWS analysis remains bounding, the equivalency equation was modified to require meeting the more stringent value of 1.29 rather than 1.

The NRC staff finds that the proposed change is consistent with the EPU ATWS analysis and meets the intent of the requirements in 10 CFR 50.62(c)(4). Therefore, the NRC staff finds the proposed change acceptable.

3.14 TS Pages 135, 136, and 137, TS Figures 3.6.1, 3.6.2, and 3.6.3 - Reactor Vessel Pressure-Temperature Limits

TS Figures 3.6.1, 3.6.2, and 3.6.3, provide the reactor vessel pressure-temperature (P-T) limits for a range of plant conditions. As discussed in Attachment 1 to the application dated September 10, 2003 (Reference 1), the limitations and requirements imposed by these figures are unchanged for the EPU. The only change proposed to each of the figures is to revise the period over which the figure is applicable. Currently, each of the figures states that it is valid through a thermal power output of 4.46×10^8 megawatt-hours (MWH). This value would be changed to 4.827×10^8 MWH in each of the figures.

As discussed in SE Section 2.1.1, the value of 4.827×10^8 MWH corresponds to EPU conditions at the end of the current VYNPS license term. As discussed in Section 2.1.2, the NRC staff found that the existing P-T limits remain bounding for EPU conditions. Based on the considerations in SE Sections 2.1.1 and 2.1.2, the NRC staff concludes that the proposed change is acceptable.

3.15 TS Pages 224, 225, and 226, TS 3/4.11.A, TS 3/4.11.B, TS 3/4.11.C - Reactor Fuel Assemblies

TS 3.11.A and TS 4.11.A provide the limiting conditions for operation (LCOs) and surveillance requirements (SRs) respectively for the average planar heat generation rate (APLHGR). TS 3.11.B and TS 4.11.B provide the LCOs and SRs respectively for the Linear Heat Generation Rate (LHGR). TS 3.11.C and TS 4.11.C provide the LCOs and SRs respectively for the Minimum Critical Power Ratio (MCPR). Each of these LCOs currently specify limits for when a specific condition is applicable based on plant operation at either ≥ 25% of RTP or < 25% of RTP. Similarly, each of these SRs specify limits for when specific actions are required based on plant operation at ≥ 25% of RTP. The proposed amendment would change each of these limits from 25% of RTP to 23% of RTP. The signs for each of the limits would remain the same.

As discussed in PUSAR Section 2.1, the 25% of RTP value, that is currently specified in the TSs for APLHGR, LHGR, and MCPR, represents the power level above which thermal margin monitoring is required. Based on the discussion in SE Section 3.4, the NRC staff finds the proposed changes acceptable.

3.16 TS Bases

The licensee has also proposed changes to the TS Bases for clarity and to conform to the changes being made to the associated TSs. The NRC staff has no objections to these changes.

3.17 License Conditions

Entergy proposed a license condition concerning the steam dryer in its letter dated December 9, 2004 (Reference 22). This license condition was subsequently revised and superceded by Entergy's letters dated March 31, 2005 (Reference 27), and September 14, 2005 (Reference 34). Entergy also proposed a license condition concerning the minimum critical power ratio in its letter dated September 28, 2005 (Reference 36).

In a letter dated October 12, 2005 (Reference 73), the NRC staff proposed three license conditions. One of the conditions slightly modified Entergy's proposed license condition concerning the minimum critical power ratio. Another condition, pertaining to monitoring and evaluating potential adverse flow effects (including steam dryer structural integrity), added new requirements and modified the steam dryer license condition proposed by Entergy's letter dated September 14, 2005. A third condition, proposed by the NRC staff, addressed transient testing of the condensate and feedwater system.

Following issuance of the NRC staff's letter dated October 12, 2005, Entergy and the NRC staff held several conference calls to discuss some minor modifications to the license conditions proposed by the NRC staff. The final version of the license conditions, as accepted by the licensee, are reflected in Entergy's letter dated October 17, 2005 (Reference 37). Based on the October 17, 2005, letter, paragraph 3 of Facility Operating License (FOL) DPR-28 would be revised to add new license conditions 3.K, 3.L, and 3.M, as discussed below.

3.17.1 Minimum Critical Power Ratio

The following would be added to the FOL as license condition 3.K. This license condition is addressed in SE Section 2.8.7.2.2.3.1.

> K. Minimum Critical Power Ratio
>
> When operating at thermal power greater than 1593 megawatts thermal, the safety limit minimum critical power ratio (SLMCPR) shall be established by adding 0.02 to the cycle-specific SLMCPR value calculated using the NRC-approved methodologies documented in General Electric Licensing Topical Report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," as amended, and documented in the Core Operating Limits Report.

3.17.2 Transient Testing

The following would be added to the FOL as condition 3.L. This condition is addressed in SE Section 2.5.4.4.

L. Transient Testing

1. During the extended power uprate (EPU) power ascension test program and prior to exceeding 168 hours of plant operation at the nominal full EPU reactor power level, with feedwater and condensate flow rates stabilized at approximately the EPU full power level, Entergy Nuclear Operations, Inc. shall confirm through performance of transient testing that the loss of one condensate pump will not result in a complete loss of reactor feedwater.

2. Within 30 days at nominal full-power operation following successful performance of the test in (1) above, through performance of additional transient testing and/or analysis of the results of the testing conducted in (1) above, confirm that the loss of one reactor feedwater pump will not result in a reactor trip.

3.17.3 Potential Adverse Flow Effects

The following would be added to the FOL as condition 3.M. This condition is addressed in SE Section 2.2.6.

M. Potential Adverse Flow Effects

This license condition provides for monitoring, evaluating, and taking prompt action in response to potential adverse flow effects as a result of power uprate operation on plant structures, systems, and components (including verifying the continued structural integrity of the steam dryer).

1. The following requirements are placed on operation of the facility above the original licensed thermal power (OLTP) level of 1593 megawatts thermal (MWt):

   a. Entergy Nuclear Operations, Inc. shall monitor hourly the 32 main steam line (MSL) strain gages during power ascension above 1593 MWt for increasing pressure fluctuations in the steam lines.

   b. Entergy Nuclear Operations, Inc. shall hold the facility for 24 hours at 105%, 110%, and 115% of OLTP to collect data from the 32 MSL strain gages required by Condition M.1.a, conduct plant inspections and walkdowns, and evaluate steam dryer performance based on these data; shall provide the evaluation to the NRC staff by facsimile or electronic transmission to the NRC project manager upon completion of the evaluation; and shall not increase power above each hold point until 96 hours after the NRC project manager confirms receipt of the transmission.

c. If any frequency peak from the MSL strain gage data exceeds the limit curve established by Entergy Nuclear Operations, Inc. and submitted to the NRC staff prior to operation above OLTP, Entergy Nuclear Operations, Inc. shall return the facility to a power level at which the limit curve is not exceeded. Entergy Nuclear Operations, Inc. shall resolve the uncertainties in the steam dryer analysis, document the continued structural integrity of the steam dryer, and provide that documentation to the NRC staff by facsimile or electronic transmission to the NRC project manager prior to further increases in reactor power.

d. In addition to evaluating the MSL strain gage data, Entergy Nuclear Operations, Inc. shall monitor reactor pressure vessel water level instrumentation or MSL piping accelerometers on an hourly basis during power ascension above OLTP. If resonance frequencies are identified as increasing above nominal levels in proportion to strain gage instrumentation data, Entergy Nuclear Operations, Inc. shall stop power ascension, document the continued structural integrity of the steam dryer, and provide that documentation to the NRC staff by facsimile or electronic transmission to the NRC project manager prior to further increases in reactor power.

e. Following start-up testing, Entergy Nuclear Operations, Inc. shall resolve the uncertainties in the steam dryer analysis and provide that resolution to the NRC staff by facsimile or electronic transmission to the NRC project manager. If the uncertainties are not resolved within 90 days of issuance of the license amendment authorizing operation at 1912 MWt, Entergy Nuclear Operations, Inc. shall return the facility to OLTP.

2. As described in Entergy Nuclear Operations, Inc. letter BVY 05-084 dated September 14, 2005, Entergy Nuclear Operations, Inc. shall implement the following actions:

   a. Prior to operation above OLTP, Entergy Nuclear Operations, Inc. shall install 32 additional strain gages on the main steam piping and shall enhance the data acquisition system in order to reduce the measurement uncertainty associated with the acoustic circuit model (ACM).

   b. In the event that acoustic signals are identified that challenge the limit curve during power ascension above OLTP, Entergy Nuclear Operations, Inc. shall evaluate dryer loads and re-establish the limit curve based on the new strain gage data, and shall perform a frequency-specific assessment of ACM uncertainty at the acoustic signal frequency.

   c. After reaching 120% of OLTP, Entergy Nuclear Operations, Inc. shall obtain measurements from the MSL strain gages and establish the steam dryer flow-induced vibration load fatigue margin for the facility, update the dryer stress report, and re-establish the steam dryer monitoring plan (SDMP) limit curve with

the updated ACM load definition and revised instrument uncertainty, which will be provided to the NRC staff.

d. During power ascension above OLTP, if an engineering evaluation is required in accordance with the SDMP, Entergy Nuclear Operations, Inc. shall perform the structural analysis to address frequency uncertainties up to ±10% and assure that peak responses that fall within this uncertainty band are addressed.

e. Entergy Nuclear Operations, Inc. shall revise the SDMP to reflect long-term monitoring of plant parameters potentially indicative of steam dryer failure; to reflect consistency of the facility's steam dryer inspection program with General Electric Services Information Letter 644, Revision 1; and to identify the NRC Project Manager for the facility as the point of contact for providing SDMP information during power ascension.

f. Entergy Nuclear Operations, Inc. shall submit the final extended power uprate (EPU) steam dryer load definition for the facility to the NRC upon completion of the power ascension test program.

g. Entergy Nuclear Operations, Inc. shall submit the flow-induced vibration related portions of the EPU startup test procedure to the NRC, including methodology for updating the limit curve, prior to initial power ascension above OLTP.

3. Entergy Nuclear Operations, Inc. shall prepare the EPU startup test procedure to include the (a) stress limit curve to be applied for evaluating steam dryer performance; (b) specific hold points and their duration during EPU power ascension; (c) activities to be accomplished during hold points; (d) plant parameters to be monitored; (e) inspections and walkdowns to be conducted for steam, feedwater, and condensate systems and components during the hold points; (f) methods to be used to trend plant parameters; (g) acceptance criteria for monitoring and trending plant parameters, and conducting the walkdowns and inspections; (h) actions to be taken if acceptance criteria are not satisfied; and (i) verification of the completion of commitments and planned actions specified in its application and all supplements to the application in support of the EPU license amendment request pertaining to the steam dryer prior to power increase above OLTP. Entergy Nuclear Operations, Inc. shall provide the related EPU startup test procedure sections to the NRC by facsimile or electronic transmission to the NRC project manager prior to increasing power above OLTP.

4. When operating above OLTP, the operating limits, required actions, and surveillances specified in the SDMP shall be met. The following key attributes of the SDMP shall not be made less restrictive without prior NRC approval:

   a. During initial power ascension testing above OLTP, each test plateau increment shall be approximately 80 MWt;

b. Level 1 performance criteria; and

c. The methodology for establishing the stress spectra used for the Level 1 and Level 2 performance criteria.

Changes to other aspects of the SDMP may be made in accordance with the guidance of NEI 99-04.

5. During each of the three scheduled refueling outages (beginning with the spring 2007 refueling outage), a visual inspection shall be conducted of all accessible, susceptible locations of the steam dryer, including flaws left "as is" and modifications.

6. The results of the visual inspections of the steam dryer conducted during the three scheduled refueling outages (beginning with the spring 2007 refueling outage) shall be reported to the NRC staff within 60 days following startup from the respective refueling outage. The results of the SDMP shall be submitted to the NRC staff in a report within 60 days following the completion of all EPU power ascension testing.

7. The requirements of paragraph 4 above for meeting the SDMP shall be implemented upon issuance of the EPU license amendment and shall continue until the completion of one full operating cycle at EPU. If an unacceptable structural flaw (due to fatigue) is detected during the subsequent visual inspection of the steam dryer, the requirements of paragraph 4 shall extend another full operating cycle until the visual inspection standard of no new flaws/flaw growth based on visual inspection is satisfied.

8. This license condition shall expire upon satisfaction of the requirements in paragraphs 5, 6, and 7 provided that a visual inspection of the steam dryer does not reveal any new unacceptable flaw or unacceptable flaw growth that is due to fatigue.

4.0 REGULATORY COMMITMENTS

The licensee made the regulatory commitments shown in the following table. The source of each commitment is shown in the "Reference" column.

No.	COMMITMENT	REFERENCE (Entergy letter number and date)	TYPE		SCHEDULED COMPLETION DATE	NRC COMMENTS
			One-time	Continuing		
1	Steam dryer inspection during RFO.	BVY 04-058 July 2, 2004 (Reference 9)	x		Fall 2005	Commitment modified by Entergy letter BVY 05-083, dated September 10, 2005 (See Commitment No. 23).
2	Steam dryer inspection during RFO.	BVY 04-058 July 2, 2004 (Reference 9)	x		Spring 2007	Commitment modified by Entergy letter BVY 05-083, dated September 10, 2005 (See Commitment No. 23).
3	Steam dryer inspection during RFO.	BVY 04-058 July 2, 2004 (Reference 9)	x		Fall 2008	Commitment modified by Entergy letter BVY 05-083, dated September 10, 2005 (See Commitment No. 23).
4	Perform flow-induced vibration monitoring.	BVY 04-058 July 2, 2004 (Reference 9)	x		EPU implementation	
5	Implement those modifications contained in ISO New England letter of March 12, 2004.	BVY 04-086 August 25, 2005 (Reference 13)	x		Prior to increasing power above CLTP	
6	Install additional capacitor banks.	BVY 04-086 August 25, 2005 (Reference 13)	x		Prior to exceeding 630 MWe (gross)	
7	Provide details regarding Action Item No. 2.	BVY 04-097 September 14, 2004 (Reference 14)	x		September 29, 2004	Per Reference 14, Attachment 1, page 3, "Action Item No. 2" pertains to Entergy providing additional details on the power ascension test plan. Commitment was satisfied by Entergy letter BVY 04-100, dated September 23, 2004 (Reference 16).

No.	COMMITMENT	REFERENCE (Entergy letter number and date)	TYPE		SCHEDULED COMPLETION DATE	NRC COMMENTS
			One-time	Continuing		
8	Provide results of the acoustic analysis model to the NRC staff during a meeting.	BVY 04-097 September 14, 2004 (Reference 14)	x		September 30, 2004	Commitment was satisfied by meeting held on September 29, 2004.
9	Perform a detailed inspection of the steam dryer during the next scheduled refueling outage. The inspection will be performed in accordance with the VYNPS vessel internals monitoring program and recommendations of GE SIL 644, Supplement 1.	BVY 04-097 September 14, 2004 (Reference 14)	x		Fall 2005	Commitment modified by Entergy letter BVY 05-083, dated September 10, 2005 (See Commitment No. 23).
10	Perform a detailed inspection of the steam dryer during two subsequent refueling outages. The inspections will be performed in accordance with the VYNPS vessel internals monitoring program and recommendations of GE SIL 644, Supplement 1.	BVY 04-097 September 14, 2004 (Reference 14)	x		Spring 2007 and Fall 2008	Overlap of scope shown in Commitment No. 23.
11	Provide the NRC staff the results of the steam dryer inspections during the next two refueling outage.	BVY 04-097 September 14, 2004 (Reference 14)	x		After Fall 2005 and Spring 2007	Commitment modified by Entergy letter BVY 05-083, dated September 10, 2005 (See Commitment No. 23).

No.	COMMITMENT	REFERENCE (Entergy letter number and date)	TYPE		SCHEDULED COMPLETION DATE	NRC COMMENTS
			One-time	Continuing		
12	Prior to implementing, discuss changes to the long term monitoring plan for the steam dryer with the NRC staff.	BVY 04-097 September 14, 2004 (Reference 14)	x		Indeterminate	Commitment was satisfied by meeting held on September 29, 2004.
13	Add the steam dryer to the VYNPS Vessel Internals Inspection Program as an augmented exam.	BVY 04-097 September 14, 2004 (Reference 14)	x		Fall 2005	Commitment was satisfied as documented in Section 4OA5 of Reference 74.
14	Existing and new steam dryer indications will be evaluated using ASME Code, Section XI criteria as guidance, along with industrial standards and practices. Fracture mechanics analysis will be incorporated into the evaluation as appropriate. Any indication predicted to result in an unacceptable failure will be repaired in accordance with approved procedures. Technical justification will be documented for all unrepaired flaw indications left in service for the next cycle.	BVY 04-097 September 14, 2004 (Reference 14)	x		Indeterminate	
15	Implement the BWROG guidelines on moisture carryover monitoring when issued.	BVY 04-097 September 14, 2004 (Reference 14)	x		Indeterminate	

No.	COMMITMENT	REFERENCE (Entergy letter number and date)	TYPE		SCHEDULED COMPLETION DATE	NRC COMMENTS
			One-time	Continuing		
16	Implement flow induced vibration and steam dryer monitoring, including associated evaluation as necessary during EPU power ascension testing as described in Entergy letter BVY 04-100.	BVY 04-100 September 23, 2004 (Reference 16)	x		During EPU power ascension testing	
17	Discuss details of the acoustic analysis and steam dryer power ascension test acceptance criteria at a meeting with NRC staff.	BVY 04-100 September 23, 2004 (Reference 16)	x		September 30, 2004	Commitment was satisfied by meeting held on September 29, 2004.
18	Implement BWROG operational (moisture carryover) response guidance.	BVY 04-100 September 23, 2004 (Reference 16)	x		During EPU power ascension testing	
19	Revise the MOV Periodic Verification Program to include periodic at-the-valve testing and formalize the process for DC motor trending.	BVY 04-101 September 30, 2004 (Reference 17)	x		December 1, 2004	Attachment 3 to Entergy letter BVY 05-083, dated September 10, 2005 (Reference 33), states that this commitment is complete.
20	Verify the RCIC start time assumed in the SCCA and complete training of operations crews on the revised procedure.	BVY 04-107 September 30, 2004 (Reference 18)	x		December 1, 2004	Entergy letter BVY 04-131, dated December 8, 2004 (Reference 23), states that this commitment is complete.

No.	COMMITMENT	REFERENCE (Entergy letter number and date)	TYPE		SCHEDULED COMPLETION DATE	NRC COMMENTS
			One-time	Continuing		
21	Entergy has established administrative controls to assure performance of a once per operating cycle tabletop review of the procedures that complete the actions to repower a VYNPS 4KV bus from the VHS. Pursuant to discussions with co-host REMVEC, a system-wide annual tabletop review will take place in October 2005. During this meeting, Entergy will lead a tabletop review of all actions required to support the restoration of 4KV AC to VYNPS. This review will review the interfaces with the operator of VHS and the regional grid operator to verify that roles and responsibilities and timelines are understood and that there have been no changes that would impact the assumption in the VYNPS SBO coping strategy. Entergy will also provide the participants with additional insights regarding offsite power issues for nuclear power stations including plant response to and consequences of an SBO.	BVY 05-072 August 1, 2005 (Reference 31)		x	October 31, 2005*	* Date represents first tabletop review. First tabletop review was performed on October 25, 2005.

No.	COMMITMENT	REFERENCE (Entergy letter number and date)	TYPE		SCHEDULED COMPLETION DATE	NRC COMMENTS
			One-time	Continuing		
22	Training on the changes to OP 2124 and OT 3122 is currently underway. Once the training cycle is complete (scheduled for September 1, 2005), the procedures will be revised and issued for use. Various operating, surveillance and administrative procedures will be revised to incorporate a higher condensate storage tank inventory limit as either a precaution or an administrative limit by October 15, 2005.	BVY 05-072 August 1, 2005 (Reference 31)	x		October 15, 2005	Commitment was satisfied as documented in Section 4OA5 of Reference 74.
23	Visual inspection of steam dryer.	BVY 05-083 September 10, 2005 (Reference 33)	x		RFO-26, RFO-27, and RFO-28	RFO-26, 27, and 28, are scheduled for spring 2007, fall 2008, and spring 2010, respectively.
24	Modification of the four susceptible isokinetic sample probes in the condensate and feedwater systems.	BVY 05-083 September 10, 2005 (Reference 33)	x		RFO-25	Commitment was satisfied as documented in Section 4OA5 of Reference 74.

No.	COMMITMENT	REFERENCE (Entergy letter number and date)	TYPE		SCHEDULED COMPLETION DATE	NRC COMMENTS
			One-time	Continuing		
25	With regard to License Condition 3.M, "Potential Adverse Flow Effects," Entergy will provide information on plant data, evaluations, walkdowns, inspections, and procedures associated with the individual requirements of that license condition to the NRC staff prior to increasing power above 1593 MWt or each specified hold point, as applicable. If any safety concerns are identified during the NRC staff review of the provided information, Entergy will not increase power above 1593 MWt or the applicable hold point, and the specific requirements in the license condition will not be satisfied.	BVY 05-096 October 17, 2005 (Reference 37)		x	Prior to increasing power above 1593 MWt	
26	Implement a plant modification which will automatically trip a reactor feedwater pump upon a loss of a condensate pump at EPU conditions.	BVY 05-101 November 2, 2005 (Reference 42)	x		Fall 2005	

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above regulatory commitment(s) are best provided by the licensee's administrative processes, including its commitment management program. The above regulatory commitments do not warrant the creation of regulatory requirements (items requiring prior NRC approval of subsequent changes).

5.0 RECOMMENDED AREAS FOR INSPECTION

As described above, the NRC staff has conducted an extensive review of the licensee's plans and analyses related to the proposed EPU and concluded that they are acceptable. The NRC staff's review has identified the following areas for consideration by the NRC inspection staff during the licensee's implementation of the proposed EPU. These areas are recommended based on past experience with EPUs, the extent and unique nature of modifications necessary to implement the proposed EPU, and new conditions of operation necessary for the proposed EPU. They do not constitute inspection requirements, but are intended to give inspectors insight into important bases for approving the EPU.

- Actions associated with licensee commitments shown in SE Section 4.0.

- Licensee corrective actions associated with EPU-related findings in NRC inspection report dated December 2, 2004 (Reference 55). The EPU-related findings are discussed in SE Section 1.6. Resolution of the issues as related to the EPU amendment review is discussed in SE Sections 2.2.4, 2.3.5, 2.6.5, and 2.11.

- As discussed in SE Section 2.2.4.2, and shown on page 50 of Attachment 1 to Supplement 5 of the EPU request (Reference 6), the licensee stated that the VYNPS air-operated valves were reviewed to identify any valves that could be adversely affected by EPU operation. The results of the licensee's review determined that there is an increase in inlet pressure, operating and shutoff pressure differential pressure for the high pressure feedwater heater drain valves and the moisture separator drain tank control valves. The licensee's submittal indicated that these changes were being evaluated for effect on drain valve operators. The results of this evaluation and any equipment modifications are recommended for inspection.

- As discussed in SE Section 2.2.4.2, and shown on page 50 of Attachment 1 to Supplement 5 of the EPU request (Reference 6), the licensee stated that, of the motor-operated valves previously identified as potentially susceptible to pressure locking or thermal binding, only one valve, RHR drywell spray valve V10-26A, was calculated to experience an increase in accident condition environmental temperature and that an evaluation would be performed to determine if any setpoint adjustment is necessary. The results of this evaluation and any setpoint adjustment are recommended for inspection.

- As discussed in SE Section 2.2.6.2.2, and shown on page 6 of Attachment 1 to Supplement 15 of the EPU request (Reference 16), the licensee stated that additional system component evaluations will be performed prior to EPU implementation to identify and evaluate plant components not currently susceptible to flow-induced vibration, but may be susceptible at EPU conditions. The results of these evaluations, including completion of licensee baseline walkdowns and inspections are areas recommended for inspection.

In addition to the recommended areas for inspection listed above, NRC Inspection Procedure 71004, "Power Uprates" (Reference 67), provides guidance for conducting inspections

associated with power uprate amendments including considerations for selecting inspection samples.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Vermont State official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, 51.33, and 51.35, a draft Environmental Assessment and finding of no significant impact was prepared and published in the *Federal Register* on November 9, 2005 (70 FR 68106). The draft Environmental Assessment provided a 30-day opportunity for public comment. The NRC staff received comments which were addressed in the final environmental assessment. The final Environmental Assessment was published in the *Federal Register* on January 27, 2006 (71 FR 4614). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

8.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

8.1 Background

The Commission issued a "Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for a Hearing" for the proposed VYNPS EPU amendment in the *Federal Register* on July 1, 2004 (69 FR 39976). This Notice provided 60 days for the public to request a hearing. On August 30, 2004, the Vermont Department of Public Service (DPS) and the New England Coalition (NEC) filed requests for hearing in connection with the proposed VYNPS EPU amendment. By Order dated November 22, 2004, the Atomic Safety and Licensing Board (ASLB) granted those hearing requests and by Order dated December 16, 2004, the ASLB issued its decision to conduct the hearing using the procedures in 10 CFR Part 2, Subpart L, "Informal Hearing Procedures for NRC Adjudications." A hearing on the proposed EPU amendment has not yet been scheduled.

The Commission's regulations in 10 CFR 2.1202 state that:

> During the pendency of any hearing under this subpart, consistent with the NRC staff's findings in its review of the application or matter which is the subject of hearing and as authorized by law, the NRC staff is expected to issue its approval or denial of the application promptly, or take other appropriate action on the underlying regulatory matter for which a hearing is provided.

In addition, 10 CFR 2.1202 states that the NRC staff's actions on the matter is effective upon issuance by the staff, except for certain types of applications not applicable here, or in 10 CFR Part 50 licensing actions that involve significant hazards considerations as defined in 10 CFR 50.92.

The Commission's regulations in 10 CFR 50.91(a)(2) state that:

> The Commission may publish in the *Federal Register* under §2.105 an individual notice of proposed action for an amendment for which it makes a proposed determination that no significant hazards consideration is involved...

Further, the regulations provide that such notice should contain the staff's proposed determination, provide a brief description of the amendment and the facility involved, solicit public comments thereon, and provide for a 30-day comment period.

The Commission's regulations in 10 CFR 50.91(a)(4) state that:

> Where the Commission makes a final determination that no significant hazards consideration is involved and that the amendment should be issued, the amendment will be effective on issuance, even if adverse public comments have been received and even if an interested person meeting the provisions for intervention called for in §2.309 of this chapter has filed a request for a hearing. The Commission need hold any required hearing only after it issues an amendment, unless it determines that a significant hazards consideration is involved, in which case the Commission will provide an opportunity for a prior hearing.

The Commission's regulations in 10 CFR 50.92(c) state that the Commission may make a final determination that a proposed license amendment involves no significant hazards consideration (NSHC) if the operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

Pursuant to 10 CFR 50.91, the NRC staff made a proposed determination that the VYNPS EPU amendment request involves NSHC. A "Notice of Consideration of Issuance of Amendment to Facility Operating License and Proposed No Significant Hazards Consideration Determination" was issued in the *Federal Register* on January 11, 2006 (71 FR 1774). The notice provided a 30-day opportunity for public comment. The NRC staff has received comments on the proposed NSHC determination. The comments and the staff's consideration thereof are discussed in SE Section 8.2.

8.2 Public Comments on Proposed NSHC Determination

Public comments were received within the 30-day comment period in response to the NRC staff's January 11, 2006, notice of proposed NSHC determination. Theses comments and the staff's responses are grouped into subject area categories and are addressed below.

8.2.1 Independent Safety Assessment

Public Comments

Comments were received which stated that an independent safety assessment should be performed at VYNPS. Some of the comments elaborated that such an assessment should be like the one performed at the Maine Yankee plant in 1996. The comments cited various reasons why such an assessment was necessary, however, the primary consideration was that the assessment was needed to provide assurance that VYNPS could operate safely under uprated power conditions.

A letter from one commenter enclosed petitions from citizens, and resolutions and letters from legislative bodies, municipal governments, individual officials and legislators all calling for the performance of an independent safety assessment at VYNPS. The commenter requested that the enclosures to its letter be made a part of the public record and requested that the NRC provide a response to these documents.

NRC Staff Response

In a letter to the Vermont Public Service Board (PSB) dated May 4, 2004, NRC Chairman Diaz described the NRC's approach in response to the PSB's request for an independent assessment of VYNPS. As noted in the letter, the NRC staff had concluded that the staff's detailed technical review of the proposed EPU amendment, combined with the inspections prescribed by the reactor oversight process, as enhanced by an improved engineering inspection, would provide the most effective method of informing the staff decision on whether VYNPS could operate safely under uprated power conditions. Section 1.6 of this SE, and other sections referred to therein, discuss the results of the engineering inspection that was completed in September 2004. The staff has concluded that the engineering inspection that was performed at VYNPS met the intent of the PSB's request for an independent assessment and that a further assessment, as requested by the commenters, is not warranted.

As requested by the PSB, the NRC's Advisory Committee on Reactor Safeguards (ACRS) reviewed the engineering inspection results during its evaluation of the VYNPS EPU request. The ACRS is a statutory committee that reports directly to the Commission and is structured to provide a forum where experts representing many technical perspectives can provide advice that is factored into the NRC's decision-making process. The ACRS Subcommittee on Power Uprates held a meeting on November 15 and 16, 2005, in Brattleboro, Vermont to receive input from the public, Entergy, and the NRC staff regarding the proposed EPU. During this meeting the NRC staff provided the results of the engineering inspection, including a discussion of all relevant inspection findings. In a letter to NRC Chairman Diaz dated January 4, 2006, the ACRS recommended approval of the VYNPS EPU amendment request. As noted in the letter, the ACRS concluded, in part, based on the results of the inspection that was performed and the performance of VYNPS as determined by the NRC's reactor oversight process, that a more extensive inspection is not warranted.

With respect to the request from a commenter that the enclosures to its letter be made a part of the public record, all comments received in response to the NRC staff's proposed NSHC determination will be added to the NRC's Agencywide Documents Access and Management System (ADAMS) as publicly-available documents. Some of these documents are letters that were previously submitted to the NRC and a response thereto was already provided. Some of the documents are letters previously sent to other organizations (e.g., the Vermont PSB). Issues within these documents, pertinent to the scope of the NRC staff's NSHC determination for the proposed EPU at VYNPS, are addressed under the relevant subject area categories in Section 8.2 of this SE.

8.2.2 Age of the Plant and Recent Operational Events

Public Comments

Comments were received which raised concerns regarding VYNPS operations at uprated power conditions, considering the age of the plant and recent operational events. The comments included concerns regarding age-related failures of components, cracking in plant components such as the steam dryer and the reactor vessel, potential deficiencies such as leaking MSIVs, and recent operational events initiated by component failures (e.g., main transformer fire and scram due to an electrical insulator failure).

NRC Staff Response

The VYNPS was granted, consistent with NRC regulations, a 40-year operating license in 1972. The NRC requires licensees to test, monitor, and inspect the condition of safety equipment and to maintain that equipment in reliable operating condition over the operating life of the plant. The NRC also requires licensees to continuously correct deficiencies that could impact plant safety (e.g., leaking valves, degraded or failed components due to aging or operational events). Over the years, the licensee has replaced or overhauled plant equipment as needed. Where appropriate, the licensee has also upgraded equipment or installed new equipment to replace or supplement original systems. The testing, monitoring, inspection, maintenance, and replacement of plant equipment provides reasonable assurance that this equipment will perform its intended safety functions during the 40-year license period. This conclusion applies both to operations under the current license and operations under EPU conditions.

8.2.3 Safe Operation of the Plant and Reduced Safety Margins

Public Comments

Comments were received which expressed general concerns regarding safe operation of the plant. Some of the comments stated that the proposed EPU would significantly reduce the plant safety margins based on considerations such as increased stress to plant components and decreased operator response times.

NRC Staff Response

The NRC's safety regulations are based on the Atomic Energy Act of 1954, as amended, and require a finding of reasonable assurance that the activities authorized by an operating license (or an amendment thereto) can be conducted without endangering the health and safety of the public, and that such activities will be conducted in compliance with the NRC's regulations. With respect to the proposed EPU, and as discussed in Section 9.0 of this SE, the Commission has concluded that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner and that the authorized activities will be conducted in compliance with the NRC's regulations.

The margin of safety is related to confidence in the ability of the fission product barriers (i.e., fuel cladding, reactor coolant pressure boundary, and containment) to limit the level of radiation dose to the public. The NRC staff evaluated the impact of the proposed EPU on the fission product barriers and concluded that the structural integrity of the fission product barriers would be maintained under EPU conditions. As such, the proposed amendment would not degrade confidence in the ability of the barriers to limit the level of radiation dose to the public. A detailed discussion of the NRC staff's review regarding the impact of the proposed EPU on the margin of safety is discussed below in SE Section 8.3.

With respect to the comments regarding increased stress to plant components, the NRC staff evaluated the impact of the expected changes to plant parameters as a result of the proposed EPU (e.g., increase in temperatures, flow rates, vibration) for the applicable systems or components as described in the relevant sub-sections in SE Section 2.0, and concluded that there was reasonable assurance that plant systems and components would continue to perform their intended safety functions under EPU conditions. Further, as summarized in SE Section 8.3, and as discussed in the previous paragraph, the staff concluded that the structural integrity of the fission product barriers would be maintained under EPU conditions (i.e., even given the increased stress to plant components). The staff concluded that, since there is reasonable assurance that the fission product barriers will limit the level of radiation dose to the public, the proposed EPU would not involve a significant reduction in a margin of safety.

With respect to the comments regarding decreased operator response times, the NRC staff evaluated the impact of the proposed EPU on human performance as described in SE Section 2.11. Although some operator response times would be reduced, the staff concluded that sufficient margin exists to allow the operators to perform their required actions under EPU conditions.

8.2.4 Risk Assessment

Public Comments

Comments were received which stated that the risk assessment in the NRC's draft SE for the proposed VYNPS EPU is flawed. The following reasons were cited:

- The risk assessment is based on the assumption that the plant is brand new, relies upon as-designed safety margins or unverified licensee representations, and does not reflect real experience.
- The risk assessment does not consider potential accident consequences.
- The staff had indicated during the ACRS meetings that there are better methods to evaluate system integrity under uprate conditions, but the NRC is not using them.

NRC Staff Response

The proposed VYNPS EPU was not submitted as a risk-informed license amendment request. As stated in SE Section 2.13, a risk assessment was conducted to determine if "special circumstances" are created by the proposed EPU that would potentially rebut the presumption of adequate protection provided by licensee compliance with the current regulatory requirements. The NRC uses an integrated decisionmaking process that considers the results of risk assessments; however, the acceptability of license amendment requests is not determined solely by the numerical results of a risk assessment. As concluded in SE Section 2.13, the staff determined that the proposed EPU does not create "special circumstances" that rebut the presumption of adequate protection of public health and safety.

Appropriate care has been taken by the licensee to ensure that the risk assessment reflects the as-built and as-operated configuration of VYNPS. The risk assessment models were constructed by reviewing engineering drawings of the plant systems, which have been periodically revised to reflect plant modifications. Risk analysts walked down the plant to ensure that the information obtained from the drawings was accurate and current. The risk assessment uses the actual operating experience of VYNPS and other nuclear power plants to determine equipment failure probabilities and initiating event frequencies. Specifically, records of equipment failures and off-normal plant events at VYNPS and other nuclear power plants have been compiled and statistically analyzed to support the risk assessment. As a result, as stated in SE Section 2.13, the risk assessment has sufficient scope, level of detail, and technical adequacy to support the risk evaluation of the proposed EPU.

The risk assessment evaluated the VYNPS in terms of two risk metrics, core-damage frequency and large early release frequency, which are appropriate and accepted surrogates for considering potential accident consequences. Core-damage frequency is a surrogate for the statistically expected latent cancer mortality risk due to a reactor accident. Large early release frequency is a surrogate for the statistically expected prompt cancer mortality risk due to a reactor accident. These measures of accident consequences (latent and prompt mortality risks) are cited in the Commission's Safety Goal Policy (51 FR 28044; August 4, 1986, as corrected and republished at 51 FR 30028; August 21, 1986). The Safety Goal Policy was developed with extensive public participation, including (a) two workshops held in 1981 that featured discussions among knowledgeable persons drawn from industry, public interest groups, universities, and elsewhere who represented a broad range of perspectives and disciplines; and (b) public comments on the proposed policy statement. Additionally, the Commission had the benefit of further comments from its Advisory Committee on Reactor

Safeguards and by senior NRC management. The licensee's risk assessment considered potential accident consequences consistent with the Commission's Safety Goal Policy, and has been determined to be acceptable, as set forth in SE Section 2.13.

With respect to the public comment that the risk assessment, discussed in the NRC staff's draft SE, relies upon unverified licensee representations, the staff reviewed the specific assumptions and methods used by the licensee as detailed in SE Section 2.13, and as such, the staff disagrees with the comment. The staff concluded that the licensee had adequately modeled and/or addressed the potential impacts associated with the implementation of the proposed EPU. In addition, with respect to the licensee's risk assessment associated with crediting containment accident pressure, the staff performed an independent scoping risk evaluation that confirmed that the licensee's risk assessment results were reasonable.

With respect to the public comment that the staff had indicated during the ACRS meetings that there are better methods to evaluate system integrity under uprate conditions, the staff has reviewed the transcripts for the ACRS meetings related to the VYNPS EPU, held in November and December 2005, and could not identify the specific statement to which this public comment is directed. However, the context of the comment appears to relate to the risk analysis performed by the staff related to crediting of containment accident pressure. As discussed in SE Section 2.13, both the staff's and Entergy's analyses provide a conservative estimate of the increase in core damage frequency. Further, although there were some differences in the assumptions used, the results obtained by Entergy and the staff both indicated that the change in risk, due to crediting of containment accident pressure, is small and meets the RG 1.174 risk acceptance guidelines. Accordingly, the risk analyses are acceptable.

8.2.5 Steam Dryer Power Ascension Testing

Public Comments

Comments were received regarding the planned VYNPS EPU power ascension test program related to the steam dryer. Specifically, the comments noted that the NRC's draft SE for the VYNPS EPU had indicated that significant uncertainty exists regarding the licensee's method for calculating stress on the steam dryer. As such, the comments expressed the concern that the testing was experimental in nature, and that testing the dryer while the reactor is online reduces safety margins.

NRC Staff Response

The NRC staff's evaluation regarding the steam dryer is discussed in SE Section 2.2.6. The staff's evaluation stated, in part, that:

> The NRC staff has reviewed the information provided by the licensee in support of its analysis of the structural integrity of the VYNPS steam dryer under EPU conditions, and for monitoring steam dryer loads and performance during plant operation. Although significant uncertainty exists regarding the licensee's method for calculating specific stress values on the VYNPS steam dryer from its CFD and ACM analyses, the

> licensee's current MSL instrumentation suggests minimal excitation of the pressure frequency spectra in the MSLs at CLTP conditions. As a result, the staff finds that the licensee has demonstrated that the flow-induced stress imposed on the VYNPS steam dryer at CLTP conditions is within the fatigue stress limits provided in the ASME Code. However, the available margin to those stress limits is not readily verifiable.

Further, the staff's evaluation concluded that power ascension should be conducted in carefully monitored stages. The staff stated that:

> In light of the large uncertainties in the CFD and ACM analyses and the fact that the ACM analysis has calculated the steam dryer pressure loads only at CLTP, the NRC staff determined that the licensee needs to closely monitor MSL strain gage data and other plant data as the reactor power is raised at VYNPS such that the ACM loads can be calculated at the increased power level to verify that the structural limits for the steam dryer are not reached. For example, the staff concluded that the new 32 MSL strain gages need to be monitored frequently during power ascension above CLTP for increasing pressure fluctuations in the steam lines. Hold points need to be established at 105%, 110%, and 115% of CLTP to collect plant data, conduct plant inspections and walkdowns, and evaluate the plant data for steam dryer performance. The time period for each hold point will need to be sufficient to complete all activities specified in the startup test procedure for the applicable hold point. Sufficient information and time will need to be provided to the NRC staff to determine whether any safety concerns exist prior to increasing power above each hold point. If any frequency peak from the MSL strain gage data exceeds the limit curve established by the licensee prior to operation above CLTP, the unit needs to be returned to a power level where the limit curve is not exceeded. The licensee would then resolve the uncertainties in the steam dryer analysis prior to further increases in reactor power. In the subsequent engineering evaluation, peak responses that fall within the ±10% frequency uncertainty band need to be considered as part of an adequate structural analysis. Further, the potential effect of the skirt in the steam dryer FEM on the stresses in the steam dryer components needs to be addressed. In addition to evaluating the MSL strain gage data, reactor pressure vessel water level instrumentation or MSL piping accelerometers need to be monitored frequently to help identify any resonance frequencies not captured by the MSL strain gage data and ACM analysis. If resonance frequencies are identified as increasing significantly above nominal levels established at CLTP conditions, power ascension needs to be stopped until an evaluation of continued steam dryer integrity is performed to demonstrate that no safety concerns exist.

As noted in the NRC staff's evaluation of the steam dryer, a license condition will be added to the VYNPS Facility Operating License as shown in SE Section 3.17.3. The license condition provides requirements for monitoring, evaluating, and taking prompt action in response to potential adverse flow effects as a result of operation at EPU conditions. The intent of the license condition is to (1) confirm the licensee's predictions regarding the hydrodynamic loads on the steam dryer; (2) confirm the licensee's predictions regarding the acoustic pressure loads on the steam dryer; and (3) confirm the safe operation of VYNPS during power ascension

above CLTP. The staff has determined that this license condition adequately addresses the staff's findings regarding uncertainties in the steam dryer analyses.

8.2.6 Containment Accident Pressure

Public Comments

Comments were received which stated that even one mis-positioned valve could cause a loss of containment integrity. The comments cited specific operating experience at VYNPS in which a valve was left in the wrong position for 9 years. The comments stated that this operating experience raises a concern regarding crediting containment accident pressure in certain accident analyses (i.e., the licensee's analysis takes credit for containment accident pressure which implicitly requires containment integrity). The comments also stated that this issue represents a significant reduction of safety margins, and that an independent safety assessment is needed to ensure containment integrity. A comment was also received which stated that any one of several single failures within the containment system would also cause the failure of the emergency core cooling system.

NRC Staff Response

In general, for piping systems penetrating containment, the loss of containment integrity requires that a minimum of two valves in the line be open. The VYNPS specific operating experience cited by the comments was documented in Entergy's Licensee Event Report (LER) 2005-002-00, dated December 1, 2005. The LER stated that two normally-closed air operated valves and a normally-closed manual sample valve located downstream of the mis-positioned valve "provided reasonable assurance that effective isolation for this flow path was maintained during plant operation." As such, containment integrity was maintained, and this experience does not support the commenters' concern.

The NRC staff's evaluation regarding the crediting of containment accident pressure is discussed in SE Section 2.6.5. The staff's evaluation noted that VYNPS containment integrity is continuously monitored during normal operation and that the plant operators would take the appropriate action in accordance with the plant's procedures if there were signs of degradation of containment integrity. This concern has been adequately addressed.

The NRC staff's response to public comments regarding a potential reduction in safety margins is addressed in SE Section 8.2.3. The staff's response to comments regarding the need for an independent safety assessment is addressed in SE Section 8.2.1.

8.2.7 High-Pressure Coolant Injection (HPCI) System Operating Experience

Public Comments

Comments were received which stated that repeated events involving inoperability of the VYNPS HPCI system call into question defense-in-depth. This operating experience was cited as a reason that an independent safety assessment was needed.

NRC Staff Response

The only recent event at VYNPS involving HPCI system inoperability was on December 22, 2005. The system was declared inoperable, in accordance with the VYNPS Technical Specification (TS) 3.5.E.2, due to problems with the flow control instrumentation. The problems were resolved and the system was returned to service the same day. VYNPS TS 3.5.E.2 allows the HPCI system to be inoperable for up to 14 days provided that certain other systems that would provide core cooling are operable (which was the case for this event). As such, adequate core cooling was ensured during the time period that the HPCI system was inoperable. If the other core cooling systems are not operable, TS 3.5.E.2 would require that an orderly shutdown be commenced and that reactor pressure be reduced within 24 hours.

The NRC's Reactor Oversight Process (ROP) requires the licensee to monitor the availability of the HPCI system using the NRC's Performance Indicator (PI) program and to report PI information to the NRC each calender quarter. Since the inception of the ROP in 2000, the VYNPS HPCI system PI has remained Green, indicating an overall availability of greater than or equal to 97.5%. Green PIs indicate a very low risk significance and therefore have little or no impact on safety. Based on the HPCI system PI data, the staff finds no evidence regarding HPCI system availability that raises a question regarding defense-in-depth.

The staff response to comments regarding the need for an independent safety assessment is addressed in SE Section 8.2.1.

8.2.8 Spent Fuel Storage at VYNPS

Public Comments

A comment was received which raised a concern regarding the amount of spent fuel stored onsite at VYNPS and the potential consequences due to an accident.

NRC Staff Response

The current licensing basis for VYNPS, as specified in TS 5.5, allows a specified maximum number of spent fuel assemblies to be stored in the spent fuel pool. As discussed in SE Section 2.8.6.2, the proposed EPU does not change the amount of spent fuel assemblies that may be stored in the VYNPS spent fuel pool.

The impact of the proposed EPU on the radiological consequences of a design basis accident, including a fuel handling accident (FHA), is addressed in SE Section 2.9.2. The licensee's FHA analysis, which was performed as part of a previously approved amendment request, postulates the dropping of a spent fuel assembly during refueling. As discussed in SE Section 2.9.2, the NRC staff has concluded that the dose criteria in 10 CFR 50.67 would be met in the event of a design basis accident, including the FHA, at EPU conditions.

8.2.9 Miscellaneous Issues

Public Comments

Comments were received pertaining to the following miscellaneous issues:

- Use of conservation or alternative energy sources other than nuclear power.
- Nuclear waste disposal and dry cask storage alternatives.
- Limits on liability in the event of a nuclear accident.
- The licensee's financial qualifications and interests, and its concern for public health and safety in Vermont.
- Social/economic conditions related to the need and/or cost of additional nuclear power generation.
- The adequacy of offsite emergency preparedness.

NRC Staff Response

The issues raised pertain to VYNPS operation generally, and are not within the scope of the NRC staff's NSHC determination for the proposed EPU at VYNPS. As such, no specific response is provided herein.

8.3 Final NSHC Determination

The NRC staff has completed its evaluation of the licensee's proposed EPU amendment as discussed in SE Section 2.0 above and has considered the comments submitted on the staff's proposed NSHC determination. Based on its evaluation, the staff has made a final determination that the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated; does not create the possibility of a new or different kind of accident from any accident previously evaluated; and does not involve a significant reduction in a margin of safety. The following evaluation in relation to the three standards of 10 CFR 50.92(c) explains the staff's final NSHC determination.

First Standard

Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

As discussed in the licensee's application dated September 10, 2003, the VYNPS EPU analyses, which were performed at or above EPU conditions, included a review and evaluation of the structures, systems, and components (SSCs) that could be affected by the proposed change. The licensee reviewed plant modifications and revised operating parameters, including operator actions, to confirm acceptable performance of plant SSCs under EPU conditions. On this basis, the licensee concluded that there is no increase in the probability of accidents previously evaluated.

Further, as also discussed in the licensee's application, while not being submitted as a risk-informed licensing action, the proposed amendment was evaluated by the licensee from a risk perspective. Using the NRC guidelines established in Regulatory Guide (RG) 1.174, and the calculated results from the VYNPS Level 1 and 2 probabilistic safety analyses, the best estimate for the core damage frequency (CDF) increase due to the proposed EPU is 3.3 E-7 per year (an increase of 4.2 percent over the pre-EPU CDF of 7.77 E-6 per year). The best estimate for the large early release frequency (LERF) increase due to the proposed EPU is 1.1 E-7 per year (an increase of 4.9 percent over the pre-EPU LERF of 2.23 E-6 per year). The NRC staff concludes, based on review of the licensee's risk evaluation and the acceptance guidelines in RG 1.174, that the proposed amendment would not involve a significant increase in the probability of an accident previously evaluated.

The NRC staff's evaluation of the proposed amendment included review of the SSCs that could be affected by the proposed change. This review included evaluation of plant modifications, revised operating parameters, changes to operator actions and procedures, the EPU test program, and changes to the plant TSs. Based on this review, the staff concludes that there is reasonable assurance that the SSCs important to safety will continue to meet their intended design basis functions under EPU conditions. Therefore, the staff concludes that there is no significant change in the ability of these SSCs to preclude or mitigate the consequences of accidents.

The NRC staff's evaluation also reviewed the impact of the proposed EPU on the radiological consequences of design-basis accidents for VYNPS. The staff's review concluded that dose criteria in 10 CFR 50.67, as well as the applicable acceptance criteria in Standard Review Plan Section 15.0.1, would continue to be met at EPU conditions.

The NRC staff concludes, based on review of the SSCs that could be affected by the proposed amendment and review of the radiological consequences, that the proposed amendment would not involve a significant increase in the consequences of an accident previously evaluated.

Based on the above, the NRC staff concludes that the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

Second Standard

Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

As stated above, the NRC staff's evaluation of the proposed amendment included review of the SSCs that could be affected by the proposed change. This review included evaluation of plant modifications, revised operating parameters, changes to operator actions and procedures, the EPU test program, and changes to the plant TSs. Based on this review, the staff concludes that the proposed amendment would not introduce any significantly new or different plant equipment, would not significantly impact the manner in which the plant is operated, and would not have any significant impact on the design function or operation of the SCCs involved. The staff's review did not identify any credible failure mechanisms, malfunctions, or accident initiators not already considered in the VYNPS design and licensing bases. Consequently, the staff concludes that the proposed change would not introduce any failure mode not previously analyzed.

Based on the above, the NRC staff concludes that the proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

Third Standard

Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No

As discussed in the licensee's application, continuing improvements in analytical techniques based on several decades of boiling-water reactor safety technology, plant performance feedback, operating experience, and improved fuel and core designs, have resulted in a significant increase in the design and operating margin between the calculated safety analyses results and the current plant licensing limits. The NRC staff's review found that the proposed EPU will reduce some of the existing design and operational margins. However, safety margins are considered to not be significantly reduced if: (1) applicable regulatory requirements, codes and standards or their alternatives approved for use by the NRC, are met, and (2) if safety analysis acceptance criteria in the licensing basis are met, or if proposed revisions to the licensing basis provide sufficient margin to account for analysis and data uncertainty.

Margin of safety is related to confidence in the ability of the fission product barriers (i.e., fuel cladding, reactor coolant pressure boundary (RCPB), and containment) to limit the level of radiation dose to the public. The NRC staff evaluated the impact of the proposed EPU on the fission product barriers as discussed below.

The NRC staff evaluated the impact of the proposed EPU to assure that acceptable fuel damage limits are not exceeded. This included consideration of the VYNPS fuel system design, nuclear system design, thermal and hydraulic design, accident and transient analyses, and fuel design limits. The evaluation included an assessment of the margin in the associated safety analyses supporting the proposed EPU. The staff's evaluation found that the licensee's analysis was acceptable based on use of approved analytical methods and that the licensee had included sufficient margin to account for analysis and data uncertainty. In addition, the licensee will continue to perform cycle-specific analysis to confirm that fuel design limits will not be exceeded during each cycle. The staff's evaluation concluded that the applicable VYNPS licensing basis requirements would continue to be met following implementation of the proposed EPU (e.g., draft General Design Criteria (GDC) 6, 7, and 8; and 10 CFR 50.46). Therefore, the NRC staff concludes that fuel cladding integrity would be maintained within acceptable limits under the proposed EPU conditions.

The NRC staff further evaluated the impact of the proposed EPU on the RCPB. The evaluation included an assessment of overpressure protection; structural integrity of the RCPB piping, components, and supports; and structural integrity of the reactor vessel. With respect to overpressure protection, the staff found that the licensee had used an NRC-approved evaluation method, had used the most limiting pressurization event, and had determined that the peak calculated pressure would remain below the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) allowable peak pressure. With respect to structural integrity of the RCPB piping, components, and supports, the staff found that the licensee had performed its evaluation using the process and methodology defined in NRC-approved topical reports. The staff's evaluation concluded that RCPB structural integrity would be maintained at EPU conditions. With respect to structural integrity of the reactor vessel, the staff found that the licensee had implemented an acceptable reactor vessel materials surveillance program in a previously-approved amendment that was based on neutron fluence values acceptable for VYNPS at EPU conditions. In addition, the staff found that the existing pressure-temperature limit curves contained in the TSs would remain bounding for EPU conditions. The staff also found that the methodology used by the licensee to evaluate the loads on the reactor vessel was consistent with an NRC-approved methodology and that the maximum stresses and fatigue usage factors for EPU conditions would be within ASME Code allowable limits. The staff's evaluation regarding the RCPB concluded that the applicable VYNPS licensing basis requirements would continue to be met following implementation of the proposed EPU (e.g., draft GDC 9, 33, 34, and 35; 10 CFR 50.60; and 10 CFR Part 50, Appendices G and H). Therefore, the NRC staff concludes that RCPB structural integrity would be maintained under the proposed EPU conditions.

Finally, the NRC staff evaluated the impact of the proposed EPU on the containment. The staff found that the licensee's analysis used acceptable calculational methods and conservative assumptions and that the containment pressure and temperature under EPU conditions would remain below existing design limits. The staff also evaluated the licensee's proposed change to the licensing basis to credit containment accident pressure to meet the net positive suction head (NPSH) requirements for the emergency core cooling system pumps. The staff found that the licensee's analysis was performed using conservative assumptions and that the credited pressure remains below the containment accident pressure that would be available under EPU

conditions. The staff's evaluation regarding the containment concluded that the applicable VYNPS licensing basis requirements would continue to be met following implementation of the proposed EPU (e.g., draft GDC 10, 41, 49, and 52; and 10 CFR Part 50, Appendix K). Therefore, the NRC staff concludes that containment structural integrity would be maintained under the proposed EPU conditions.

In summary, the NRC staff has concluded that the structural integrity of the fission product barriers (i.e., fuel cladding, RCPB and containment) would be maintained under EPU conditions. As such, the proposed amendment would not degrade confidence in the ability of the barriers to limit the level of radiation dose to the public.

Based on the above, the NRC staff concludes that the proposed change would not involve a significant reduction in a margin of safety.

Conclusion

On the basis of the above evaluation, the NRC staff has made a final determination that the proposed amendment does not involve a significant hazards consideration.

9.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

10.0 REFERENCES

1. Entergy letter (BVY 03-80) to NRC dated September 10, 2003, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Extended Power Uprate."

2. Entergy letter (BVY 03-90) to NRC dated October 1, 2003, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 1, Extended Power Uprate -Technical Review Guidance."

3. Entergy letter (BVY 03-95) to NRC dated October 28, 2003, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 2, Extended Power Uprate - Grid Impact Study."

4. Entergy letter (BVY 03-98) to NRC dated October 28, 2003, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 3, Extended Power Uprate - Updated Information."

5. Entergy letter (BVY 04-009) to NRC dated January 31, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 4, Extended Power Uprate - NRC Acceptance Review."

6. Entergy letter (BVY 04-008) to NRC dated January 31, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 5, Extended Power Uprate - Response to Request for Additional Information."

7. Entergy letter (BVY 04-025) to NRC dated March 4, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 6, Extended Power Uprate - Withholding Proprietary Information."

8. Entergy letter (BVY 04-050) to NRC dated May 19, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 7, Extended Power Uprate - Confirmatory Results."

9. Entergy letter (BVY 04-058) to NRC dated July 2, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 8, Extended Power Uprate - Response to Request for Additional Information."

10. Entergy letter (BVY 04-071) to NRC dated July 27, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 9, Extended Power Uprate - Revised Containment Overpressure Envelope."

11. Entergy letter (BVY 04-074) to NRC dated July 30, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 10, Extended Power Uprate - Response to Request for Additional Information."

12. Entergy letter (BVY 04-081) to NRC dated August 12, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 11, Extended Power Uprate - Response to Request for Additional Information."

13. Entergy letter (BVY 04-086) to NRC dated August 25, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 12, Extended Power Uprate - Revised Grid Impact Study."

14. Entergy letter (BVY 04-097) to NRC dated September 14, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 13, Extended Power Uprate - Response to Steam Dryer Action Items."

15. Entergy letter (BVY 04-098) to NRC dated September 15, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 14, Extended Power Uprate - Response to Request for Additional Information."

16. Entergy letter (BVY 04-100) to NRC dated September 23, 2004, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263, Supplement No. 15, Extended Power Uprate - Response to Steam Dryer Action Item No. 2."

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Date: March 2, 2006

Attachment: List of Acronyms

ATTACHMENT - LIST OF ACRONYMS

ACRONYM	DEFINITION
AAC	alternate alternating current
AC	alternating current
ACM	acoustic circuit model
ACRS	Advisory Committee on Reactor Safeguards
ACS	alternate cooling system
ADAMS	Agencywide Documents Access and Management System
ADS	automatic depressurization system
AEC	Atomic Energy Commission
AL	analytical limit
ALARA	as low as reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
AOP	Abnormal Operating Procedure
AOV	air-operated valve
APLHGR	average planar linear heat generation rate
APRM	average power range monitor
ARAVS	auxiliary and radwaste area ventilation system
ARI	alternate rod injection
ART	adjusted reference temperature
ARTS	Average Power Range Monitor, Rod Block Monitor Technical Specifications
ASHRAE	American Society of Heating, Refrigerating and Air-Conditioning Engineers
ASME	American Society of Mechanical Engineers
ASP	accident sequence precursor

ACRONYM	DEFINITION
AST	alternative source term
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
AV	allowable value
B&PV	Boiler and Pressure Vessel
BIIT	boron injection initiation temperature
BOC	beginning of cycle
BOP	balance-of-plant
BTP	Branch Technical Position
BTU/lbm	British thermal units per pounds mass
BWR	boiling-water reactor
BWROG	Boiling-Water Reactors Owner's Group
BWRVIP	Boiling-Water Reactors Vessels and Internals Project
CADS	containment atmosphere dilution system
cal/gm	calories per gram
CCFP	conditional containment failure probability
CD	complete dependency
CDF	core damage frequency
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
CFS	condensate and feedwater system
CLTP	current licensed thermal power (1593 MWt)
CLTR	constant pressure power uprate licensing topical report (Reference 52)
CO	condensation oscillation
CP	condensate pump
CPPU	constant pressure power uprate

ACRONYM	DEFINITION
CPR	critical power ratio
CRAVS	control room area ventilation system
CRD	control rod drive
CRDA	control rod drop accident
CS	core spray
CSC	containment spray cooling
CST	condensate storage tank
CT	current transformer
CUF	cumulative usage factor
CWS	circulating water system
DBA	design-basis accident
DBLOCA	design-basis loss-of-coolant accident
DC	direct current
DHR	decay heat removal
DIVOM	delta critical power ratio (CPR) over initial CPR versus oscillation magnitude
DR	decay ratio
EAB	exclusion area boundary
ECCS	emergency core cooling system
EDG	emergency diesel generator
EFDS	equipment and floor drainage system
EFPY	effective full-power years
ELTR1	GE Licensing Topical Report NEDC-32424P-A (Reference 63)
ELTR2	GE Licensing Topical Report NEDC-32523P-A (Reference 64)
EOC	end-of-cycle
EOL	end-of-life
EOP	emergency operating procedure

ACRONYM	DEFINITION
EOS	emergency overspeed
EPGs	emergency procedure guidelines
EPRI	Electric Power Research Institute
EPU	extended power uprate
EQ	environmental qualification
ES	extraction steam
ESF	engineered safety features
ESFAS	engineered safety features actuation system
ESFVS	engineered safety feature ventilation system
EVT	enhanced visual testing
F&O	finding and observation
FAC	flow-accelerated corrosion
FEM	finite element model
FES	Final Environmental Statement
FHA	fuel handling accident
FIV	flow-induced vibration
FIVE	fire induced vulnerability evaluation
FMCPR	final minimum critical power ratio
FOL	Facility Operating License
FPC	fuel pool cooling
FPCDS	fuel pool cooling and demineralizer system
FPP	fire protection program
FR	Federal Register
ft	feet
ft-lb	foot-pounds
FW	feedwater

ACRONYM	DEFINITION
FWC	feedwater control
Gd	Gadolinium
GDC	General Design Criteria (or Criterion)
GE	General Electric
GESTAR	General Electric Standard Application for Reactor Fuels
GL	Generic Letter
GMP	Green Mountain Power
GNF	Global Nuclear Fuel
gpm	gallons per minute
GWd/MTU	gigawatt days per metric ton uranium
GWd/ST	gigawatt days per short ton
GWMS	gaseous waste management systems
HCLPF	high confidence of low probability of failure
HCU	hydraulic control unit
HELB	high energy line break
HEM	homogeneous equilibrium model
HEP	human error probability
HEPA	high efficiency particulate air
HP	high pressure
HPCI	high-pressure coolant injection
hr	hour
HVAC	heating, ventilating, and air conditioning
IASCC	irradiation assisted stress corrosion cracking
ICPR	initial critical power ratio
IEEE	Institute of Electrical and Electronics Engineers
IGSCC	intergranular stress corrosion cracking

ACRONYM	DEFINITION
ILRT	integrated leak rate test
IORV	inadvertent open relief valve
IPE	individual plant examinations
IPEEE	individual plant examinations of external events
IR	Inspection Report
ISI	inservice inspection
ISO-NE	Independent System Operator - New England
ISP	integrated surveillance program
KA	kiloamps
ksi	1000 pounds per square inch
kV	kilovolts
kW/ft	kilowatts per foot
LBLOCA	large-break loss-of-coolant accident
LBPCT	licensing basis peak cladding temperature
LCO	limiting condition for operation
LER	licensee event report
LERF	large early release frequency
LES	Large Eddy Simulation
LHGR	linear heat generation rate
LLHS	light load handling system
LOCA	loss-of-coolant accident
LOFW	loss of feedwater
LOOP	loss of offsite power
LP	low pressure
LPCI	low pressure coolant injection
LPRM	local power range monitor

ACRONYM	DEFINITION
LPSI	low pressure safety injection
LPZ	low population zone
LTR	licensing topical report
LWMS	liquid waste management system
MAAP	material access authorization program
MAPLHGR	maximum average planar linear heat generation rate
MBTU/hr	million British thermal units per hour
MCC	motor control center
MCES	main condenser evacuation system
MCS	main condenser system
MELLLA	maximum extended load line limit analysis
mg	milligram
Mlb/ft^2	million pounds per square foot
MOV	motor-operated valve
MS	main steam
MSIV	main steam isolation valve
MSL	main steam line
MSLB	main steam line break
MSSS	main steam supply system
MTU	metric ton uranium
MVA	megavolt amperes
MVAR	megavolt amperes reactive
MWe	megawatts electric
MWH	megawatt hours
MWt	megawatts thermal
n/cm^2	neutrons per centimeter squared

ACRONYM	DEFINITION
NAI	Nuclear Applications, Inc.
NEI	Nuclear Energy Institute
NEPOOL	New England Power Pool
NFPCS	normal fuel pool cooling system
NOS	normal overspeed
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NRR	NRC's Office of Nuclear Reactor Regulation
NSHC	no significant hazards conbsideration
NSSS	nuclear steam supply system
NUMARC	Nuclear Management and Resource Council, Inc.
O&M	Operation and Maintenance
ODCM	offsite dose calculation manual
OLMCPR	operating limit minimum critical power ratio
OLTP	original licensed thermal power
OOS	out of service
ORNL	Oak Ridge National Laboratory
OSP	offsite power
P-T	pressure-temperature
PCT	peak cladding temperature
pf	power factor
PSA	probabilistic safety assessment
PSB	Public Service Board
PSD	power spectral density
psi	pounds per square inch
psia	pounds per square inch absolute

ACRONYM	DEFINITION
psid	pounds per square inch differential
psig	pounds per square inch gauge
Pu	Plutonium
PUSAR	Power Uprate Safety Analysis Report (Attachment 4 to Reference 1)
RACWS	reactor auxiliary cooling water systems
RAI	request for additional information
RBCCWS	reactor building closed cooling water system
RBM	rod block monitor
RCIC	reactor core isolation cooling
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
REMVEC	Rhode Island, Eastern Massachusetts, Vermont Energy control
RES	NRC's Office of Nuclear Regulatory Research
RFO	refueling outage
RFP	reactor feedwater pump
RG	Regulatory Guide
RHR	residual heat removal
RHRSW	residual heat removal service water
RHRSWS	residual heat removal service water system
RIPD	reactor internal pressure difference
RMS	root-mean-square
RPT	recirculation pump trip
RPV	reactor pressure vessel
RR	reactor recirculation
RRS	reactor recirculation system
RRU	reactor recirculation unit

ACRONYM	DEFINITION
RTP	rated thermal power
RVFW	reactor vessel feedwater
RWCS	reactor water cleanup system
RWCU	reactor water cleanup
RWE	rod withdrawal error
RWM	rod worth minimizer
SAGs	severe accident guidelines
SAMG	severe accident management guidelines
SBO	station blackout
SCC	stress corrosion cracking
SCCA	Safe Shutdown Capability Analysis
SDC	shutdown cooling
SDM	shutdown margin
SDMP	Steam Dryer Monitoring Plan
SDP	significance determination process
SE	Safety Evaluation
SFP	spent fuel pool
SFPAVS	spent fuel pool area ventilation system
SFPCCS	spent fuel pool cooling and cleanup system
SFPCS	standby fuel pool cooling system
SGTS	standby gas treatment system
SIL	Services Information Letter
SL	safety limit
SLC	standby liquid control
SLCS	standby liquid control system
SLMCPR	safety limit minimum critical power ratio

ACRONYM	DEFINITION
SLO	single loop operation
SMA	seismic margins assessment
SMT	scale model test
SORV	stuck-open relief valve
SPAR	Standardized Plant Analysis Risk
SPC	suppression pool cooling
SPDS	safety parameter display system
SR	surveillance requirement
SRO	Senior Reactor Operator
SRP	Standard Review Plan
SRSS	square-root-of-the-sum-of-the squares
SRV	safety relief valve
SSCs	structures, systems, and components
SSE	safe shutdown earthquake
SSV	spring safety valve
SW	service water
SWS	service water system
T	thickness
TAVS	turbine area ventilation system
TEDE	total effective dose equivalent
TGSS	turbine gland sealing system
TI	Temporary Instruction
TIP	traversing incore probe
TS	Technical Specification
TSBS	turbine steam bypass system
TT	turbine trip

ACRONYM	DEFINITION
UAT	unit auxiliary transformer
UFSAR	Updated Final Safety Analysis Report
UHS	ultimate heat sink
UO2	uranium dioxide
USAS	United States of America Standards
USE	upper shelf energy
UT	ultrasonic testing
UTL	upper tolerance limit
VAR	volt amperes reactive
VF	void fraction
VHS	Vernon Hydroelectric Station
VYNPS	Vermont Yankee Nuclear Power Station
ZD	zero dependency