May 1, 2006

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
ENTERGY NUCLEAR VERMONT YANKEE, LLC and ENTERGY NUCLEAR OPERATIONS, INC.	)	Docket No. 50-271-OLA
	)	ASLBP No. 04-832-02-OLA
(Vermont Yankee Nuclear Power Station)	)	

NRC STAFF'S ANSWER TO NEW ENGLAND COALITION'S
REQUEST FOR LEAVE TO FILE NEW CONTENTIONS

ATTACHMENT 4

Entergy

Entergy Nuclear Vermont Yankee, LLC
Entergy Nuclear Operations, Inc.
185 Old Ferry Road
Brattleboro, VT 05302-0500

September 10, 2003
BVY 03-80

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: **Vermont Yankee Nuclear Power Station**
License No. DPR-28 (Docket No. 50-271)
Technical Specification Proposed Change No. 263
Extended Power Uprate

Pursuant to 10CFR50.90, Vermont Yankee[1] (VY) hereby proposes to amend its Facility Operating License, DPR-28, for the Vermont Yankee Nuclear Power Station (VYNPS). The proposed license amendment will increase the maximum authorized power level from 1593 megawatts thermal (MWt) to 1912 MWt. This request includes supporting Technical Specification (TS) changes necessary to implement the increased power level. The proposed extended power uprate (EPU) represents an increase of approximately 20% above original rated thermal power (RTP).

Attachment 1 to this letter contains a description and summary justification of each proposed change to the operating license and TS. Attachment 2 contains the determination of no significant hazards consideration associated with the license amendment request. VY has reviewed the proposed change to the current licensing basis in accordance with 10CFR50.92 and concludes that the proposed change does not involve a significant hazards consideration.

Attachment 3 provides a list of modifications and tests necessary to support EPU. These modifications will be implemented during the next two refueling outages (i.e., the refueling outages expected to begin in the Spring of 2004 (i.e., RFO-24) and Fall 2005 (i.e., RFO-25)). Modifications performed during RFO-24 will allow for an approximate 15% increase in RTP. Modifications completed subsequent to RFO-24 should allow the facility to achieve the full uprate to 1912 MWt. VY has evaluated the modifications currently planned to support EPU and determined that they do not constitute a material alteration to the plant, as discussed in *10CFR50.92*. These modifications constitute planned actions on the part of VY. Further evaluations may identify the need for additional modifications or obviate the need for some modifications as currently identified. As such, this is not a formal commitment to implement the modifications exactly as described or per the proposed schedule. Also, included as part of Attachment 3, is a Power Ascension Test Plan matrix that specifies expected EPU testing at different power levels and a Comparison of Initial Startup Testing and Planned EPU Testing.

This request for license amendment was prepared following the guidelines contained in the NRC-approved, NEDC-33004P-A[2]. Attachment 4 contains NEDC-33090P[3], which is the Power Uprate

[1] Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. are the licensees of the Vermont Yankee Nuclear Power Station.

[2] GE Nuclear Energy, "Constant Pressure Power Uprate," Licensing Topical Report NEDC-33004P-A (Proprietary), July 2003, and NEDO-33004-A (Non-Proprietary), July 2003.

[3] GE Nuclear Energy, "Safety Analysis Report for Vermont Yankee Nuclear Power Station Constant Pressure Power Uprate," NEDC-33090P, September 2003.

A001

ADAMS Accession
No. ML0325800890 (partial)

Safety Analysis Report (PUSAR) for VYNPS. The PUSAR is a summary of the results of the safety analyses and evaluations performed specifically for the VYNPS EPU. The PUSAR contains information which General Electric Company (GE) considers proprietary. GE requests that the proprietary information in this report be withheld from public disclosure in accordance with 10CFR9.17(a)(4) and 10CFR2.790(a)(4). An affidavit supporting this request is provided in Attachment 5. The NRC may duplicate this submittal, including the PUSAR, for the purpose of internal review. A non-proprietary version of NEDC-33090 is included as Attachment 6.

As part of the power ascension test plan, VY is not planning to conduct large transient testing which requires an automatic scram from high power (e.g., main steam isolation valve (MSIV) closure). Attachment 7 provides justification for not performing this testing.

This request for license amendment, while not being submitted as a risk informed licensing action, as defined by Regulatory Guide 1.174[4], was evaluated from a risk perspective. As demonstrated in Section 10.5 of the PUSAR, when the guidelines established in Regulatory Guide 1.174 are applied, the calculated results from the Level 1 and Level 2 Probabilistic Safety Analyses represent a very small risk increase in core damage frequency (CDF) and large early release frequency (LERF). The best estimate of the risk increase for at-power internal events due to the EPU is a delta CDF of 3.3 E-7/year (i.e., an increase of 4.2% over the base CDF of 7.77 E-6/year. The best estimate for at-power internal events results in a delta LERF of 1.1 E-7/year (i.e., an increase of 4.9% over the base LERF of 2.23 E-6/year. VY considers these revised estimates continue to present an acceptable level of risk.

Transition to the GE14 fuel design is necessary to achieve the full EPU. VYNPS' current reactor core is partially GE-14 fuel. Complete transition should occur over the next two refueling cycles.

VY has performed an assessment of environmental impacts of the proposed EPU from 1593 MWt to 1912 MWt. This assessment was performed by comparing the environmental impacts of the EPU to those previously identified by the U.S. Atomic Energy Commission in the 1972 Final Environmental Statement[5] (FES) for continued construction and proposed issuance of an operating license for VYNPS. The comparisons show that the conclusions of the FES and Environmental Assessment remain valid for operation at 1912 MWt. Attachment 8 contains the VYNPS Environmental Assessment Report for EPU. The intent of the assessment is to provide sufficient information for the NRC to evaluate the environmental impact of the power uprate in accordance with the requirements of 10CFR51.

As part of the proposed license amendment, VY is proposing a change to the licensing bases with regard to the crediting of containment overpressure for calculating certain pump net-positive suction head (NPSH) following a loss-of-coolant accident (LOCA), station blackout and Appendix R fire events. VYNPS currently complies with the provisions of Safety Guide 1[6] (i.e., Regulatory Guide 1.1). As a result of the proposed EPU, VY is revising these design bases, recognizing the contribution to NPSH provided by increased containment pressure following the postulated events. Credit for containment overpressure will be taken to assure that adequate NPSH is available for low pressure emergency core cooling system (ECCS) pumps. This change is consistent with actions taken by other utilities who have sought EPUs. PUSAR Section 4.2.6 provides the justification for the proposed change in licensing bases.

[4] U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.

[5] U.S. Atomic Energy Commission, "Final Environmental Statement Related to the Operation of Vermont Yankee Nuclear Power Station," Docket No. 50-271, July 1972.

[6] U.S. Atomic Energy Commission, Safety Guide 1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," November 2, 1970.

In support of the VYNPS EPU, VY has previously requested two other license amendments. Acceptance of the other two license amendment requests is necessary to achieve full EPU. The first of the submittals[7] (i.e., ARTS/MELLLA) supports operation of VYNPS in a core flow region which is above the rated rod line. The implementation of the Average Power Range Monitor, Rod Block Monitor Technical Specification (ARTS) will increase plant operating efficiency by updating the thermal limits requirements and improving plant instrumentation responses and accuracy. The changes requested by VY letter BVY 03-23 are based on current RTP (i.e., 1593 MWt), but are updated herein to reflect EPU conditions. In addition, certain RTP based parameters are affected by EPU; changes to these parameters are discussed in the PUSAR.

The second submittal[8] (i.e., Alternative Source Term) requests, in accordance with 10CFR50.67, a full scope application of an Alternative Source Term (AST) for VYNPS. The VYNPS EPU was analyzed, and the VYNPS PUSAR was prepared, based on AST methodology.

PUSAR Section 3.2.1, "Fracture Toughness," summarizes evaluations supporting the TS pressure versus temperature (P/T) limitations for the reactor coolant system under EPU conditions. The NRC staff is currently reviewing a proposed change[9] to VYNPS TS 3.6.A, "Pressure and Temperature Limitations," which revises the existing normal operating and hydrostatic and leak testing P/T curves. The analysis used to support Proposed Change No. 258 is updated in terms of power output to account for EPU. The P/T limits did not change as a consequence.

VY is actively assessing emergent BWR steam dryer issues together with industry and NRC staff as they may relate to EPU. As discussed in PUSAR Section 3.4.2 VY has performed a qualitative evaluation of its steam dryer and has identified certain modifications and inspections to ensure dryer structural integrity at EPU conditions. VY also expects to appropriately implement recommendations in a revision to a GE Service Information Letter addressing steam dryer issues.

An assessment of the effects of EPU on plant and transmission grid stability (as described in PUSAR section 6.1.1) is underway and will be provided to the NRC staff as part of its review in this matter. Upon completion of third-party reviews, VY expects to submit the results of the study to NRC in October 2003.

Attachments 9 and 10, respectively, provide marked-up pages and re-typed pages to the operating license and TSs, incorporating the revisions resulting from EPU. Some of the marked-up and re-typed TS pages in Attachments 9 and 10 are based on the aforementioned outstanding requests for license amendment, and therefore assume their prior acceptance. The subject marked-up pages are clearly identified in this regard.

As previously stated, modifications performed during the next refueling outage will allow for a power uprate of approximately 15%. As such, VYNPS will be in a position to implement the first uprate at the completion of the Spring 2004 refueling outage. VY requests that the EPU license amendment be issued with an effective date of July 2004. Timely issuance of the proposed EPU license amendment could facilitate initial uprate implementation to provide for summer 2004 peak electrical load demands. To support implementation of the Technical Specification changes, VY requests an implementation period of 120 days after the license amendment becoming effective.

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[7] Vermont Yankee letter to U.S. Nuclear Regulatory Commission, "Implementation of ARTS/MELLLA at Vermont Yankee," Proposed Change No. 257, BVY 03-23, March 20, 2003.

[8] Vermont Yankee letter to U.S. Nuclear Regulatory Commission, "Alternative Source Term," Proposed Change No. 260, BVY 03-70, July 31, 2003.

[9] Vermont Yankee letter to U.S. Nuclear Regulatory Commission, "RPV Fracture Toughness and Material Surveillance Requirements," Proposed Change No. 258, BVY 03-29, March 26, 2003.

If you have any questions, please contact Mr. Jim Devincentis at (802) 258-4236.

Sincerely,

Jay K. Thayer
Site Vice President

STATE OF VERMONT)
)ss
WINDHAM COUNTY)

Then personally appeared before me, Jay K. Thayer, who, being duly sworn, did state that he is Site Vice President of the Vermont Yankee Nuclear Power Station, that he is duly authorized to execute and file the foregoing document, and that the statements therein are true to the best of his knowledge and belief.

Sally A. Sandstrum, Notary Public
My Commission Expires February 10, 2007

SALLY A. SANDSTRUM
NOTARY
PUBLIC
WINDHAM COUNTY, VERMONT

Attachments:

1. Proposed Changes and Summary Justifications
2. Determination of No Significant Hazards Consideration
3. Modifications and Tests
4. NEDC-33090P (Proprietary Information)
5. Affidavit for Withholding NEDC-33090P from Public Disclosure
6. Non-Proprietary Version of NEDC-33090
7. Justification for Exception to Large Transient Testing
8. VYNPS Environmental Assessment Report for EPU
9. Marked-up Technical Specification Pages
10. Re-typed Technical Specification Pages

cc: (with attachments, except as noted)

USNRC Region 1 Administrator
USNRC Resident Inspector - VYNPS
USNRC Project Manager – VYNPS (two copies)
Vermont Department of Public Service (w/o proprietary information)

Docket No. 50-271
BVY 03-80

Attachment 6

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263

Extended Power Uprate

Non-Proprietary Version of NEDC-33090

GE Nuclear Energy

175 Curtner Ave., San Jose, CA 95125

NEDO-33090
Revision 0
Class III
0000-0007-5271
September 2003

SAFETY ANALYSIS REPORT

FOR

VERMONT YANKEE NUCLEAR POWER STATION

CONSTANT PRESSURE POWER UPRATE

Prepared by: E. D. Schrull

Approved by: Michael J. Dick, Project Manager
General Electric Company

Approved by: Craig J. Nichols, Project Manager
Entergy Nuclear Operations, Inc.

ADAMS Accession
No. ML032580103
(Partial)

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

The only undertakings of the General Electric Company (GE) respecting information in this document are contained in the contract between Entergy Nuclear Operations, Inc. (ENOI) and GE, Contract Order No. VY015144, effective November 13, 2002, and nothing contained in this document shall be construed as changing the contract. The use of this information by anyone other than ENOI, or for any purpose other than that for which it is intended, is not authorized; and, with respect to any unauthorized use, GE makes no representation or warranty, express or implied, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document, or that its use may not infringe privately owned rights.

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4.4 MAIN CONTROL ROOM ATMOSPHERE CONTROL SYSTEM

The VYNPS topic addressed in this evaluation is:

Topic	CPPU Disposition	VYNPS Result
Iodine intake	[[	]]

The Control Room Heating, Ventilation, and Air Conditioning (HVAC) system is designed to provide appropriate temperature conditions for personnel and equipment in the Control Room during any mode of operation or the most adverse emergency condition. CPPU adds negligible heat loading to this area, as the majority of the heat loads are from components and personnel already in the Control Room prior to CPPU.

The Control Room HVAC system is also designed to allow manual isolation of the Control Room, from within the Control Room, by placing the HVAC system into the recirculation mode.

VYNPS has separately submitted an LAR (Reference 27) describing the full implementation of the Alternative Source Term (AST) methodology that complies fully with RG 1.183. The AST evaluation calculated the Control Room dose for all DBAs with the assumption that the post-isolation air inleakage is equal to the pre-isolation value. The evaluation also considered new source-to-receptor pathways to comprehensively evaluate the Control Room dose. The conservative results demonstrate that the CPPU dose to the Control Room occupants will be less than the 30-day 5-rem Total Effective Dose Equivalent (TEDE) dose limit for the limiting DBA LOCA. Table 4-4 summarizes the Control Room doses from the AST analyses.

4.5 STANDBY GAS TREATMENT SYSTEM

The Standby Gas Treatment System (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 particulates and halogens, the SGTS limits off-site dose following a postulated DBA. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
Flow capacity	[[	
Iodine removal capability		]]

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. [[

Table 4-4
VYNPS Control Room Dose from DBAs

DBA	Dose (rem TEDE)
LOCA (30 day)	3.4
MSLB [1.] (2 hour)	2.0
Control Rod Drop Accident (CRDA) (30 day)	0.4
Fuel Handling Accident (FHA) (2 hour)	0.153

Note:

1. MSLB results based on 4.0 μCi/gm I-131 Dose Equivalent.

The primary function of the Gaseous Waste Management (Offgas) System is to process and control the release of gaseous radioactive effluents to the site environs so that the total radiation exposure of persons in offsite areas is within the guideline values of 10 CFR 50, Appendix I.

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. [[

]] Thus, the recombiner and condenser, as well as downstream system components, are designed to handle an average increase in thermal power of as much as 70% relative to CLTP, without exceeding the design basis temperatures, flow rates, or heat loads. Therefore, the gaseous radwaste system at VYNPS is confirmed to be consistent with GE design specifications for radiolytic flow rate [[
]]

8.3 RADIATION SOURCES IN THE REACTOR CORE

During power operation, the radiation sources in the core are directly related to the fission rate. These sources include radiation from the fission process, accumulated fission products and neutron reactions as a secondary result of fission. Historically, these sources have been defined in terms of energy or activity released per unit of reactor power. Therefore, for a CPPU, the percent increase in the operating source terms is no greater than the percent increase in power. The topic addressed in this evaluation is:

Topic	CPPU Disposition	VYNPS Result
Post operational radiation sources for radiological and shielding analysis	[[	]]

The post-operation radiation sources in the core are primarily the result of accumulated fission products. Two separate forms of post-operation source data are normally applied. The first of these is the core gamma-ray source, which is used in shielding calculations for the core and for individual fuel bundles. This source term is defined in terms of MeV/sec per Watt of reactor thermal power (or equivalent) at various times after shutdown. The total gamma energy source, therefore, increases in proportion to reactor power.

The second set of post-operation source data consists primarily of nuclide activity inventories for fission products in the fuel. These data are needed for post-accident and SFP evaluations, which are performed in compliance with regulatory guidance that applies different release and transport assumptions to different fission products. The core fission product inventories for these evaluations are based on an assumed fuel irradiation time, which develops "equilibrium" activities in the fuel (typically 3 years). Most radiologically significant fission products reach equilibrium within a 60-day period. [[

]] The radionuclide inventories are provided in terms of Curies per megawatt of reactor thermal power at various times after shutdown.

The VYNPS specific parameters are enveloped by the bounding parameters of the radiation sources in the reactor core generic description provided in the CLTR. The results of the VYNPS plant-specific radiation sources evaluation are included in the LOCA, FHA, and CRDA radiological analyses presented in Section 9.2. A plant-specific analysis for NUREG-0737, Item II.B.2, post-accident mission doses was performed in which the evaluated mission doses for VYNPS are demonstrated to be less than 5 rem TEDE. Details of the analysis are contained in the AST submittal (Reference 27), which describes the full implementation of the AST methodology at CPPU conditions.

8.4 RADIATION SOURCES IN REACTOR COOLANT

Radiation sources in the reactor coolant at VYNPS include activation products and activated corrosion and fission products. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
8.4.1 Coolant Activation Products	[[	
8.4.2 Activated Corrosion and Fission Products		]]

8.4.1 Coolant Activation Products

During reactor operation, the coolant passing through the core region becomes radioactive as a result of nuclear reactions. The coolant activation, especially N-16 activity, is the dominant source in the turbine building and in the lower regions of the drywell. The activation of the water in the core region is in approximate proportion to the increase in thermal power. [[

]]

8.4.2 Activated Corrosion Products and Fission Products

The reactor coolant contains activated corrosion products, which are the result of metallic materials entering the water and being activated in the reactor region. Under the CPPU conditions, the feedwater flow increases with power and the activation rate in the reactor region increases with power. The net result is an increase in the activated corrosion product production.

Fission products in the reactor coolant are separable into the products in the steam and the products in the reactor water. The activity in the steam consists of noble gases released from the core plus carryover activity from the reactor water. This activity is the noble gas offgas that is included in the plant design. The calculated offgas rates for CPPU after thirty minutes decay are well below the original design basis of 0.03 curies/sec. Therefore, no change is required in the design basis for offgas activity for the CPPU.

The fission product activity in the reactor water, like the activity in the steam, is the result of minute releases from the fuel rods. Fission product activity levels in the reactor water at design carry over rates were calculated to be less than the design basis, therefore requiring no change.

8.5 RADIATION LEVELS

For CPPU at VYNPS, normal operation radiation levels increase by approximately the percentage increase in power level. Some areas reflect an additional small increase due to accelerated steam flow. For conservatism, many aspects of the plant were originally designed for higher-than-expected radiation sources. Thus, the increase in radiation levels does not affect radiation zoning or shielding in the various areas of the plant because it is offset by conservatism in the original design, source terms used, and analytical techniques. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
Normal operational radiation levels	[[	
Post-operation radiation levels		
Post-accident radiation levels		]]

The normal operating radiation levels specified for VYNPS are generally based on dose rate measurements at various locations during plant operation at CLTP conditions. The normal operating radiation levels specified for CLTP conditions were evaluated to increase in proportion to the increase in thermal power. The increased normal radiation levels were evaluated and determined to have no adverse effect on safety-related plant equipment as indicated in Sections 10.3.1 and 10.3.2. Individual worker exposures can be maintained within acceptable limits by controlling access to radiation areas in conjunction with procedural controls and the site ALARA (As Low as Reasonably Achievable) program. In addition, VYNPS has previously implemented noble metal chemical addition to limit the increase in normal radiation doses from the implementation of hydrogen water chemistry.

[[

]] Regardless, individual worker exposures can be maintained within acceptable limits by controlling access to radiation areas using the site ALARA program. Procedural controls compensate for increased radiation levels. Radiation measurements will be made at selected power levels to ensure the protection of personnel.

Post-accident radiation levels were evaluated for radiological consequences using the RG 1.183 AST methodology, as part of the VYNPS plant-specific accident analyses presented in Section 9.2. Accident radiation levels at CLTP were evaluated using the TID source term methodology. Post-accident radiation levels remain below established regulatory limits for CPPU conditions. Details of the accident radiological analysis are contained in a separate VYNPS LAR (Reference 27) describing full implementation of the AST methodology at CPPU conditions. The increased post-accident radiation doses have no adverse effect on safety-related plant equipment as indicated in Sections 10.3.1 and 10.3.2. A plant-specific analysis for NUREG-0737, Item II.B.2, post-accident mission doses has been performed, the details of which are provided in the AST LAR (Reference 27).

Section 9.2 addresses the accident doses for the Control Room.

8.6 NORMAL OPERATION OFF-SITE DOSES

The primary sources of normal operation offsite doses at VYNPS are airborne releases from the Offgas System and gamma shine from the plant turbines. The topics addressed are:

Topic	CPPU Disposition	VYNPS Result
Plant gaseous emissions	[[	
Plant skyshine from the turbine		]]

The increase in normal operation activity levels in the reactor coolant is proportional to the percentage increase in core thermal power, i.e., 20%. Noble gas levels in the steam phase are expected to be approximately the same as pre-CPPU conditions because the increase in steaming rate is approximately the same as the production rate due to CPPU. Noble gas release through the off-gas system and release of tritium is conservatively estimated to increase proportionally to the CPPU. Steam activity levels for species related to carryover (halogens & particulates) and volatile halogens will increase proportionally to changes in reactor coolant and the moisture carryover fraction. Examination of the normal operation radiological effluent doses reported for the last five years (1997-2001) indicates that the estimated doses due to the pre-CPPU gaseous releases (~1 mrem) are a very small fraction of the 10CFR 50 Appendix I guidelines; and that there were no radiological liquid effluents discharged during this time period. While the normal operation releases and doses are expected to increase due to CPPU, the dose effect remains well within the limits of 10 CFR 20, 10 CFR 50, Appendix I, and 40 CFR 190.

there is no increase in highest flow control line for the VYNPS CPPU. [[

]]

9.2 DESIGN BASIS ACCIDENTS

This section addresses the radiological consequences of DBAs for VYNPS. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
Main Steam Line Break Outside Containment	[[	
Instrument Line Break		
LOCA Inside Containment		
Fuel Handling Accident		
Control Rod Drop Accident		]]

The magnitude of radiological consequences of a DBA is proportional to the quantity of radioactivity released to the environment. This quantity is a function of the fission products released from the core as well as the transport mechanism between the core and the release point.

VYNPS has submitted an LAR (Reference 27) describing full implementation of the AST methodology, at CPPU conditions, that complies with Regulatory Guide 1.183. This methodology has been used in the evaluation of DBA radiological consequences.

The Main Steam Line Break Accident (MSLBA) analysis for VYNPS is based on hot standby conditions and [[
]] Therefore, the resulting radiological consequences remain within applicable regulatory criteria for the MSLBA at CPPU conditions.

The Instrument Line Break (ILB) is not considered a DBA for VYNPS.

For the LOCA inside Containment, FHA, and CRDA, the whole body and thyroid doses were calculated at the exclusion area boundary, Low Population Zone (LPZ), and in the Control Room. The doses resulting from the accidents analyzed are compared with the applicable dose limits in Tables 9-1 through 9-3, for both the CPPU and pre-CPPU RTP levels. The effect of extended burnup on the FHA was not evaluated, per RG 1.183, based on CPPU core average bundle power of 5.3 MWt and peak exposure of 58 GWD/MT. The [[ ]] results for the CPPU remain below established regulatory limits.

9.3 SPECIAL EVENTS

This section considers two special events: ATWS and SBO. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
9.3.1 ATWS (Overpressure) - Event Selection	[[	
9.3.1 ATWS (Overpressure) - Limiting Events		
9.3.1 ATWS (Suppression Pool Temperature) - Event Selection		
9.3.1 ATWS (Suppression Pool Temperature) - Limiting Events		
9.3.1 ATWS (Peak Cladding Temperature)		
9.3.2 Station Blackout		
9.3.3 ATWS with Core Instability		]]

9.3.1 Anticipated Transient Without Scram

The overpressure evaluation includes consideration of the most limiting RPV overpressure case. [[

]]

For VYNPS, the LOOP does not result in a reduction in the RHR pool cooling capability relative to the MSIVC and PRFO cases. With the same RHR pool cooling capability, the containment response for the MSIVC and PRFO cases bound the LOOP case. [[

]]

VYNPS meets the ATWS mitigation requirements defined in 10 CFR 50.62:

1. Installation of an Alternate Rod Insertion (ARI) system;
2. Boron injection equivalent to 86 gpm; and
3. Installation of automatic Recirculation Pump Trip (RPT) logic (i.e., ATWS-RPT).

In addition, plant-specific ATWS analyses were performed to ensure that the following ATWS acceptance criteria are met:

1. Peak vessel bottom pressure less than ASME Service Level C limit of 1500 psig;

2. Peak suppression pool temperature less than 281°F (Wetwell shell design temperature); and
3. Peak containment pressure less than 62 psig (110% of drywell design pressure).

The limiting events for the acceptance criteria discussed above are the PRFO event and the MSIVC event.

The ATWS analyses were performed for CLTP and for CPPU RTP to demonstrate the effect of the CPPU on the ATWS acceptance criteria. There is no change to the required hot shutdown boron weight for the CPPU ATWS analysis. The key inputs to the ATWS analysis are provided in Table 9-4. The results of the analysis are provided in Table 9-5.

The results of the ATWS analyses meet the above ATWS acceptance criteria. Therefore, the VYNPS response to an ATWS event at CPPU is acceptable.

Coolable core geometry is assured by meeting the 2200°F peak cladding temperature and the 17% local cladding oxidation acceptance criteria of 10 CFR 50.46. [[

]]

9.3.2 Station Blackout

SBO was reevaluated using the guidelines of NUMARC 87-00. The plant response to and coping capabilities for an SBO event are affected slightly by operation at CPPU RTP, due to the increase in the initial power level and decay heat. Decay heat was conservatively evaluated assuming end-of-cycle (18-month) and GE14 fuel. There are no changes to the systems and equipment used to respond to an SBO, nor is the required coping time changed.

Areas containing equipment necessary to cope with an SBO event were evaluated for the effect of loss-of-ventilation due to an SBO. The evaluation shows that equipment operability is bounded due to conservatism in the existing design and qualification bases. The battery capacity remains adequate to support HPCI/RCIC operation after CPPU. Adequate compressed gas capacity exists to support the SRV actuations.

The current CST inventory reserve and restoration of Alternate AC within 10 minutes ensures that adequate water volume is available to remove decay heat, depressurize the reactor, and maintain reactor vessel level above the top of active fuel. Consistent with the DBA-LOCA condition, the required NPSH margin for the RHR pumps has been evaluated (see Section 4.2.6) and a component acceptability review has been completed (see Section 3.10).

Based on the above evaluations, VYNPS continues to meet the requirements of 10 CFR 50.63 after the CPPU.

9.3.3 ATWS with Core Instability

The ATWS with core instability event occurs at natural circulation following a recirculation pump trip. Therefore, it is initiated at approximately the same power level as a result of CPPU operation because the MELLLA upper boundary is not increased. The core design necessary to achieve CPPU operations may affect the susceptibility to coupled thermal-hydraulic/neutronic core oscillations at the natural circulation condition, but would not significantly affect the event progression.

Several factors affect the response of an ATWS instability event, including operating power and flow conditions and core design. The limiting ATWS core instability evaluation presented in References 28 and 29 was performed for an assumed plant initially operating at CLTP and the MELLLA minimum flow point. [[

]] CPPU allows plants to increase their operating thermal power but does not allow an increase in control rod line. [[

]] The conclusion of Reference 29 and the associated NRC SER that the analyzed operator actions effectively mitigate an ATWS instability event are applicable to the operating conditions expected for CPPU at VYNPS.

[[

]]

Table 9-1
LOCA Radiological Consequences

Location	Current	Limit [1]	CPPU	Limit [2]
Exclusion Area				
Whole Body Dose	0.043	≤ 25	N/A	N/A
Thyroid Dose	94	≤ 300	N/A	N/A
TEDE Dose	N/A	N/A	3.14	≤ 25
Low Population Zone				
Whole Body Dose	0.28	≤ 25	N/A	N/A
Thyroid Dose	8.4	≤ 300	N/A	N/A
TEDE Dose	N/A	N/A	0.52	≤ 25
Control Room				
Whole Body Dose	0.003	≤ 5	N/A	N/A
Thyroid Dose	20.2	≤ 30	N/A	N/A
Beta Dose	N/A	N/A	N/A	N/A
TEDE Dose	N/A	N/A	3.40	≤ 5

Notes:

1. 10 CFR 100 limit (rem)
2. 10 CFR 50.67 limit (rem TEDE)

Table 9-2
FHA Radiological Consequences

Location	Current	Limit [1.]	CPPU	Limit [2.]
Exclusion Area				
Whole Body Dose	0.027	≤ 25	N/A	N/A
Thyroid Dose	32	≤ 300	N/A	N/A
TEDE Dose	N/A	N/A	0.472	≤ 6.30
Low Population Zone				
Whole Body Dose	0.00084	≤ 25	N/A	N/A
Thyroid Dose	3.4	≤ 300	N/A	N/A
TEDE Dose	N/A	N/A	< 0.472	≤ 6.30
Control Room				
Whole Body Dose	N/A	N/A	N/A	N/A
TEDE Dose	N/A	N/A	0.153	≤ 5

Notes:

1. 10 CFR 100 limit (rem)
2. 10 CFR 50.67 limit (rem TEDE)

Table 9-3
CRDA Radiological Consequences

Location	Current	Limit [1.]	CPPU	Limit [2.]
Exclusion Area				
Whole Body Dose	0.015	≤ 25	N/A	N/A
Thyroid Dose	3.0	≤ 300	N/A	N/A
Beta Dose	0.023	≤ 300	N/A	N/A
TEDE Dose	N/A	N/A	0.38	≤ 6.30
Low Population Zone				
Whole Body Dose	0.0074	≤ 25	N/A	N/A
Thyroid Dose	1.8	≤ 300	N/A	N/A
Beta Dose	0.012	≤ 300	N/A	N/A
TEDE Dose	N/A	N/A	0.081	≤ 6.30
Control Room				
Whole Body Dose	0.0097	≤ 5	N/A	N/A
Thyroid Dose	28	≤ 30	N/A	N/A
Beta Dose	0.37	≤ 30	N/A	N/A
TEDE Dose	N/A	N/A	0.40	≤ 5

Notes:

1. 10 CFR 100 limit (rem)
2. 10 CFR 50.67 limit (rem TEDE)

10.3 ENVIRONMENTAL QUALIFICATION

Safety-related components are required to be qualified for the environment in which they are required to operate. The topics addressed in this evaluation are:

Topic	CPPU Disposition	VYNPS Result
10.3.1 Electrical Equipment	[[	
10.3.2 Mechanical Equipment With Non-Metallic Components		
10.3.3 Mechanical Component Design Qualification		]]

10.3.1 Electrical Equipment

The safety-related electrical equipment was reviewed to assure the existing qualification for the normal and accident conditions expected in the area where the devices are located remain adequate. Table 10-2 provides a listing of the EQ effects and parameter changes associated with CPPU.

Inside Containment

EQ for safety-related electrical equipment located inside the containment is based on MSLB and/or DBA/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, but remain bounded by the normal temperatures used in the EQ analyses. The CPPU accident conditions are compared to CLTP accident conditions in Section 4. The accident conditions for temperature and pressure, used in the current EQ analyses, bound the CPPU conditions described in Section 4.

The current radiation levels under normal plant conditions were evaluated to increase in proportion to the increase in RTP for the eight years remaining of the operating license, resulting in 4% increase over the 40 years. The accident radiation levels increase by $\leq$ 17% above the levels used in the current EQ Program. The total integrated doses (normal plus accident) for CPPU conditions were determined to challenge the qualification of some equipment located inside containment. Equipment that required further evaluation included certain cable types, splices, and electrical penetrations. A qualitative evaluation, using equipment-specific radiation dose assessment, indicates that with additional analysis, the equipment should be acceptable for the CPPU conditions. The EQ documentation and radiation analyses will be revised to demonstrate qualification to CPPU conditions.

Outside Containment

Accident temperature, pressure, and humidity environments used for qualification of equipment outside containment result from an MSLB, or other HELB, whichever is limiting for each plant

area, considering the safety function for the HELB. The peak HELB temperatures at CPPU RTP, in some cases, exceed the values used for equipment qualification at CLTP conditions. The temperature peaks that are not bounded by the CLTP conditions were evaluated. The qualification of several power supplies and distribution panels was exceeded. Such components will be requalified to the CPPU conditions by crediting new qualification tests, analysis, or relocation of the equipment to a milder environment location. The accident temperature resulting from a LOCA/MSLB inside containment increased some reactor building areas due to the 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.

The normal temperature, pressure, and humidity conditions in the reactor building do not change as a result of CPPU, except that the normal steam tunnel temperature is expected to increase slightly due to increased FW temperature. The current radiation levels under normal plant conditions were conservatively evaluated to increase in proportion to the increase in RTP for the remaining eight years of operation. The outside containment accident radiation levels increase by ≤ 17% above the levels used in the current EQ Program. The total integrated doses (normal plus accident) for CPPU conditions were evaluated. There were several types of equipment located outside of containment that were adversely affected by the radiation dose increase. A qualitative evaluation, using equipment specific radiation dose assessment indicates that with additional analysis, the equipment should be acceptable for the CPPU conditions. These components will require additional evaluation prior to CPPU implementation. The EQ documentation and radiation analyses will be revised to demonstrate qualification to CPPU conditions.

10.3.2 Mechanical Equipment With Non-Metallic Components

The temperatures, accident radiation level, and the normal radiation level increase slightly due to CPPU as discussed in Section 10.3.1. Although the VYNPS EQ Program does not specifically address mechanical equipment with non-metallic components, the VYNPS design control program ensures that non-metallic components (e.g., seals, gaskets, lubricants, diaphragms) are properly specified and procured for the environment in which they are intended to function.

10.3.3 Mechanical Component Design Qualification

The mechanical design of equipment/components (e.g., pumps, heat exchangers) in certain systems is affected by operation at CPPU due to increased temperatures and flows. The design qualification of mechanical components was evaluated relative to the revised operating conditions and determined to be adequate for CPPU operation.

The effects of increased fluid induced loads on safety-related components are described in Sections 3 and 4.1. Increased nozzle loads and component support loads due to the revised operating conditions were evaluated within the piping assessments in Section 3. These increased loads are insignificant, and become negligible (i.e., remain bounded) when combined with the