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JUN 0 7 2004

LR-N04-0062
LCR H04-01

U.S. Nuclear Regulatory Commission
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Washington, DC 20555

REQUEST FOR LICENSE AMENDMENT
ARTS/MELLLA IMPLEMENTATION
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

Pursuant to 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby requests a revision to the Technical Specifications (TS) for the Hope Creek Generating Station. In accordance with 10 CFR 50.91(b)(1), a copy of this submittal has been sent to the State of New Jersey.

The proposed changes reflect an expanded operating domain resulting from implementation of Average Power Range Monitor/Rod Block Monitor/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA). The average power range monitor (APRM) flow-biased flux scram setpoint and the APRM and rod block monitor (RBM) flow-biased rod block trip setpoints would be revised to permit operation in the MELLLA region. In addition, the APRM scram and rod block trip setdown requirement would be replaced by more direct power- and flow-dependent thermal limits to reduce the need for APRM gain adjustments and to allow more direct thermal limits administration during operation at other than rated conditions.

PSEG also proposes to make changes in the methods used to evaluate annulus pressurization (AP) and jet loads resulting from the postulated Recirculation Suction Line Break (RSLB).

Operation in the MELLLA region will provide improved power ascension capability by extending plant operation at rated power with less than rated core flow. Operation in the MELLLA region can result in the need for fewer control rod manipulations to maintain rated power during the fuel cycle. Replacement of the APRM scram and rod block trip setdown requirement will improve reliability and provide more direct protection

APOI

This letter forwards proprietary information in accordance with 10CFR 2.390. The balance of this letter may be considered non-proprietary upon removal of Attachment 4.

of plant safety. The NRC has approved similar changes for other licensees as described in Attachment 1.

PSEG has evaluated the proposed changes in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c), and has determined this request involves no significant hazards considerations. An evaluation of the requested changes is provided in Attachment 1 to this letter. The marked up Technical Specification and Updated Final Safety Analysis Report (UFSAR) pages affected by the proposed changes are provided in Attachments 2 and 3. Attachment 4 contains safety analyses performed in support of the proposed changes.

The fuel dependent portions of the safety analyses described in Attachments 1 and 4 are based on the assumption of a representative core with GE14 fuel and in some cases existing analyses for plants similar to Hope Creek. For the fuel dependent portions of the safety analyses, PSEG will perform plant and fuel specific analyses to justify operation in the ARTS/MELLLA condition using NRC approved methodologies. The non-fuel dependent evaluations described in Attachments 1 and 4 are based on the Hope Creek plant configuration and are being submitted for review in advance of the fuel dependent analyses to allow additional time for the NRC staff to review aspects of PSEG's request that are not dependent on fuel.

Attachment 4 contains information which General Electric Company (GE) considers to be proprietary. GE requests that the proprietary information in Attachment 4 be withheld from public disclosure in accordance with 10 CFR 2.390. An affidavit in support of this request is provided in Attachment 5.

Attachment 6 provides a summary of the regulatory commitments made in this submittal. PSEG will submit the Hope Creek fuel dependent analyses results and, if required, any additional Technical Specification changes associated with the fuel dependent evaluations, by 09/30/04. PSEG requests approval of the proposed License Amendment within six months after submittal of the fuel dependent evaluations, with 120 days for implementation.

Should you have any questions regarding this request, please contact Mr. Paul Duke at 856-339-1466.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 6/7/2004
(date)

Michael H. Brothers
Vice President - Site Operations

Attachments (6)

This letter forwards proprietary information in accordance with 10CFR 2.390. The balance of this letter may be considered non-proprietary upon removal of Attachment 4.

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HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

EVALUATION OF PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS FOR ARTS/MELLLA IMPLEMENTATION

REQUEST FOR LICENSE AMENDMENT
ARTS/MELLLA IMPLEMENTATION

Table of Contents

1. DESCRIPTION 1

2. PROPOSED CHANGE 1

3. BACKGROUND 3

4. TECHNICAL ANALYSIS 5

5. REGULATORY SAFETY ANALYSIS 12

    5.1 No Significant Hazards Consideration 12

    5.2 Applicable Regulatory Requirements/Criteria 16

6. ENVIRONMENTAL CONSIDERATION 17

7. REFERENCES 18

1. DESCRIPTION

The proposed amendment would revise the Hope Creek Technical Specifications (TSs) contained in Appendix A to the Operating License to reflect an expanded operating domain resulting from implementation of Average Power Range Monitor/Rod Block Monitor/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA). The average power range monitor (APRM) flow-biased flux scram setpoint and the APRM and rod block monitor (RBM) flow-biased rod block trip setpoints would be revised to permit operation in the MELLLA operating domain. The flow-biased APRM scram and rod block trip setpoint setdown requirements would be replaced by more direct power and flow dependent thermal limits to reduce the need for APRM setpoint or gain adjustments and to allow more direct thermal limits administration. In addition, the methods used to evaluate annulus pressurization (AP) and jet loads resulting from the postulated Recirculation Suction Line Break (RSLB) would be changed to permit more realistic but still conservative evaluation of the loads associated with the RSLB in the MELLLA condition.

2. PROPOSED CHANGE

The marked up pages for the proposed changes to the Technical Specifications are included in Attachment 2 of this submittal.

1. APRM and RBM trip setpoints would be revised as follows to permit operation in the MELLLA operating domain:

    a. The APRM flow-biased simulated thermal power – upscale trip setpoint in TS Table 2.2.1-1 would be changed to:

    $\leq 0.66\ (w - \Delta w) + 66\%$.

    The allowable value would be changed to:

    $\leq 0.66\ (w - \Delta w) + 69\%$.

    The APRM high flow clamped setpoints would not be changed.

    b. The APRM flow biased neutron flux-upscale rod block trip setpoint in TS Table 3.3.6-2 would be changed to:

    $\leq 0.66\ (w - \Delta w) + 57\%$

    The allowable value would be changed to:

    $\leq 0.66\ (w - \Delta w) + 60\%$

c. The RBM upscale flow biased trip setpoint in TS Table 3.3.6-2 would be changed to:

$$\leq 0.66 (w - \Delta w) + 65\%.$$

The allowable value would be changed to:

$$\leq 0.66 (w - \Delta w) + 68\%.$$

The RBM upscale high flow clamped setpoint in TS Table 3.3.6-2 would be changed to ≤ 116%. The allowable value would be changed to ≤ 119%.

2. TS 3/4.2.2, "APRM Setpoints," which includes requirements for flow-biased APRM scram and rod block trip setpoint setdown, and the associated TS Bases would be deleted. The following additional changes would be made to reflect the deletion of TS 3/4.2.2:

    a. References to TS 3/4.2.2 would be deleted from TS 3/4.4.1 Actions a.2 and a.3 and from footnotes to TS Tables 4.3.1.1-1 and 3.3.6-2.

    b. Reference to the Maximum Fraction of Limiting Power Density (MFPLD) would be deleted from the specified conditions in the Applicability for Limiting Condition for Operation (LCO) 3.3.7.7.

    c. Definitions for "Core Maximum Fraction of Limiting Power Density," "Fraction of Limiting Power Density" "Fraction of Rated Thermal Power," and "Maximum Fraction of Limiting Power Density" would be deleted from TS Section 1.0.

    d. References to APRM trip setpoint adjustments would be deleted from TS Bases 2.2.1.

    e. TS 3/4.2.1, "Average Planar Linear Heat Generation Rate," 3/4.2.3 "Minimum Critical Power Ratio" and "3/4.2.4 Linear Heat Generation Rate" and associated TS Bases will be revised as appropriate to include a description of power- and flow-dependent thermal limits. These TS changes will be supplied in a separate transmittal with the Hope Creek fuel dependent analyses.

    f. The TS Index would be revised.

3. Two changes will be made in the methods of evaluation for the postulated Recirculation Suction Line Break (RSLB) in the reactor pressure vessel (RPV) shield annulus region. For the RSLB at the MELLLA minimum

pump speed point with consideration of the reduced feedwater temperatures allowed by the current Operating License Condition C.11 (RFWT), the mass and energy release profile will be calculated using the LAMB code in lieu of the current methodology described in NEDO-24548 (Reference 1). Jet reaction plus jet impingement loads on the RPV and the biological shield wall will be defined as a function of time in lieu of the currently assumed 1.0 millisecond rise time to steady state loads.

PSEG concluded the changes described above meet the criterion stated in 10 CFR 50.59(c)(2)(viii) since they represent departures from methods of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses. The marked up pages for the proposed changes to the Updated Final Safety Analysis Report (UFSAR) are included in Attachment 3 of this submittal.

The proposed changes are consistent with the requirements of NUREG-1433, "Standard Technical Specifications - General Electric Plants, BWR/4," Revision 2 and with changes previously approved by the NRC for other licensees as described in Section 4 of this Attachment. Operation in the MELLLA region will provide improved power ascension capability by extending plant operation at rated power with less than rated core flow. Operation in MELLLA can also result in the need for fewer control rod manipulations to maintain rated power during the fuel cycle, thereby improving operational flexibility. Replacement of the APRM trip setdown requirement will improve reliability and provide more direct protection of plant safety. The proposed changes will reduce the need for APRM gain adjustments and allow more direct thermal limits administration.

3. BACKGROUND

Many factors restrict the flexibility of a Boiling Water Reactor (BWR) during power ascension from the low-power/low-core flow condition to the high-power/high-core flow condition. Some of the factors that limit plant flexibility in achieving rated power are:

1. the currently licensed allowable operating power/flow map;

2. the APRM flow biased simulated thermal power-upscale scram and flow-biased neutron flux-upscale control rod block setdown requirements; and

3. the RBM flow-referenced rod block trip.

Once rated power is achieved, periodic control rod and core flow adjustments must be made to compensate for reactivity changes due to xenon effects and fuel burnup.

Hope Creek is currently licensed to operate in the Extended Load Line Limit Analysis (ELLLA) region up to approximately the 108% rod line and Increased Core Flow (ICF) region up to 105% core flow which results in a core flow window of 87% to 105% at rated thermal power (References 2 and 3).

A further expansion of the operating domain (MELLLA) and implementation of ARTS would allow for more efficient and reliable power ascensions and would allow rated power to be maintained over a wider core flow range, thereby reducing the frequency of control rod manipulations that require power maneuvers to implement. Expansion of the operating domain beyond the current power-flow map requires changes to the APRM and RBM trip functions described below.

APRM and RBM Trip Setpoints

The APRM flow biased trip setpoint varies as a function of reactor recirculation loop flow but is clamped such that it is always less than the APRM fixed neutron flux-upscale setpoint.

The APRM flow biased neutron flux upscale rod block function is designed to avoid conditions that would require reactor protection system (RPS) action if allowed to proceed. The APRM rod block alarm setting is selected to initiate a rod block before the APRM high neutron flux scram setting is reached.

The rod block monitor (RBM) is designed to prohibit erroneous withdrawal of a control rod during operation at high power levels. This prevents local fuel damage during a single rod withdrawal error. Because local fuel damage poses no significant threat relative to radioactive release from the plant, the RBM is a power generation system and is not used for accident mitigation.

APRM Trip Setpoint Setdown Requirement

TS Limiting Condition for Operation (LCO) 3.2.2 currently requires the APRM flow biased scram and rod block trip setpoints to be reduced when the fraction of rated thermal power (FRTP) is less than the core maximum fraction of limiting power density (CMFLPD). The trip setdown requirement ensures that margins to the fuel cladding Safety Limit are preserved during operation at other than rated conditions. As an alternative to adjusting the APRM setpoints, the APRM gains may be adjusted such that the APRM readings are greater than or equal to 100% times CMFLPD. Hope Creek's normal operating practice is to adjust APRM gains when required to meet LCO 3.2.2. Each APRM channel is typically bypassed while the required gain adjustment is made.

The setdown requirement originated from the Hench-Levy Minimum Critical Heat Flux Ratio (MCHFR) thermal limit criterion. Improved methodologies have subsequently been developed to provide more effective alternatives to the setdown requirement.

RSLB AP and Jet Loads

The current RSLB blowdown mass and energy release profiles for AP loads were calculated based on the method described in NEDO-24548 (Reference 1) with 50% of the blowdown flow assumed to be released into the annulus and the remaining 50% vented to the drywell atmosphere. The Reference 1 methodology was conservative since it used a simple bounding approximation for a complex blowdown process. Subsequent to this analysis, before initial plant operation, a flow diverter was added to the shield wall penetration, thus reducing the portion of the blowdown flow entering the annulus to 25 percent. The flow diverter mitigated the annulus pressurization load effects. However, the presence of the diverter created a jet load on the diverter/shield wall, which did not exist in the original design basis. Based on the limited documentation available, it appears that the evaluation performed in support of the diverter installation was limited to a qualitative assessment of the effects on the original analysis; the combined effects of the new set of loads (based on the configuration with the flow diverter) were not quantified in detail.

In the current design basis, the AP and jet loads were generated for the thermal-hydraulic conditions consistent with the normal operation at the 100% power / 100% core flow point of the power/flow map. During plant operation at off-rated conditions, the mass and energy release from the RSLB can be higher due to the lower enthalpy in the downcomer. Mass and energy releases were evaluated over the range of power/flow conditions associated with the MELLLA boundary. The mass and energy release data at the MELLLA minimum pump speed point with RFWT was evaluated as the governing condition for determining the RSLB asymmetric loads.

For the MELLLA minimum pump speed with RFWT condition, the mass and energy release profile was calculated in accordance with the current methodology and the resulting loads exceed the current design basis calculation.

The original methodology for calculating the RSLB jet reaction and jet impingement loads ('jet loads') on the RPV and the biological shield wall is documented in HCGS UFSAR Appendix 3C. This original methodology conservatively assumed a 1.0 millisecond rise time to the steady state loads. However, using this methodology causes the net effect of the AP, jet reaction and jet impingement loads calculated for the MELLLA minimum pump speed with RFWT condition to exceed the current design basis.

4. TECHNICAL ANALYSIS

The proposed changes would reflect an expanded operating domain resulting from implementation of Maximum Extended Load Line Limit Analysis (MELLLA). In addition, the flow-biased APRM scram and rod block trip setpoint setdown requirements would be replaced by more direct power and flow dependent thermal limits to reduce the need for manual setpoint adjustments and to allow more direct thermal limits administration. The proposed changes would be implemented during Hope Creek Cycle 13, currently scheduled to begin in the fourth quarter 2004.

Safety analyses performed in support of the proposed changes are described in Attachment 4. These analyses include fuel performance event evaluations (Sections 3.0 and 4.0), an evaluation of vessel overpressure protection (Section 5.0), an evaluation of thermal-hydraulic stability (Section 6.0), an evaluation of the loss-of-coolant accident (Section 7.0), containment response evaluations (Section 8.0), reactor internals integrity evaluations (Section 9.0), an evaluation of an anticipated transient without scram (Section 10.0), high energy line break evaluations (Section 11.0), and an evaluation of steam dryer and separator performance (Section 12.0). The technical analysis discussion below summarizes or supplements the information in Attachment 4.0.

Attachment 4 Section 1.0, Introduction, and Section 2.0, Overall Analysis Approach, provide a description and background for the implementation of ARTS/MELLLA at the Hope Creek Generating Station (HCGS). The content of Sections 1.0 and 2.0, relative to fuel dependent evaluations, accurately describes the approach HCGS is taking to justify and implement the ARTS/MELLLA bases. However, the assumptions and conclusions described in Section 1.0 and 2.0 for fuel dependent evaluations are based upon a representative core of GE14 fuel, and in some cases on existing analyses for plants similar to HCGS. The assumptions and conclusions relative to fuel dependent evaluations will be validated or, if required, updated based upon HCGS fuel dependent analyses results that will be submitted in a separate transmittal.

The content of Attachment 4 Sections 1.0 and 2.0, relative to non-fuel dependent evaluations, accurately describes the approach HCGS is taking to justify and implement the ARTS/MELLLA bases and correctly reflect the HCGS plant configuration. In addition, the assumptions and conclusions described in Sections 1.0 and 2.0 relative to non-fuel dependent evaluations are correct and applicable for HCGS. Therefore, the non-fuel related evaluations are considered complete.

Attachment 4 Sections 3.0, Fuel Thermal Limits, 4.0, Rod Withdrawal Error, 5.0, Vessel Overpressure Protection, 6.0, Thermal-Hydraulic Stability, and Section 7.0, Loss-Of-Coolant Accident Analysis describe particular aspects of the implementation of ARTS/MELLLA at the Hope Creek Generating Station (HCGS). These sections describe fuel dependent evaluations. The content of the sections accurately describes the approach HCGS is taking to justify and implement the ARTS/MELLLA bases. However, the assumptions and conclusions for the fuel dependent evaluations are based upon a representative core of GE14 fuel, and in some cases on existing analyses for plants similar to HCGS. The assumptions and conclusions relative to fuel dependent evaluations will be validated or, if required, updated based upon HCGS fuel dependent analyses results that will be submitted in a separate transmittal.

Attachment 4 Section 8.0, Containment Response, describes a non-fuel dependent evaluation. The section accurately describes the approach HCGS is taking to justify and implement the ARTS/MELLLA bases and correctly reflects the HCGS plant configuration. In addition, the assumptions and conclusions described are correct and applicable for HCGS. Therefore, the Section 8.0 non-fuel related evaluation is considered complete.

Attachment 4 Section 9.0, Reactor Internals Integrity, describes non-fuel dependent evaluations with the exception of Section 9.1, Reactor Internal Pressure Differences, which contains some fuel–dependent aspects. The section accurately describes the approach HCGS is taking to justify and implement the ARTS/MELLLA bases and correctly reflects the HCGS plant configuration. In addition, the assumptions and conclusions described are correct and applicable for HCGS. Although Section 9.1 has aspects that are fuel dependent, further fuel dependent evaluation is not required. The section describes that the existing HCGS ELLLA bases is bounding relative to the MELLLA application and therefore no specific fuel evaluations are required to justify the ARTS/MELLLA bases. Therefore, the Section 9.0 evaluations are considered complete.

Attachment 4 Section 10.0, Anticipated Transient Without SCRAM (ATWS), describes an evaluation that can be considered fuel dependent. The ATWS evaluation described in Section 10 is a HCGS plant specific evaluation; however, the evaluation uses a representative GE14 core. The content of the section accurately describes the approach HCGS is taking to justify and implement the ARTS/MELLLA bases. However, since the assumptions and conclusions for the fuel dependent aspects are based upon a representative core of GE14 fuel further fuel dependent validation is required. The assumptions and conclusions relative to the fuel dependent aspects will be validated or, if required, updated based upon HCGS fuel dependent analyses results that will be submitted in a separate transmittal.

Attachment 4 Sections 11.0, High Energy Line Break, and 12.0 Steam Dryer and Separator Performance describe non-fuel dependent evaluations relative to the effects of the ARTS/MELLLA bases. The sections accurately describe the approach HCGS is taking to justify and implement the ARTS/MELLLA bases and correctly reflect the HCGS plant configuration. In addition, the assumptions and conclusions described are correct and applicable for HCGS. Therefore, the sections are considered complete.

ARTS/MELLLA Implementation

The expanded operating domain includes the operating domain changes for ARTS/MELLLA consistent with approved operating domain improvements for other BWRs. The operating domain extends along approximately the 119% rod line to 100% power at approximately 77% core flow.

The APRM flow control trip reference cards will be modified to implement the proposed setpoint changes. Plant modifications will be evaluated in accordance with the requirements of 10 CFR 50.59 as part of the PSEG's design change process. No physical modifications are being made to the RBM.

The ARTS/MELLLA application is determined on a plant-specific basis via a safety and impact evaluation in meeting thermal and reactivity margins for BWR plants. When compared to the existing power/flow operating domain, operation in the MELLLA region results in plant operation along a higher rod line, which at off-rated operation allows for higher core power at a given core flow. This increases the fluid subcooling in the downcomer region of the reactor vessel and alters the power distribution in the core in a manner that can potentially affect steady-state operating thermal limit and transient/accident performances. The effect of this operating mode relative to fuel dependent analyses will be evaluated to confirm compliance with the required fuel thermal margins during plant operation. Attachment 4 presents the results of the safety analyses and system response evaluations for the non-fuel dependent tasks and the assumptions and conclusions that will be validated or updated for the fuel dependent tasks performed for operation of HCGS in the region above the ELLLA and up to the MELLLA boundary line.

With the proposed power/flow map expansion to include the MELLLA region, the upper boundary of the licensed operating domain would be extended to approximately the 119% rod line. To accommodate this expanded operating domain, the APRM flow biased scram and rod block trip setpoints would be revised. A high flow clamped setpoint for the APRM rod block would be added to maintain margin between the rod block and scram setpoints at high flows. The clamped trip setpoint does not meet any of the criteria in 10 CFR 50.36 for inclusion in TS. Consistent with NUREG-1433, the clamped setpoint will be controlled by plant procedures.

The high flow clamped setpoint for the flow-biased APRM scram would not be changed. Although they are part of the Hope Creek design configuration and Technical Specifications, the APRM flow-biased simulated thermal power scram line and the APRM flow-biased rod block line are not credited in any Hope Creek safety licensing analyses. The proposed setpoint changes would permit plant operation in the MELLLA region for operational flexibility purposes.

Representative results of the rod withdrawal error (RWE) event (without the RBM hardware) demonstrate the safety limit MCPR (SLMCPR) and fuel thermal-mechanical design limits are not exceeded without taking credit for the mitigating effect of the flow-biased RBM setpoints. PSEG will perform analyses for operating Cycle 13 and subsequent cycles to confirm these limits are not exceeded for an unblocked RWE event. On this basis, the RBM flow biased setpoints are being revised to alleviate operational constraints on operation in the MELLLA region.

One objective of the ARTS/MELLLA APRM improvements is to justify removal of the APRM trip setdown requirement (TS 3/4.2.2) using the following criteria:

1. MCPR safety limit shall not be violated as a result of any AOO.

2. All fuel thermal-mechanical design bases shall remain within the licensing limits.

3. Peak cladding temperature and maximum cladding oxidation fraction following a LOCA shall remain within the limits defined in 10 CFR 50.46.

Power and flow dependent adjustments to the MCPR and MAPLHGR (or LHGR) thermal limits will be determined using NRC approved analytical methods specified in TS 6.9.1.9. These adjustments will ensure that the above three criteria are met during operation at other than rated conditions without the APRM trip setdown.

The following additional changes would be made to reflect the deletion of TS 3/4.2.2:

a. References to TS 3/4.2.2 would be deleted from TS 3/4.4.1 Actions a.2 and a.3 and from footnotes to TS Tables 4.3.1.1-1 and 3.3.6-2.

b. Reference to the Maximum Fraction of Limiting Power Density (MFPLD) would be deleted from the specified conditions in the Applicability for Limiting Condition for Operation (LCO) 3.3.7.7.

c. Definitions for "Core Maximum Fraction of Limiting Power Density" (TS 1.8), "Fraction of Limiting Power Density" (TS 1.15), "Fraction of Rated Thermal Power," (TS 1.16) and "Maximum Fraction of Limiting Power Density" (TS 1.23) would be deleted from TS Section 1.0. These definitions are only used in TS 3/4.2.2 and LCO 3.3.7.7.

d. References to APRM trip setpoint adjustments would be deleted from TS Bases 2.2.1.

e. TS 3/4.2.1, "Average Planar Linear Heat Generation Rate," 3/4.2.3 "Minimum Critical Power Ratio" and "3/4.2.4 Linear Heat Generation Rate" and associated TS Bases will be revised as appropriate to include a description of power- and flow-dependent thermal limits. These TS changes will be supplied in a separate transmittal with the Hope Creek fuel dependent analyses.

The proposed changes are consistent with changes previously approved by the NRC for the LaSalle County Station, Units 1 and 2 (References 4 and 5), Dresden Nuclear Power Station, Units 2 and 3 (Reference 6), Quad Cities Nuclear Power Station, Units 1 and 2 (Reference 7), and Vermont Yankee Nuclear Power Station (Reference 10).

<u>RSLB AP and Jet Loads</u>
Attachment 4, Section 8.5 describes the evaluation of reactor asymmetric loads at the bounding condition (MELLLA minimum pump speed with RFWT). A more realistic blowdown mass and energy release profile was determined using the GE LAMB code in lieu of the Reference 1 methodology. The LAMB analysis considers the same pipe break separation time history as in the current basis, but ignores the fluid inertia effect, and thus, still provides conservative mass and energy release results. LAMB has been used in several plant licensing applications to calculate the blowdown mass flow rate and energy release profile in the event of an RSLB and has been accepted for licensing applications for power/flow map extension (MELLLA) associated with BWR extended power uprates (Reference 8). The LAMB methodology has been used to calculate the mass and energy releases for short-term post-LOCA containment response analysis for several applications. However, this is the first use of LAMB for calculating the mass and energy releases for AP loads.

Based on the mass and energy data calculated using LAMB for the MELLLA condition, annulus pressure time histories on the RPV and the biological shield wall were generated using the COMPARE code, taking into consideration the installed flow diverter which limits the break flow into the annulus to a nominal value of 25%. The AP pressure time histories were then converted to nodal force time histories by GE for application in the structural model.

A more realistic, but still conservative refinement was made to the method for calculating RPV and biological shield wall jet loads as a function of time. While the jet reaction / jet impingement load on the RPV has a sudden rise at the moment of the RSLB (because the RPV is the source of the high pressure fluid), the biological shield wall does not experience the jet reaction/impingement load until the flow diverter is pressurized by the break flow. This calculation is still conservative, because it assumes that the flows in the recirculation suction nozzle, the recirculation pipe on the side of the recirculation pump, and the flow diverter are established instantaneously, i.e., reach the quasi-steady state values, for a given RSLB pipe separation at each time step as if there is no inertia in the fluid. The calculational procedure followed ANSI/ANS-58.2-1988 (e.g., the loads were calculated at discrete times using Equation 6-2 of ANSI/ANS-58.2-1988), and the guidelines provided in paragraphs (2) and (3) of Section III.2.c of Standard Review Plan (SRP) 3.6.2. This is the first use of the above calculational procedure for calculating the jet reaction plus jet impingement loads on the RPV and the biological shield wall.

The MELLLA-based combined responses of the AP, jet reaction, jet impingement and pipe whip restraint loads, which form the load input to the RPV, vessel internals, CRD mechanism, shield wall, and main steam and recirculation piping, were found to be bounded by those of the current design calculation. PSEG evaluated the structural response of the remaining affected drywell piping and concluded that the MELLLA-based combined responses described above, which form the load input to the remaining affected drywell piping, are bounded by those of the current design calculations. PSEG also re-evaluated the containment internal structures for the MELLLA-based RSLB applicable load increases and determined that design margins, using the current design methods, still satisfy the current structural acceptance criteria.

Anticipated Transients Without Scram (ATWS)
Attachment 4, Section 10.0 describes the plant-specific analyses performed to demonstrate that the ATWS acceptance criteria are met for operation in the MELLLA region. All ATWS acceptance criteria were met. The peak vessel pressure and suppression pool temperature reported in Attachment 4 are for an assumed initial power level of 3952 MWt. Although the peak suppression pool temperature at hot shutdown is below the suppression pool temperature limit, the peak suppression pool temperature for post hot shutdown depressurization was also obtained. These results bound ATWS events at the current licensed thermal power (CLTP) since the peak integrated safety-relief valve (SRV) flow at time of hot shutdown for the MELLLA/CLTP condition is bounded by the peak SRV flow for the MELLLA/3952 MWt condition.

A peak suppression pool temperature and a corresponding containment pressure were obtained for the limiting ATWS scenario by modeling long-term post hot-shutdown depressurization with the SHEX code. The peak temperature was less than the maximum allowable bulk pool temperature of 201°F, which corresponds to a local pool temperature of 218°F at the SRV quencher locations. The 17°F difference between local and bulk pool temperature was conservatively determined based on a review of NEDC 30154 (Reference 9), specifically Case 2C (normal depressurization at isolated hot shutdown, 2 RHR loops). The local pool temperature limit of 218°F provides 20°F of subcooling at the SRV quencher location when credit is taken for both the containment pressure and the increase in pool level, as determined by the SHEX analysis. Consequently, the containment response following the limiting scenario satisfies ATWS acceptance criteria.

High Energy Line Breaks (HELBs)
Attachment 4, Section 11.0 states that the mass and energy release profiles assumed in the current design basis analysis for the Reactor Water Cleanup (RWCU) HELB analysis are not bounding at any of the four break locations for MELLLA conditions. PSEG evaluated the effect of the higher mass and energy release profiles and concluded the resulting subcompartment pressures,

temperatures and humidity levels are acceptable with respect to existing design criteria.

Conclusion:
The proposed changes will increase operating flexibility in power ascension and operation at rated power. Replacement of the APRM trip setdown requirement with more direct power and flow dependent thermal limits will reduce the need for manual setpoint or gain adjustments and allow more direct thermal limits administration. This will improve the human/machine interface, update thermal limits administration, increase reliability, and provide more direct protection of plant safety.

5. REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

PSEG Nuclear (PSEG) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment" as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   Proposed Change No. 1:
   The proposed change expands the power and flow operating domain by relaxing the restrictions imposed by the formulation of the flow-biased Average Power Range Monitor (APRM) rod block and scram trip setpoints and the Rod Block Monitor (RBM) flow-biased rod block trip setpoints. The APRM and RBM are not involved in the initiation of any accident; and the APRM flow-biased simulated thermal power scram and rod block functions are not credited in any Hope Creek safety licensing analyses. The revised evaluation of the rod withdrawal error (RWE) event will demonstrate acceptable results without crediting operation of the RBM. Anticipated operational occurrences and postulated accidents within the expanded operating domain will be evaluated using NRC approved methods to confirm they will remain bounded by NRC approved criteria.

   Since the proposed changes will not affect any accident initiator, or introduce any initial conditions that would result in NRC approved criteria being exceeded, and since the Average Power Range Monitor (APRM) and Rod Block Monitor (RBM) will remain capable of performing their design functions, the proposed change will not involve a significant increase in the probability or radiological consequences of an accident previously evaluated.

Proposed Change No. 2:
The proposed change will replace the flow-biased APRM scram and rod block trip setpoint setdown requirements with power and flow dependent adjustments to the Minimum Critical Power Ratio (MCPR) and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) or Linear Heat Generation Rate (LHGR) thermal limits. The adjustments to the thermal limits will be determined using NRC approved analytical methods as required by Technical Specification 6.9.1.9 and will be specified in the Core Operating Limits Report. The proposed change will have no effect upon any accident initiating mechanism. Adjustments to thermal limits will be determined using NRC approved methodologies. The power and flow dependent adjustments will ensure that the MCPR safety limit will not be violated as a result of any anticipated operational occurrence, that the fuel thermal and mechanical design bases will be maintained, and that the consequences of the postulated loss of coolant accident will remain within acceptable limits. Therefore, the proposed change will not involve a significant increase in the probability or radiological consequences of an accident previously evaluated.

Proposed Change No. 3:
The proposed change uses more realistic, but still conservative, methods of analysis to determine Annulus Pressurization (AP) and jet loads resulting from the postulated Recirculation Suction Line Break (RSLB). The loads are being evaluated at off-rated conditions which are more limiting than the current design basis (100% power, 100% flow) conditions. Loads resulting from the RSLB at off-rated conditions have been demonstrated to be bounded by the current design basis loads.

Since the proposed changes do not affect any accident initiator and since the RSLB AP and jet loads remain bounded by the current design basis loads, the proposed changes do not involve a significant increase in the probability or radiological consequences of an accident previously evaluated.

Therefore the proposed changes do not involve a significant increase in the probability or radiological consequences of any accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

Proposed Change No. 1:
Changing the formulation for the flow-biased APRM rod block and scram trip setpoints and the RBM flow biased rod block trip setpoint does not change their respective functions and manner of operation. The change does not introduce a sequence of events or introduce a new failure mode that would create a new or different type of accident. The APRM rod block trip setpoint will continue to block control rod withdrawal when core power significantly exceeds normal limits and approaches the scram level. The APRM scram trip setpoint will continue to initiate a scram if the increasing power/flow condition continues beyond the APRM rod block setpoint. The RBM will continue to prevent rod withdrawal when the flow-biased RBM rod block setpoint is reached. No new failure mechanisms, malfunctions, or accident initiators are being introduced by the proposed changes. In addition, operating within the expanded power flow map will not require any systems, structures or components to function differently than previously evaluated and will not create initial conditions that would result in a new or different accident. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Proposed Change No. 2:
The proposed change eliminates the requirement for setdown of the flow-biased APRM scram and rod block trip setpoints under specified conditions and will substitute adjustments to the MCPR and MAPLHGR or LHGR thermal limits. Because the thermal limits will continue to be met, no transient event will escalate into a new or different type of accident due to the initial starting conditions permitted by the adjusted thermal limits. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any previously evaluated.

Proposed Change No. 3:
The proposed changes to the methods of analysis to determine Annulus Pressurization (AP) and jet loads resulting from the postulated Recirculation Suction Line Break (RSLB) do not change the design function or operation of any plant equipment. No new failure mechanisms, malfunctions, or accident initiators are being introduced by the proposed changes. Therefore, the proposed

changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

Proposed Change No. 1:
The APRM rod block trip setpoint will continue to block control rod withdrawal when core power significantly exceeds normal limits and approaches the scram level. The APRM scram trip setpoint will continue to initiate a scram if the increasing power/flow condition continues beyond the APRM rod block setpoint. The RBM will continue to prevent rod withdrawal when the flow-biased RBM rod block setpoint is reached. The MCPR and MAPLHGR or LHGR thermal limits will be developed to ensure that fuel thermal-mechanical design bases shall remain within the licensing limits during a rod withdrawal error event and to ensure that the MCPR safety limit will not be violated as a result of a rod withdrawal error event. Operation in the expanded operating domain will not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. Anticipated operational occurrences and postulated accidents within the expanded operating domain will be evaluated using NRC approved methods. Therefore, the proposed change will not involve a reduction in the margin of safety.

Proposed Change No. 2:
Replacement of the APRM setpoint setdown requirement with power and flow dependent adjustments to the MCPR and MAPLHGR or LHGR thermal limits will ensure that margins to the fuel cladding Safety Limit are preserved during operation at other than rated conditions. The fuel cladding safety limit will not be violated as a result of any anticipated operational occurrence. The flow and power dependent adjustments will be determined using NRC approved methodologies. The flow and power dependent adjustments will also ensure the all fuel thermal-mechanical design bases shall remain within the licensing limits. The 10 CFR 50.46 acceptance criteria for the performance of the emergency core cooling system (ECCS) following postulated loss-of-coolant accidents (LOCAs) will continue to be met. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

Proposed Change No. 3:
Annulus Pressurization (AP) and jet loads resulting from the postulated Recirculation Suction Line Break (RSLB) remain bounded by the current design basis loads. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, PSEG concludes that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

BOLD UL

10 CFR 50, Appendix A, General Design Criterion (GDC) 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. The assumptions and conclusions relative to fuel dependent evaluations will be validated or, if required, updated based upon Hope Creek fuel dependent analyses results to ensure the requirements of GDC 10 continue to be met.

10 CFR 50 Appendix A, GDC 12 requires that the reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed. The assumptions and conclusions relative to fuel dependent evaluations will be validated or, if required, updated based upon Hope Creek fuel dependent analyses results to ensure the requirements of GDC 12 continue to be met.

10 CFR 50 Appendix A, GDC 50 requires that the reactor containment structure be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. Evaluations described in Attachment 4, Section 8.0 demonstrate that all containment parameters stay within their design limits.

10 CFR 50.46 sets forth acceptance criteria for the performance of the emergency core cooling system (ECCS) following postulated loss-of-coolant accidents (LOCAs). 10 CFR 50 Appendix K describes required and acceptable features of the evaluation models used to calculate ECCS performance. The assumptions and conclusions relative to fuel

dependent evaluations will be validated or, if required, updated based upon Hope Creek fuel dependent analyses results to ensure the requirements of 10 CFR 50.46 continue to be met.

10 CFR 50.49 establishes requirements for environmental qualification of electric equipment important to safety for nuclear power plants. Evaluations described in Attachment 4, Section 11.0 demonstrate acceptable results for steam line breaks and feedwater line break. For the RWCU HELB, the resulting subcompartment pressures, temperatures and humidity levels are acceptable with respect to existing design criteria.

10 CFR 50.62, in part, specifies the equivalent flow rate, level of boron concentration and boron-10 isotope enrichment required for BWR standby liquid control systems. The assumptions and conclusions relative to fuel dependent evaluations will be validated or, if required, updated based upon Hope Creek fuel dependent analyses results to ensure the requirements of 10 CFR 50.62 continue to be met.

PSEG has determined that the proposed changes will comply with the applicable regulatory requirements.

In conclusion, based on the considerations discussed above, (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.

6. ENVIRONMENTAL CONSIDERATION

PSEG has determined the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or a surveillance requirement. The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

7. REFERENCES

1. GE Nuclear Energy, "Technical Description - Annulus Pressurization Load Adequacy Evaluation," NEDO-24548, January 1979.

2. GE Nuclear Energy, "Increased Core Flow and Extended Load Line Limit Analysis for Hope Creek Generating Station, Unit 1, Cycle 2," NEDC-31487, November 1987 (GE proprietary information).

3. Amendment No. 15 to Facility Operating License No. NPF-57 for the Hope Creek Generating Station, March 15, 1988 (TAC No. 66775).

4. LaSalle County Station, Units 1 and 2 - Issuance of Amendments (TAC Nos. M89631 and M89632), April 13, 1995

5. LaSalle County Station, Units 1 and 2 - Issuance of Amendments Regarding Power Uprate (TAC Nos. MA6070 AND MA6071), May 9, 2000

6. Dresden Nuclear Power Station, Units 2 and 3 - Issuance of Amendments for Extended Power Uprate (TAC Nos. MB0844 and MB0845), December 21, 2001

7. Quad Cities Nuclear Power Station, Units 1 and 2 - Issuance of Amendments for Extended Power Uprate (TAC Nos. MB0842 and MB0843), December 21, 2001

8. GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," Licensing Topical Report NEDC-32424P-A, Class III (Proprietary), February 1999

9. GE Nuclear Energy, "Hope Creek Generating Station Suppression Pool Temperature Response," NEDC-30154, Class III (Proprietary), June 1983.

10. Vermont Yankee Nuclear Power Station - Issuance of Amendment Re: Implementation of ARTS/MELLLA (TAC NO. MB8070), April 14, 2004

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
REQUEST FOR LICENSE AMENDMENT

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Facility Operating License No. NPF-57 are affected by this change request:

Technical Specification	Page
INDEX	i vi xvii
1.8	1-2
1.15	1-3
1.16	1-3
1.23	1-4
Table 2.2.1-1	2-4
Bases 2.2.1	B 2-7
3/4.2.2	3/4 2-2
Table 4.3.1.1-1	3/4 3-8
Table 3.3.6-2	3/4 3-59
3/4.3.7	3/4 3-89
3/4.4.1	3/4 4-1 3/4 4-2
Bases 3/4.2.2	B 3/4 2-1

SECTION

SECTION	PAGE
1.0 DEFINITIONS	
1.1 ACTION	1-1
1.2 AVERAGE PLANAR EXPOSURE	1-1
1.3 AVERAGE PLANAR LINEAR HEAT GENERATION RATE	1-1
1.4 CHANNEL CALIBRATION	1-1
1.5 CHANNEL CHECK	1-1
1.6 CHANNEL FUNCTIONAL TEST	1-1
1.7 CORE ALTERATION	1-2
1.8 ~~CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY~~ DELETED	1-2
1.9 CORE OPERATING LIMITS REPORT	1-2
1.10 CRITICAL POWER RATIO	1-2
1.11 DOSE EQUIVALENT I-131	1-2
1.12 Ē-AVERAGE DISINTEGRATION ENERGY	1-2
1.13 EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	1-2
1.14 END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME	1-3
1.15 ~~FRACTION OF LIMITING POWER DENSITY~~ DELETED	1-3
1.16 ~~FRACTION OF RATED THERMAL POWER~~ DELETED	1-3
1.17 FREQUENCY NOTATION	1-3
1.18 IDENTIFIED LEAKAGE	1-3
1.19 ISOLATION SYSTEM RESPONSE TIME	1-3
1.20 LIMITING CONTROL ROD PATTERN	1-3
1.21 LINEAR HEAT GENERATION RATE	1-4
1.22 LOGIC SYSTEM FUNCTIONAL TEST	1-4
1.23 ~~MAXIMUM FRACTION OF LIMITING POWER DENSITY~~ DELETED	1-4
1.24 MEMBER(S) OF THE PUBLIC	1-4
1.25 MINIMUM CRITICAL POWER RATIO	1-4

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION		PAGE
3/4 2.2	~~APRM SETPOINTS~~ DELETED	3/4 2-2
3/4.2.3	MINIMUM CRITICAL POWER RATIO	3/4 2-3
3/4.2.4	LINEAR HEAT GENERATION RATE	3/4 2-5
3/4.3 INSTRUMENTATION		
3/4.3.1	REACTOR PROTECTION SYSTEM INSTRUMENTATION	3/4 3-1
	Table 3.3.1-1 Reactor Protection System Instrumentation	3/4 3-2
	Figure 4.3.1.1-1 Reactor Protection System Surveillance Requirements	3/4 3-7
3/4.3.2	ISOLATION ACTUATION INSTRUMENTATION	3/4 3-9
	Table 3.3.2-1 Isolation Actuation Instrumentation	3/4 3-11
	Table 3.3.2-2 Isolation Actuation Instrumentation Setpoints	3/4 3-22
	Table 4.3.2.1-1 Isolation Actuation Instrumentation Surveillance Requirements	3/4 3-28

INDEX

BASES

SECTION PAGE

3/4.0 APPLICABILITY.......................................B 3/4 0-1

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 SHUTDOWN MARGIN.......................................B 3/4 1-1

3/4.1.2 REACTIVITY ANOMALIES.......................................B 3/4 1-1

3/4.1.3 CONTROL RODS.......................................B 3/4 1-2

3/4.1.4 CONTROL ROD PROGRAM CONTROLS.......................................B 3/4 1-3

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM.......................................B 3/4 1-4

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR ~~HEAT GENERATION RATE~~.......................................B 3/4 2-1

3/4.2.2 ~~APRM SETPOINTS~~ DELETED.......................................B 3/4 2-1

3/4.2.3 MINIMUM CRITICAL POWER RATIO.......................................B 3/4 2-2

3/4.2.4 LINEAR HEAT GENERATION RATE.......................................B 3/4 2-3

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION.......................................B 3/4 3-1

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION.......................................B 3/4 3-2

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION.......................................B 3/4 3-2

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION.......................................B 3/4 3-3

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION....................................... B 3/4 3-4

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION.......................................B 3/4 3-4

3/4.3.7 MONITORING INSTRUMENTATION

Radiation Monitoring Instrumentation.......................................B 3/4 3-4

CORE ALTERATION

1.7 CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement), and

b. Control rod movement, provided there are no fuel assemblies in the associated core cell.

Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

DELETED

~~CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY~~

~~1.8 The CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD) shall be highest value of the FLPD which exists in the core.~~

CORE OPERATING LIMITS REPORT

1.9 The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.9. Plant operation within these limits is addressed in individual specifications.

CRITICAL POWER RATIO

1.10 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the applicable NRC-approved critical power correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.12 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.13 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.14 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker from initial movement of the associated:

a. Turbine stop valves, and

b. Turbine control valves.

The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

DELETED

~~FRACTION OF LIMITING POWER DENSITY~~

~~1.15 The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.~~

~~FRACTION OF RATED THERMAL POWER~~

~~1.16 The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.~~

DELETED

FREQUENCY NOTATION

1.17 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

IDENTIFIED LEAKAGE

1.18 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

1.19 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

LIMITING CONTROL ROD PATTERN

1.20 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

1.21 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.22 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e., all relays and contacts, all trip units, solid state logic elements, etc, of a logic circuit, from sensor through and including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

~~MAXIMUM FRACTION OF LIMITING POWER DENSITY~~ DELETED

~~1.23 The MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be highest value of the FLPD which exists in the core.~~

MEMBER(S) OF THE PUBLIC

1.24 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, it contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

MINIMUM CRITICAL POWER RATIO

1.25 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

OFF-GAS RADWASTE TREATMENT SYSTEM

1.26 An OFF-GAS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting reactor coolant system offgases from the main condenser evacuation system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

OFFSITE DOSE CALCULATION MANUAL

1.27 The OFFSITE DOSE CALCULATIONAL MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating Report and the Annual Radioactive Effluent Release Report required by Specifications 6.9.1.6 and 6.9.1.7.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-Upscale	66%	69%
1) Flow Biased	≤ 0.66(w-Δw)+~~51%~~** with a maximum of	≤ 0.66(w-Δw)+~~54%~~** with a maximum of
2) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
c. Fixed Neutron Flux-Upscale	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 1037 psig	≤ 1057 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches above instrument zero*	≥ 11.0 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed

*See Bases Figure B 3/4 3-1.

**The Average Power Range Monitor Scram function varies as a function of recirculation loop drive flow (w). Δw is defined as the difference in indicated drive flow (in percent of drive flow which produces rated core flow) between two loop and single loop operation at the same core flow. Δw = 0 for two recirculation loop operation. Δw = 9% for single recirculation loop operation.

LIMITING SAFETY SYSTEM SETTINGS

BASES

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks and because several rods must be moved to change power by a significant amount, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the trip level, the rate of power rise is not more than 5% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the Run position.

The APRM trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation for the case of the Fixed Neutron Flux-Upscale setpoint; i.e, for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer associated with the fuel. For the Flow Biased Simulated Thermal Power-Upscale setpoint, a time constant of 6 ± 0.6 seconds is introduced into the flow biased APRM in order to simulate the fuel thermal transient characteristics. A more conservative maximum value is used for the flow biased setpoint as shown in Table 2.2.1-1.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown. ~~The flow referenced trip setpoint must be adjusted by the specified formula in Specification 3.2.2 in order to maintain these margins when CMFLPD is greater than or equal to FRTP.~~

3. Reactor Vessel Steam Dome Pressure-High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This trip setpoint is effective at low power/flow conditions when the turbine control valve fast closure and turbine stop valve closure trip are bypassed. For a load rejection or turbine trip under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

DELETED

POWER DISTRIBUTION LIMITS

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-upscale scram trip setpoint (S) and flow biased neutron flux-upscale control rod block trip setpoint (S_{RB}) shall be established according to the following relationships:

TRIP SETPOINT	ALLOWABLE VALUE
$S \le (0.66(w-\Delta w)^{**} + 51\%)T$	$S \le (0.66(w-\Delta w)^{**} + 54\%)T$
$S_{RB} \le (0.66(w-\Delta w)^{**} + 42\%)T$	$S_{RB} \le (0.66(w-\Delta w)^{**} + 45\%)T$

where: S and S_{RB} are in percent of RATED THERMAL POWER,

W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 100 million lbs/hr.

T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER (FRTP) divided by the CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD). T is applied only if less than or equal to 1.0.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the APRM flow biased simulated thermal power-upscale scram trip setpoint and/or the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or S_{RB}, as above determined, initiate corrective action within 15 minutes and adjust S and/or S_{RB} to be consistent with the Trip Setpoint values* within 6 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The FRTP and the CMFLPD shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale scram and flow biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:

a. At least once per 24 hours,
b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
c. Initially and at least once per 12 hours when the reactor is operating with CMFLPD greater than or equal to FRTP.
d. The provisions of Specification 4.0.4 are not applicable.

---

* With CMFLPD greater than the FRTP, rather than adjusting the APRM setpoints, the APRM may be adjusted such that the APRM readings are greater than or equal to 100% times CMFLPD provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

**The Average Power Range Monitor Scram function varies as a function of recirculation loop drive flow (w). Δw is defined as the difference in indicated drive flow (in percent of drive flow which produces rated core flow) between two loop and single loop operation at the same core flow. $\Delta w = 0$ for two recirculation loop operation. $\Delta w = 9\%$ for single recirculation loop operation.

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High				
a. Float Switch	NA	Q	R	1, 2, 5(j)
b. Level Transmitter/Trip Unit	S	Q(k)	R	1, 2, 5(j)
9. Turbine Stop Valve - Closure	NA	Q	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	Q	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	W	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days.

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. ~~Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.~~

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

(g) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing recirculation loop flow (APRM % flow).

(h) This calibration shall consist of verifying the 6 ± 0.6 second simulated thermal power time constant.

(i) This item intentionally blank

(j) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(k) Verify the tripset point of the trip unit at least once per 92 days.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale		
i. Flow Biased	≤ 0.66 (w-Δw) + ~~40%~~ 65%*	≤ 0.66 (w-Δw) + ~~43%~~ 68%*
ii. High Flow Clamped	≤ ~~106%~~ 116%	≤ ~~109%~~ 119%
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux - Upscale	≤ 0.66(w-Δw) + ~~42%~~ 57%*	≤ 0.66(w-Δw) + ~~45%~~ 60%*
b. Inoperative	NA	NA
c. Downscale	≥ 4% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 1.0 x 10^5 cps	≤ 1.6 x 10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 3 cps	≥ 1.8 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 divisions of full scale	≤ 110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High (Float Switch)	109'1" (North Volume) 108'11.5" (South Volume)	109'3" (North Volume) 109'1.5" (South Volume)
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ 111% of rated flow	≤ 114% of rated flow
b. Inoperative	NA	NA
c. Comparator	≤ 10% flow deviation	≤ 11% flow deviation
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	NA

* The rod block function is varied as a function of recirculation loop flow (w) and Δw which is defined as the difference in indicated drive flow (in percent of drive flow which produces rated core flow) between two loop and single loop operation at the same core flow. ~~The trip setting of the Average Power Range Monitor Rod Block function must be maintained in accordance with Specification 3.2.2.~~

INSTRUMENTATION

TRAVERSING IN-CORE PROBE SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.7. The traversing in-core probe system shall be OPERABLE with:

a. Five movable detectors, drives and readout equipment to map the core, and

b. Indexing equipment to allow all five detectors to be calibrated in a common location.

APPLICABILITY: When the traversing in-core probe is used for:

a. Recalibration of the LPRM detectors, and

b.* Monitoring the APLHGR, LHGR, or MCPR~~, or MFLPD~~.

ACTION:

With the traversing in-core probe system inoperable, suspend use of the system for the above applicable monitoring or calibration functions. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.7 The traversing in-core probe system shall be demonstrated OPERABLE by normalizing each of the above required detector outputs within 72 hours prior to use for the LPRM calibration function.

*Only the detector(s) in the required measurement location(s) are required to be OPERABLE.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation with:

- a. Total core flow greater than or equal to 45% of rated core flow, or
- b. THERMAL POWER less than or equal to the limit specified in Figure 3.4.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation:

1. Within 4 hours:
   - a) Place the recirculation flow control system in the Local Manual mode, and
   - b) Reduce THERMAL POWER to ≤ 70% of RATED THERMAL POWER, and
   - c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit per Specification 2.1.2, and
   - d) Reduce the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit to a value specified in the CORE OPERATING LIMITS REPORT for single loop operation, and
   - e) DELETED.
   - f) Limit the speed of the operating recirculation pump to less than or equal to 90% of rated pump speed, and
   - g) Perform surveillance requirement 4.4.1.1.2 if THERMAL POWER is ≤ 38% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is ≤ 50% of rated loop flow.

2. Within 4 hours, reduce the Average Power Range Monitor (APRM) Scram Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specification~~s~~ 2.2.1 ~~and 3.2.2~~ otherwise, with the Trip Setpoints and Allowable Values associated with one trip system not reduced to those applicable for single recirculation loop operation, place the affected trip system in the tripped condition and within the following 6 hours, reduce the Trip Setpoints and Allowable Values of the affected channels to those applicable for single recirculation loop operation per Specification~~s~~ 2.2.1 ~~and 3.2.2~~.

3. Within 4 hours, reduce the APRM Control Rod Block Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specification~~s 3.2.2 and~~ 3.3.6; otherwise, with the Trip Setpoint and Allowable Values associated with one trip function not reduced to those applicable for single recirculation loop operation, place at least one affected channel

* See Special Test Exception 3.10.4.

ACTION (Continued)

in the tripped condition and within the following 6 hours, reduce the Trip Setpoints and Allowable Values of the affected channels to those applicable for single recirculation loop operation per Specification~~s 3.2.2 and~~ 3.3.6.

4. Within 4 hours, reduce the Rod Block Monitor Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specification 3.3.6; otherwise, with the Trip Setpoints and Allowable Values associated with one trip function not reduced to those applicable for single recirculation loop operation, place at least one affected channel in the tripped condition and within the following 6 hours, reduce the Trip setpoints and Allowable Values of the remaining channels to those applicable for single recirculation loop operation per Specification 3.3.6.

5. The provisions of Specification 3.0.4 are not applicable.

6. Otherwise be in at least HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant system recirculation loops in operation, immediately initiate action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours and initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.

c. With one or two reactor coolant system recirculation loops in operation and total core flow less than 45% but greater than 40% of rated core flow and THERMAL POWER greater than the limit specified in Figure 3.4.1.1-1:

1. Determine the APRM and LPRM* noise levels (Surveillance 4.4.1.1.4):

   a) At least once per 8 hours, and
   b) Within 30 minutes after the completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER.

2. With the APRM or LPRM* neutron flux noise levels greater than three times their established baseline noise levels, within 15 minutes initiate corrective action to restore the noise levels to within the required limits within 2 hours by increasing core flow to greater than 45% of rated core flow or by reducing THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1.

d. With one or two reactor coolant system recirculation loops in operation and total core flow less than or equal to 40% and THERMAL POWER greater than the limit specified in Figure 3.4.1.1-1, within 15 minutes initiate corrective action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 or increase core flow to greater than 40% within 4 hours.

* Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

BASES

The specifications in this section help assure that the fuel can be operated safely and reliably during normal operation. In addition, the limits specified in these specifications help ensure that the fuel does not exceed specified safety and regulatory limits during anticipated operational occurrences and design basis accidents. Specifically, these limits:

1. Ensure that the limits specified in 10CFR50.46 are not exceeded following the postulated design basis loss of coolant accident.

2. Ensure reactor operations remains within licensed, analyzed power/flow limits.

3. Ensure that the MCPR Safety Limit is not violated following any anticipated operational occurrence.

4. Ensure fuel centerline temperatures remain below the melting temperature and peak cladding strain remains below 1% during steady state operation.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is a measure of the average Linear Heat Generation Rate (LHGR) of all the fuel rods in a fuel assembly at any axial location. The Technical Specification APLHGR is the LHGR of the highest-powered rod divided by its local peaking factor. Limits on the APLHGR are specified to ensure that the fuel design limits are not exceeded. The limiting value of the APLHGR limit is specified in the CORE OPERATING LIMITS REPORT. The calculation procedure used to establish the APLHGR is based on a loss-of-coolant accident analysis. The post LOCA peak cladding temperature (PCT) is primarily a function of the APLHGR and is dependent only secondarily on the rod to rod power distribution within an assembly. The analytical modes used in evaluating the postulated loss-of-coolant accidents are described in References 1 and 2. These models are consistent with the requirements of Appendix K to 10CFR50.

For plant operation with a single recirculation loop, a lower value for the APLHGR limit is specified in the CORE OPERATING LIMITS REPORT. This lower value accounts for an earlier transition from nucleate boiling which occurs following a loss-of-coolant accident in the single loop operation compared to two loop operation.

3/4.2.2 ~~APRM SETPOINTS~~ DELETED

~~The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased simulated thermal power-upscale scram setting and the flow biased neutron flux-upscale control rod block trip setpoints must be adjusted to ensure that the MCPR does not become less than the fuel cladding Safety Limit or that > 1% plastic strain does not occur in the degraded situation. The scram setpoints and rod block setpoints are adjusted in accordance with the formula in Specification 3.2.2 whenever it is known that the existing power distribution would cause the design LHGR to be exceeded at RATED THERMAL POWER.~~

General Electric Company

AFFIDAVIT

I, George B. Stramback, state as follows:

(1) I am Manager, Regulatory Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the GE proprietary report NEDC-33066P, *Hope Creek Generating Station APRM/RBM/Technical Specifications / Maximum Extended Load Line Limit Analysis (ARTS/MELLLA),* Revision 1, Class III (GE Proprietary Information), dated May 2004. The proprietary information is delineated by a double underline inside double square brackets. Figures and large equation objects are identified with double square brackets before and after the object. In each case, the superscript notation{3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, <u>Critical Mass Energy Project v. Nuclear Regulatory Commission,</u> 975F2d871 (DC Cir. 1992), and <u>Public Citizen Health Research Group v. FDA,</u> 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, resulting in potential products to General Electric;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a., and (4)b, above.

(5) To address 10 CFR 2.390 (b) (4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed results of analytical models, methods and processes, including computer codes, which GE has developed, obtained NRC approval of, and applied to perform evaluations of transient and accident events in the GE Boiling Water Reactor ("BWR"). The development and approval of these system, component, and thermal hydraulic models and computer codes was achieved at a significant cost to GE, on the order of several million dollars.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 28th day of May 2004.

George B. Stramback

George B. Stramback
General Electric Company

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
REQUEST FOR LICENSE AMENDMENT

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by PSEG in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

Regulatory Commitment	Due Date
The assumptions and conclusions relative to fuel dependent evaluations will be validated or, if required, updated based upon Hope Creek fuel dependent analyses results that will be submitted in a separate transmittal. TS 3/4.2.1, "Average Planar Linear Heat Generation Rate," 3/4.2.3 "Minimum Critical Power Ratio" and "3/4.2.4 Linear Heat Generation Rate" and associated TS Bases will be revised as appropriate to include a description of power- and flow-dependent thermal limits. These TS changes will be supplied in a separate transmittal with the Hope Creek fuel dependent analyses.	09/30/04