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U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
FUEL VENDOR CHANGE
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

Reference: Margaret E. Harding (Global Nuclear Fuel) letter to NRC, "Transmittal of GNF-A Proprietary Report, NEDC-33107P, 'GEXL80 Correlation for SVEA96+ Fuel,' dated September 2003," dated November 24, 2003

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 10CFR50.91(b)(1), a copy of this submittal has been sent to the State of New Jersey.

The proposed changes support the use of General Electric Company (GE) fuel and reload analysis methods beginning with the upcoming Cycle 13. The proposed changes are consistent with NUREG-1433, "Standard Technical Specifications (STS) General Electric Plants, BWR/4," Revision 2.

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 pages affected by the proposed changes are provided in Attachment 2.

PSEG plans to include GE14 fuel in the reload for Cycle 13, which is currently scheduled to begin in Fall 2004. PSEG therefore requests approval of the proposed License Amendment by September 16, 2004, to be implemented within 60 days of the completion of the Hope Creek Fall 2004 refueling outage.

The reference letter requests NRC review and approval of the GEXL80 correlation for modeling the Westinghouse SVEA96+ fuel design by March 31, 2004 to support reload analysis for Cycle 13.

A001

PSEG proposes to meet with the staff at their earliest convenience to review the plans and schedule for transition to GE14 fuel at Hope Creek.

Should you have any questions or require additional information, please contact Mr. Paul Duke at (856) 339-1466.

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

Executed on 24 December 2003
(date)

John T. Carlin
Vice President - Nuclear Assessments

Attachments (2)

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REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
FUEL VENDOR CHANGE

Table of Contents

1. DESCRIPTION

This letter is a request to amend Operating License NPF-57 for the Hope Creek Generating Station. The proposed changes are being made to support the introduction of GE14 fuel. To facilitate this new fuel introduction (NFI), NRC approved GE calculation methodologies will be used exclusively to determine fuel thermal limits and reload transient analysis results. The changes to the Hope Creek Technical Specifications: 1) reflect the exclusive use of GE methods by removing references to other methodologies, 2) modify and add Action statements to provide further thermal limit control during Single Loop Operation (SLO) consistent with GE methodology requirements, 3) revise TS Definitions and TS requirements for average planar linear heat generation rate (APLHGR) consistent with NUREG-1433, "Standard Technical Specifications (STS) General Electric Plants, BWR/4," Revision 2 (Reference 1), and 4) correct an error in TS 6.9.1.9 introduced during implementation of a previous amendment. The references for TS Section 6.9.1.9 would be identified in the format prescribed in NUREG-1433, Rev. 2.

The TS Bases would also be revised to be consistent with GE methodology requirements. NRC approval for the GE methodologies and requirements was provided in Amendment 26 to GESTAR II, and included in GESTAR II Revision 14, June 2000.

The proposed changes are required to support the transition to General Electric Company (GE) fuel and reload analysis methods beginning with the upcoming Cycle 13 which will begin in Fall 2004.

2. PROPOSED CHANGE

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

1. One reference to ABB/CE calculational methodology would be deleted from the list of analytical methods that are used to determine the core operating limits in TS Section 6.9.1.9, "Core Operating Limits Report" (COLR). The references for TS Section 6.9.1.9 would be renumbered and identified in the format prescribed in NUREG-1433, Rev. 2.

2. Limiting Condition for Operation (LCO) 3.4.1.1, "Recirculation Loops," would be revised as follows:

    a. Action a.1.d would be revised to require the Average Planar Linear Heat Generation Rate (APLHGR) limit to be reduced to a value specified in the Core Operating Limits Report for SLO.

b. A new Action a.1.e would be added to require the LHGR limit to be reduced to a value specified in the COLR during SLO.

The associated TS Bases would also be revised to reflect the use of APLHGR and LHGR limits during SLO.

3. Limiting Condition for Operation (LCO) 3.2.1, "Average Planar Linear Heat Generation Rate," would be revised consistent with NUREG-1433, "Standard Technical Specifications (STS) General Electric Plants, BWR/4," Revision 2 (Reference 1). Specifically, the references to fuel type and average planar exposure would be deleted. TS 1.2, the definition for "Average Planar Exposure" would be deleted. The definition for "Average Planar Linear Heat Generation Rate" (APLHGR) in TS 1.3 would be revised consistent with NUREG-1433 to be applicable to the GE14 fuel design.

4. A reference in TS 6.9.1.9 to CENPD-397-P-A, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology," which was inadvertently deleted in a previous amendment, would be restored.

Changes to the TS Bases would also be made to reflect the application of NRC approved GE methodologies. The marked up Bases pages are also included in Attachment 2 of this submittal.

3. BACKGROUND

1. For the current operating cycle, the Hope Creek core contains a mixture of Westinghouse SVEA96+ and GE9B fuel. Core operating limits were determined using NRC approved Westinghouse methodology. PSEG plans to load GE14 fuel during the Hope Creek Fall 2004 refueling outage. NRC approved GE calculation methodologies will be used exclusively to determine fuel thermal limits and reload transient analysis results.

2. The current TS required action to reduce the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit to a value specified in the COLR for single loop operation is inconsistent with the approved GE methodology which establishes limits on APLHGR and LHGR for single loop operation.

3. Limiting Condition for Operation (LCO) 3.2.1, "Average Planar Linear Heat Generation Rate," refers to APLHGR limits for each fuel type as a function of average planar exposure. The APLHGR limits are established in accordance with the approved analytical methods listed in TS 6.9.1.9. The LCO 3.2.1 references to fuel type and average planar exposure are not needed since this information is located in the COLR.

4. TS Amendment 131 revised TS 6.9.1.9 to add a reference to Topical Report CENPD-397-P-A, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology." The reference was inadvertently deleted during preparation of the retyped pages for HC TS Amendment 145.

4. TECHNICAL ANALYSIS

1. TS 6.9.1.9 identifies the previously reviewed and approved analytical methods used to determine the core operating limits. The proposed change deletes one reference to ABB/CE calculational methodology and retains the reference to NEDE-24011-P-A which will be used exclusively to determine core operating limits beginning with Cycle 13. The references for TS Section 6.9.1.9 will be identified in the format consistent with NUREG-1433, Rev. 2. TS 6.9.1.9 will be revised to state that the COLR will contain the complete identification of each of the TS referenced topical reports used to prepare the COLR.

2. The proposed changes to LCO 3.4.1.1, Actions a.1.d and a.1.e are consistent with the approved GE methodology and ensure the appropriate adjustments are made to core operating limits for single loop operation.

3. The proposed change to LCO 3.2.1 removes unnecessary detail from the TS while continuing to ensure fuel design limits are not exceeded. The APLHGR limits will continue to be established in accordance with the approved analytical methods listed in TS 6.9.1.9.

   TS 1.2 is being deleted because the term "Average Planar Exposure" is being removed from LCO 3.2.1 and is not used elsewhere in the TS.

   The proposed change to TS 1.3 is consistent with NUREG-1433 and makes the definition for "Average Planar Linear Heat Generation Rate" (APLHGR) applicable to the GE14 fuel design.

4. The proposed change to TS 6.9.1.9 adding a reference to Topical Report CENPD-397-P-A is administrative in nature, correcting an error that was introduced during preparation of the retyped pages for HC TS Amendment 145.

The changes to the TS Bases are being made in support of the proposed TS changes and reflect the use of NRC reviewed and approved methods of evaluation.

5. REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

PSEG Nuclear LLC (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.

   The revised information and references relative to the fuel vendor's calculation methodologies throughout the Technical Specifications are considered to be administrative in nature because they reflect the NRC approved methodologies to be used by PSEG Nuclear LLC and the fuel vendor to develop operating and safety limits for the fuel and core designs. The changes to the Recirculation System Action statements ensure the appropriate adjustments are made to core operating limits for single loop operation, and the Core Operating Limits Report (COLR) will still be developed in accordance with NRC approved methods. These proposed changes do not alter the method of operating the plant and have no effect on the probability of an accident initiating event or transient.

   There are no significant increases in the radiological consequences of an accident previously evaluated. The basis of the COLR and the PSEG Nuclear LLC and fuel vendor calculation methodologies is to ensure that no mechanistic fuel damage is calculated to occur if the limits on plant operation are not violated. The COLR will continue to preserve the existing margin to fuel damage and the probability of fuel damage is not increased.

   Therefore, the proposed change does not involve an increase in the probability or radiological consequences of an 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.

   These changes do not involve any new method for operating the facility, any changes to setpoints, or any new facility modifications

for the rèload core opèration. No new initiating events or transients result from these changes.

The revised information and references relative to the fuel vendor's calculation methodologies throughout the Technical Specifications are considered to be administrative in nature because they reflect the NRC approved methodologies to be used by PSEG Nuclear LLC and the fuel vendor to develop operating and safety limits for the fuel and core designs. The changes to the Recirculation System Action statements ensure the appropriate adjustments are made to core operating limits for single loop operation, and the COLR will still be developed in accordance with NRC approved methods.

Therefore, the proposed Technical Specification changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

The revised information and references relative to the fuel vendor's calculation methodologies throughout the Technical Specifications are considered to be administrative in nature because they reflect the NRC approved methodologies to be used by PSEG Nuclear LLC and the fuel vendor to develop operating and safety limits for the fuel and core designs. The changes to the Recirculation System Action statements ensure the appropriate adjustments are made to core operating limits for single loop operation, and the COLR will still be developed in accordance with NRC approved methods. The proposed changes will continue to ensure that the plant is operated within specified acceptable fuel design limits. Therefore, the proposed Technical Specifications changes do 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

10 CFR 50.36(c)(2)(ii) Criterion 2 requires that TS LCOs include process variables, design features, and operating restrictions that are an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. LCO 3.4.1.1 requires adjustments to core operating limits for single loop operation. The proposed changes ensure the appropriate adjustments are made to core operating limits for single loop operation. The proposed change to LCO 3.2.1 continues to ensure fuel design limits are not exceeded.

10 CFR 50.36(c)(5) requires that TS will include provisions relating to organization and management, procedures, record keeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. The proposed change to TS 6.9.1.9 lists the NRC-approved methods that will be used to determine core operating limits.

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. NUREG-1433, "Standard Technical Specifications - General Electric Plants, BWR/4," Revision 2.

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
REVISIONS TO THE TECHNICAL SPECIFICATIONS

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 xxv
1.2	1-1
1.3	1-1
3/4.2.1	3/4 2-1
3/4.4.1	3/4 4-1
6.9.1.9	6-21
References	6-26
Bases 2.1.2	B 2-2
Bases 3/4.1.1	B 3/4 1-1
Bases 3/4.1.4	B 3/4 1-3 B 3/4 1-5
Bases 3/4.2.1	B 3/4 2-1
Bases 3/4.2.3	B 3/4 2-2
Bases 3/4.2.4	B 3/4 2-3
Bases 3/4.4.1	B 3/4 4-1

Insert A (TS 6.9.1.9)

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel (GESTAR-II)"

2. CENPD-397-P-A, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology"

The CORE OPERATING LIMITS REPORT will contain the complete identification for each of the TS referenced topical reports used to prepare the CORE OPERATING LIMITS REPORT (i.e., report number title, revision, date, and any supplements).

Insert B (Bases 3/4.2.4)

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. This specification assures that the Linear Heat Generation Rate (LHGR) in any fuel rod is less than the design linear heat generation even if fuel pellet densification is postulated. Limits on LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs), and to ensure that the peak clad temperature (PCT) during postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46. Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during normal operation or the anticipated operational occurrences identified in Reference 1.

The analytical methods and assumptions used in evaluating the fuel system design limits are presented in Reference 1. The analytical methods and assumptions used in evaluating AOOs and normal operation that determine the LHGR limits are presented in Reference 1.

LHGR limits are developed as a function of exposure to ensure adherence to fuel design limits during the limiting AOOs. The exposure dependent LHGR limits are reduced by an LHGR multiplier (LHGRFAC) at various operating conditions to ensure that all fuel design criteria are met for normal operation and AOOs. A complete discussion of the analysis code is provided in Reference 2.

For single recirculation loop operation, the LHGRFAC multiplier is limited to a maximum value as given in the CORE OPERATING LIMITS REPORT. This maximum limit is due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe cladding heatup during a LOCA.

SECTION

INDEX

ADMINISTRATIVE CONTROLS

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

~~AVERAGE PLANAR EXPOSURE~~

~~1.2 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.~~ DELETED

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle. at that height

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, display, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is tested.

shall be less than or equal to

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

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

ACTION:

With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT:

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 a LIMITING CONTROL ROD PATTERN for APLHGR.

d. The provisions of Specification 4.0.4 are not applicable.

Reduce the LINEAR HEAT GENERATION RATE (LHGR) limit to a value specified in the CORE OPERATING LIMITS REPORT for single loop operation, and

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*.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

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 Specifications 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 Specifications 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 Specifications 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.

CORE OPERATING LIMITS REPORT (Continued)

the following documents:

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC as applicable in ~~References 1, 2~~ and 3.

INSERT A

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the USNRC Administrator, Region 1, within the time period specified for each report.

6.9.3 Violations of the requirements of the fire protection program described in the Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the USNRC Administrator, Region 1, via the Licensee Event Report System within 30 days.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

SPECIAL REPORTS

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

6.15 TECHNICAL SPECIFICATION (TS) BASES CONTROL PROGRAM

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. PSEG may make changes to the Bases without prior NRC approval provided the changes do not require either of the following:

   1. A change in the TS incorporated in the License, or

   2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. Proposed changes to the Bases that require either condition of Specification 6.15.b above shall be reviewed and approved by the NRC prior to implementation.

d. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

e. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

REFERENCES

1. CENPD-300-P-A, "reference Safety Report for Boiling Water Reactors Reload Fuel," (latest approved revision)

2. NEDE-24011-P-A (latest approved revision), "General Electric Standard Application for Reactor Fuel (GESTAR-II)"

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using a statistical model that combines all of the uncertainties in operating parameters and in the procedures used to calculate critical power. Calculation of the Safety Limit MCPR is defined in Reference 1 ~~for GE fuel and Reference 2 for ABB/CE fuel~~

Reference:

1. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A (The approved revision at the time the reload analyses are performed. The approved revision number shall be identified in the CORE OPERATING LIMITS REPORT.)

~~2. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactors Reload Fuel" (The approved revision at the time the reload analyses are performed. The approved revision number shall be identified in the CORE OPERATING LIMITS REPORT.)~~

3/4.1.1 SHUTDOWN MARGIN

SHUTDOWN MARGIN (SDM) requirements are specified to ensure:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Bases Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SDM can be demonstrated by using solely analytical methods or by performing a test. SDM can be measured only by performing a test. A test involves collecting data with the reactor at a specified condition or series of conditions. The primary purpose of a SDM Demonstration is to ensure that SDM is equal to or greater than the SDM Limit for a specific core exposure. The primary purpose of a SDM Measurement is to provide SDM in % delta k/k that can be used for: 1) ensuring that SDM is equal to or greater than the SDM Limit for a range of core exposures, 2) determining the need for additional SDM Measurements during the cycle, 3) providing a benchmark for the core design (design vs. actual SDM), and 4) providing a benchmark for potential future analysis of SDM for such events as control rods incapable of full insertion. This higher level of application requires that a SDM Measurement is determined from testing and not through solely analytical methods. Since a SDM Measurement satisfies the primary purpose of a SDM Demonstration, it can be considered a special type of SDM Demonstration.

All SDM Demonstrations involve some usage of analytical methods. The performance of tests lessens the usage of analytical methods, reduces uncertainty in the results, and thus requires a smaller SDM Limit needed to show adequate SDM. At one end of the spectrum is a series of local criticals where both SDM and the highest worth control rod are determined by test. Although this technique has the minimum uncertainty and thus has the smallest SDM Limit, it still uses analytical methods to determine the worth of all the other control rods. At the other end of the spectrum is usage of solely analytical methods prior to core verification. This technique has the maximum uncertainty and thus has the largest SDM Limit.

The SDM Limit must be increased if the highest worth control rod is determined solely analytically versus a test using the reactor (requires a series of local criticals). This higher limit accounts for uncertainties in the calculation of the highest worth control rod.

SDM is demonstrated to satisfy a variety of OPCON 5 surveillances at the beginning of each cycle and, if necessary, at any future entry to OPCON 5 during the cycle if the assumptions of the previous SDM Demonstration are no longer valid. In most situations, the SDM Demonstration will be based solely on analytical methods and a test will not be performed. If SDM is demonstrated by using solely analytical methods, then SDM must be adjusted to account for

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 10% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RWM to be OPERABLE when THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER provides adequate control.

The RWM provides automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.4.9 of the FSAR and the techniques of the analysis are presented in ~~a topical report~~ Reference 1.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods. Operability of a RBM channel is assured for a given control rod when ≥ 50% of the LPRM inputs for each detector level are available for that rod. When < 50% of the LPRM inputs on either detector level are available, a case-by-case evaluation of channel operability is required.

rate, solution concentration or boron equivalent to meet the ATWS Rule must not invalidate the original system design basis. Paragraph (c)(4) of 10 CFR 50.62 states that:

> "Each boiling water reactor must have a Standby Liquid Control System (SLCS) with a minimum flow capacity and boron control equivalent in control capacity to 86 gallons per minute of 13 weight percent sodium pentaborate solution (natural boron enrichment)."

The described minimum system parameters (82.4 gpm, 13.6 percent concentration and natural boron equivalent) will ensure an equivalent injection capability that exceeds the ATWS Rule requirement. The stated minimum allowable pumping rate of 82.4 gallons per minute is met through the simultaneous operation of both pumps.

The standby liquid control system will also provide the capability to raise and maintain the long-term post-accident coolant inventory pH levels to 7 or above. This will prevent significant fractions of the dissolved iodine from being converted to elemental iodine and then re-evolving to the containment atmosphere.

1. ~~CENPD-284-P-A, "Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification," July, 1996.~~

NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved version).

3/4.2 POWER DISTRIBUTION LIMITS

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.

can assumed conservatively small

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

fuel

assumed in the LOCA analyses

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 models 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.

models

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

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.

The exposure dependent APLHGR limits are reduced by an APLHGR Multiplier (MAPFAC) at various operating conditions to ensure that all fuel design criteria are met for normal operation and LOCA.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.2.3 is obtained.

~~The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-3 that are input to an ABB/CE core dynamic behavior transient computer program. The codes used to evaluate transients are discussed in Reference 2.~~

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state (MCPR$_f$ and MCPR$_p$ respectively) to ensure adherence to fuel design limits during the worst transient with moderate frequency that is postulated in Chapter 15.

Flow dependent MCPR limits (MCPR$_f$) are determined by steady state methods using a three dimensional BWR simulator code (Reference 2). MCPR$_f$ curves are provided based on the maximum credible flow runout transient (i.e., runout of both loops).

~~A three dimensional BWR simulator code and a one dimensional transient code (Reference 2) determine power dependent MCPR limits (MCPR$_p$). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scram limits are bypassed, high and low MCPR$_p$ operating limits are provided for operation between 25% of RATED THERMAL POWER and the bypass power levels.~~

MINIMUM CRITICAL POWER RATIO (Continued)

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

INSERT B

3/4.2.4 LINEAR HEAT GENERATION RATE

~~This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.~~

References:

~~1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566, November 1975.~~

~~2. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactors Reload Fuel" (The approved revision at the time the reload analyses are performed. The approved revision number shall be identified in the CORE OPERATING LIMITS REPORT.)~~

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved version).

2. NEDO-24154-A, "Qualification of the One-Dimensional Core Transient Model (ODYN) for Boiling Water Reactors," August 1986, and NEDE-24154-P-A, Supplement 1, Volume 4, Revision 1, February 2000.

LHGR limits are decreased by the factor given in the COLR,

3/4.4.1 RECIRCULATION SYSTEM

the CORE OPERATING LIMITS REPORT (COLR)

The impact of single recirculation loop operation upon plant safety is assessed and shows that single loop operation is permitted if the MCPR fuel cladding Safety Limit is increased as noted by Specification 2.1.2, APRM scram and control rod block setpoints are adjusted as noted in Tables 2.2.1-1 and 3.3.6-2 respectively. MAPLHGR limits are decreased by the factor given in Specification 3.2.1 and MCPR operating limits are adjusted per Specification 3/4.2.3

as specified in the COLR

Additionally, surveillance on the pump speed of the operating recirculation loop is imposed to exclude the possibility of excessive core internals vibration. The surveillance on differential temperatures below 38% THERMAL POWER or 50% rated recirculation loop flow is to mitigate the undue thermal stress on vessel nozzles, recirculating pump and vessel bottom head during the extended operation of the single recirculation loop mode.

recirculation

An inoperable jet pump is not in itself a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis-accident, increase the blowdown area and reduce the capability of reflooding the core, thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation.

Recirculation loop flow mismatch limits are in compliance with the ECCS LOCA analysis design criteria for two recirculation loop operation. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. In the case where the mismatch limits cannot be maintained during two loop operation, continued operation is permitted in a single recirculation loop mode.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Sudden equalization of a temperature difference > 145°F between the reactor vessel bottom head coolant and the coolant in the upper region of the reactor vessel by increasing core flow rate would cause undue stress in the reactor vessel bottom head.

The objective of BWR plant and fuel design is to provide stable operation with margin over the normal operating domain. However, at the high power/low flow corner of the operating domain, a small probability of limit cycle neutron flux oscillations exists depending on combinations of operating conditions (e.g., rod pattern, power shape). To provide assurance that neutron flux limit cycle oscillations are detected and suppressed, APRM and LPRM neutron flux noise levels should be monitored while operating in this region.

Conservative

Stability tests at operating BWRs were reviewed to determine a generic region of the power/flow map in which surveillance of neutron flux noise levels should be performed. A conservation decay ratio of 0.6 was chosen as the bases for determining the generic region for surveillance to account for the plant to plant variability of decay ratio with core and fuel designs. This generic region has been determined to correspond to a core flow of less than or equal to 45% of rated core flow and a THERMAL POWER greater than that specified in Figure 3.4.1.1-1.