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APR 2 7 2004

PSEG
Nuclear LLC

LR-N04-0183
LCR H04-03

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

REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
SAFETY LIMIT MINIMUM CRITICAL POWER RATIO
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

Reference: LR-N03-0511, "Request for Change to Technical Specifications: Fuel Vendor Change," dated December 24, 2003.

Pursuant to 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby requests a revision to the Technical Specifications 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.

This proposed change will revise the Safety Limit Minimum Critical Power Ratio (SLMCPR) values for two recirculation loop and one recirculation loop operation. Each safety limit value will be applicable for all fuel types in the Hope Creek core. In Reference 1, PSEG requested changes to the Technical Specifications to support the use of GE14 fuel and General Electric Company (GE) reload analysis methods beginning with the upcoming Cycle 13.

PSEG has evaluated the proposed changes in accordance with 10 CFR 50.91(a)(1), using the criteria in 10 CFR 50.92(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.

Attachment 3 provides a summary of the relevant input parameters and results of the SLMCPR evaluations for Cycle 13. Attachment 3 contains information that Global Nuclear Fuel - Americas, LLC (GNF) considers to be proprietary. GNF requests that the proprietary information in Attachment 3 be withheld from public disclosure in

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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 3.

accordance with 10 CFR 2.390. An affidavit in support of this request is included with Attachment 3. A non-proprietary version of the GNF document is provided in Attachment 4.

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 October 29, 2004, to be implemented within 60 days of NRC approval.

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 27 April 2004
(date)

Michael H. Brothers
Vice President - Site Operations

Attachments (4)

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 3.

C: Mr. H. Miller, Administrator – Region I
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King of Prussia, PA 19406

U. S. Nuclear Regulatory Commission
ATTN: Mr. D. Collins, Licensing Project Manager - Hope Creek
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USNRC Senior Resident Inspector – Hope Creek (X24)

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REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
SAFETY LIMIT MINIMUM CRITICAL POWER RATIO

Table of Contents

1. DESCRIPTION

This letter is a request to amend Operating License NPF-57 for the Hope Creek Generating Station. The proposed change revises Technical Specification (TS) 2.1.2 to incorporate revised Safety Limit Minimum Critical Power Ratio (SLMCPR) values for two recirculation loop and single recirculation loop operation.

The revised SLMCPR values are required to permit resumption of power operation after the Fall 2004 refueling outage. Therefore, PSEG Nuclear LLC (PSEG) requests approval of the proposed license amendment by October 29, 2004.

2. PROPOSED CHANGE

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

The SLMCPR value for two recirculation loop operation contained in TS 2.1.2 would be changed to 1.06 for all fuel types. The SLMCPR value for single recirculation loop operation would be changed to 1.08 for all fuel types.

Changes to the TS Bases would also be made to reflect the changes to TS 2.1.2. The marked up Bases pages are also included in Attachment 2 of this submittal.

3. BACKGROUND

The fuel cladding is one of the principal barriers to the release of radioactive materials to the environment. The SLMCPR is applied to ensure that fuel cladding integrity is not lost due to overheating during normal plant operation and anticipated operational occurrences. The SLMCPR is set such that no mechanistic fuel damage is calculated to occur if the limit is not violated. Because the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling (i.e., transition boiling) have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that the onset of boiling transition would not necessarily result in damage to boiling water reactor 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 SLMCPR is defined as the critical power ratio 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 core for the current operating cycle (Cycle 12) consists of a mixture of SVEA96+ and GE9B fuel. The current SLMCPR values for the SVEA96+ fuel are determined in accordance with the methodology described in CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactors Reload Fuel," Revision 0 (Reference 1). SLMCPR values for GE fuel are determined in accordance with the methodology described in Revision 13 to NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel (GESTAR II)" (Reference 2).

PSEG plans to load GE14 fuel for Cycle 13. In Reference 3, PSEG requested changes to the Technical Specifications to support the introduction of GE14 fuel in Cycle 13. The TS changes proposed in Reference 3 reflect the exclusive use of General Electric Company (GE) calculational methods to determine core operating limits, including MCPR.

Recently completed cycle-specific calculations for Cycle 13 have resulted in lower SLMCPR values of 1.06 and 1.08 for two recirculation loop and single recirculation loop operation, respectively.

4. TECHNICAL ANALYSIS

The proposed change revises the SLMCPR values in TS 2.1.2 to reflect the results of a cycle-specific evaluation for Cycle 13. This SLMCPR evaluation was performed using NRC approved methodology, as described in Amendment 25 to NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel (GESTAR II)," (Reference 2) and other NRC approved vendor documents referenced in Attachment 3. The analysis methodology incorporates plant and cycle-specific parameters that include: 1) the expected reference loading pattern; 2) conservative variations of projected control blade patterns; 3) the actual bundle parameters; 4) the full cycle exposure range; and 5) reduced power distribution uncertainties associated with the process computer system. PSEG intends to use the GE 3D-MONICORE core monitoring system beginning with Cycle 13.

Table 1 in Attachment 3 provides a summary of the relevant input parameters and results of the SLMCPR value determination for the Cycle 13 core, including identification of core design characteristics. Table 2 of Attachment 3 provides the uncertainties used in the SLMCPR evaluation.

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 SLMCPR ensures that no mechanistic fuel damage occurs in the core if the limit is not violated. The revised SLMCPR values maintain the appropriate conservative margin to boiling transition and the probability of fuel damage is not increased. The derivation of the revised SLMCPR values specified in the Technical Specifications has been performed using NRC approved methods and uncertainties. The analysis methodology incorporates appropriate cycle-specific parameters and uncertainties in determining the revised SLMCPR values. The analyses do not change the method of operating the plant and have no effect on the probability of an accident initiating event or transient. The revised SLMCPR values do not affect the performance of systems or components used to mitigate the consequences of accidents previously evaluated.

   Therefore, the proposed change does not involve a significant 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.

   The revised SLMCPR values specified in the Technical Specifications have been calculated in accordance with NRC approved methods and uncertainties. The changes do not involve any new method for operating the facility and do not involve any facility modifications. No new initiating events or anticipated operational occurrences result from these changes.

Therefore, the proposed change does 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.

   The revised SLMCPR values are calculated using NRC approved methods and uncertainties, The revised SLMCPR values continue to ensure that greater than 99.9% of all fuel rods in the core are expected to avoid boiling transition if the safety limits are not violated, thereby maintaining the fuel cladding integrity during normal plant operation and anticipated operational occurrences.

   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

10 CFR 50.36 states that safety limits shall be included in Technical Specifications. General Design Criterion (GDC) 10 of Appendix A to 10 CFR 50 states that the reactor core and associated coolant, control, and protection systems shall 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 proposed SLMCPR values will continue to ensure that 99.9% of the fuel rods in the core are not expected to experience boiling transition during any condition of normal operation, including the effects of anticipated operational occurrences.

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. However, 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. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactors Reload Fuel," Revision 0

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

3. LR-N03-0511, "Request For Change To Technical Specifications: Fuel Vendor Change," dated December 24, 2003

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
2.1.2	2-1
Bases 2.0	B 2-1
Bases 2.1.2	B 2-2

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 With reactor steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow:

The MINIMUM CRITICAL POWER RATIO (MCPR) ~~for GE fuel~~ shall be ≥ ~~1.10~~ 1.06 for two recirculation loop operation and shall be ≥ ~~1.12~~ 1.08 for single recirculation loop operation. ~~The MCPR for ABB/CE fuel shall be ≥ 1.10 for two recirculation loop operation and shall be ≥ 1.13 for one recirculation loop operation.~~

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With reactor steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow and the MCPR below the values for the fuel stated in LCO 2.1.2, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

2.1 SAFETY LIMITS

BASES

2.0 INTRODUCTION

1.06

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such
1.08 that the MCPR ~~for GE fuel~~ is ≥ ~~1.10~~ for two recirculation loop operation and ≥ ~~1.12~~ for single recirculation loop operation. ~~and the MCPR for ABB/CE fuel is ≥ 1.10 for two recirculation loop operation and ≥ 1.13 for single recirculation loop operation.~~ These MCPR values represent a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the applicable NRC-approved critical power correlations are not valid for all critical power calculations performed at reduced pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

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/GE 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.)~~

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

Additional Information Regarding the Cycle Specific SLMCPR for Hope Creek Cycle 13 (proprietary version)

Affidavit

I, Glen A. Watford, state as follows:

(1) I am General Manager, Performance Services – GE Nuclear Energy 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 attachment, "Additional Information Regarding the Cycle Specific SLMCPR for Hope Creek Cycle 13," dated April 19, 2004. GNF proprietary information is indicated by enclosing it in double brackets. 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 or licensee, GNF-A 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 and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information," and some portions also qualify under the narrower definition of "trade secret," within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 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 GNF-A's competitors without license from GNF-A 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 cost or price information, production capacities, budget levels, or commercial strategies of GNF-A, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future GNF-A customer–funded development plans and programs, of potential commercial value to GNF-A;

e. 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 the 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 GNF-A, and is in fact so held. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in (6) and (7) following. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GNF-A, 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.

(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, or subject to the terms under which it was licensed to GNF-A. Access to such documents within GNF-A 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 GNF-A 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) is classified as proprietary because it contains details of GNF-A's fuel design and licensing methodology.

The development of the methods used in these analyses, along with the testing, development and approval of the supporting methodology was achieved at a significant cost, on the order of several million dollars, to GNF-A or its licensor.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GNF-A's competitive position and foreclose or reduce the availability of profit–making opportunities. The fuel design and licensing methodology is part of GNF-A'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 GNF-A or its licensor.

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.

GNF-A's competitive advantage will be lost if its competitors are able to use the results of the GNF-A 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 GNF-A 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 GNF-A of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining 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 at Wilmington, North Carolina, this 19th day of April, 2004.

19 April 2004

Glen A. Watford
GE Nuclear Energy

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

Additional Information Regarding the Cycle Specific SLMCPR for Hope Creek Cycle 13
(non-proprietary version)

Proprietary Information Notice

This document is the GNF non-proprietary version of the GNF proprietary report. From the GNF proprietary version, the information denoted as GNF proprietary (enclosed in double brackets) was deleted to generate this version.

References

[1] Letter, Frank Akstulewicz (NRC) to Glen A. Watford (GE), "Acceptance for Referencing of Licensing Topical Reports NEDC-32601P, Methodology and Uncertainties for Safety Limit MCPR Evaluations; NEDC-32694P, Power Distribution Uncertainties for Safety Limit MCPR Evaluation; and Amendment 25 to NEDE-24011-P-A on Cycle Specific Safety Limit MCPR," (TAC Nos. M97490, M99069 and M97491), March 11, 1999.

[2] Letter, Thomas H. Essig (NRC) to Glen A. Watford (GE), "Acceptance for Referencing of Licensing Topical Report NEDC-32505P, Revision 1, R-Factor Calculation Method for GE11, GE12 and GE13 Fuel," (TAC Nos. M99070 and M95081), January 11, 1999.

[3] General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, NEDO-10958-A, January 1977.

[4] Letter, Glen A. Watford (GNF-A) to U. S. Nuclear Regulatory Commission Document Control Desk with attention to R. Pulsifer (NRC), "Confirmation of 10x10 Fuel Design Applicability to Improved SLMCPR, Power Distribution and R-Factor Methodologies", FLN-2001-016, September 24, 2001.

[5] Letter, Glen A. Watford (GNF-A) to U. S. Nuclear Regulatory Commission Document Control Desk with attention to J. Donoghue (NRC), "Confirmation of the Applicability of the GEXL14 Correlation and Associated R-Factor Methodology for Calculating SLMCPR Values in Cores Containing GE14 Fuel", FLN-2001-017, October 1, 2001

[6] Letter, Glen A. Watford (GNF-A) to U. S. Nuclear Regulatory Commission Document Control Desk with attention to J. Donoghue (NRC), "Final Presentation Material for GEXL Presentation – February 11, 2002", FLN-2002-004, February 12, 2002.

[7] GEXL80 Correlation for SVEA96+ Fuel, NEDC-33107P Revision 0, Class III, September 2003.

Attachment | Additional Information Regarding the Cycle Specific SLMCPR for Hope Creek Cycle 13 | 15 April 2004

Discussion

The SLMCPR evaluations for the Hope Creek Cycle 13 were performed using NRC approved methodology and uncertainties [1]. Table 1 summarizes the relevant input parameters and results for Cycle 13. Additional information is provided in response to NRC questions on similar submittals regarding changes in Technical Specification values of SLMCPR. NRC questions pertaining to how GE14 applications satisfy the conditions of the NRC SER[1] have been addressed in Reference [4]. Other generically applicable questions related to application of the GEXL14 correlation, and to the applicable range for the R-factor methodology, are addressed in Reference [5]. Items that require a plant/cycle specific response are presented below.

In general, the calculated safety limit is dominated by two key parameters: (1) flatness of the core bundle-by-bundle MCPR distributions, and (2) flatness of the bundle pin-by-pin power/R-factor distributions. Greater flatness in either parameter yields more rods susceptible to boiling transition and thus a higher calculated SLMCPR. The value of these parameters on Hope Creek Cycle 13 is summarized in Table 1.

The core loading information for Hope Creek Cycle 13 is provided in Figure 1. The impact of the fuel loading pattern differences on the calculated SLMCPR is correlated to the values of [[

{3}]]

[[{3}]]

Pin-by-pin power distributions are characterized in terms of R-factors using the NRC approved methodology [2]. For the Hope Creek Cycle 13 limiting case analyzed at EOR, [[{3}]].

The revised power distribution methodology was used for the Hope Creek Cycle 13 analysis. This methodology has been justified, reviewed and approved by the NRC (reference NEDC-32601P-A). When applying the revised model to calculate a lower SLMCPR, the conservatism that remains was reviewed, approved and documented by the USNRC. It was noted on page A-24 of NEDC-32601P-A [[

{3}]].

The SLMCPR was calculated for Hope Creek Cycle 13 using the reduced power distribution uncertainties described in Reference [1].

These calculations use the GEXL14 correlation for GE14 fuel and GEXL80 correlation for SVEA96+ fuel (Reference [7]). [[

{3}]]

The Two Loop and SLO SLMCPR values calculated for Hope Creek Cycle 13 are shown in Table 1.

Summary

The calculated 1.06 SLMCPR and 1.08 SLO SLMCPR for Hope Creek Cycle 13 are consistent with expectations [[

[3]]] these values are appropriate when the approved methodology and the reduced uncertainties given in NEDC-32601P-A and NEDC-32694P-A are used.

Based on all of the information and discussion presented above, it is concluded that the calculated 1.06 SLMCPR and 1.08 SLO SLMCPR for the Hope Creek Cycle 13 core are appropriate.

Table 1
Hope Creek Cycle 13 SLMCPR

QUANTITY, DESCRIPTION	Hope Creek Cycle 13
Number of Bundles in Core	764
Limiting Cycle Exposure Point	EOR[1]
Cycle Exposure at Limiting Point (MWd/MT)	10704 (EOR-1102)
Reload Fuel Type	GE14
Latest Reload Batch Fraction, %	21.5%
Latest Reload Average Batch Weight % Enrichment	4.02%
Core Average Weight % Enrichment	3.64%
Core MCPR (for limiting rod pattern)	1.42
Batch Fraction for GE14	21.5%
Batch Fraction for SVEA	78.5%
[[	{3}]]
[[	{3}]]
[[	{3}]]
Power distribution methodology	Revised NEDC-32601P-A
Power distribution uncertainty	Reduced NEDC-32694P-A
Non-power distribution uncertainty	Revised NEDC-32601P-A
Calculated Safety Limit MCPR (Two Loop)	**1.06**
Calculated Safety Limit MCPR (SLO)	**1.08**

[1] End of Rated (EOR) is defined as end-of-cycle all rods out, 100% power / 100% flow and normal feedwater temperature. The actual analysis is performed prior to EOR in order to have sufficient control rod density to force some bundles near to the OLMCPR.

Table 2a

Standard Uncertainties

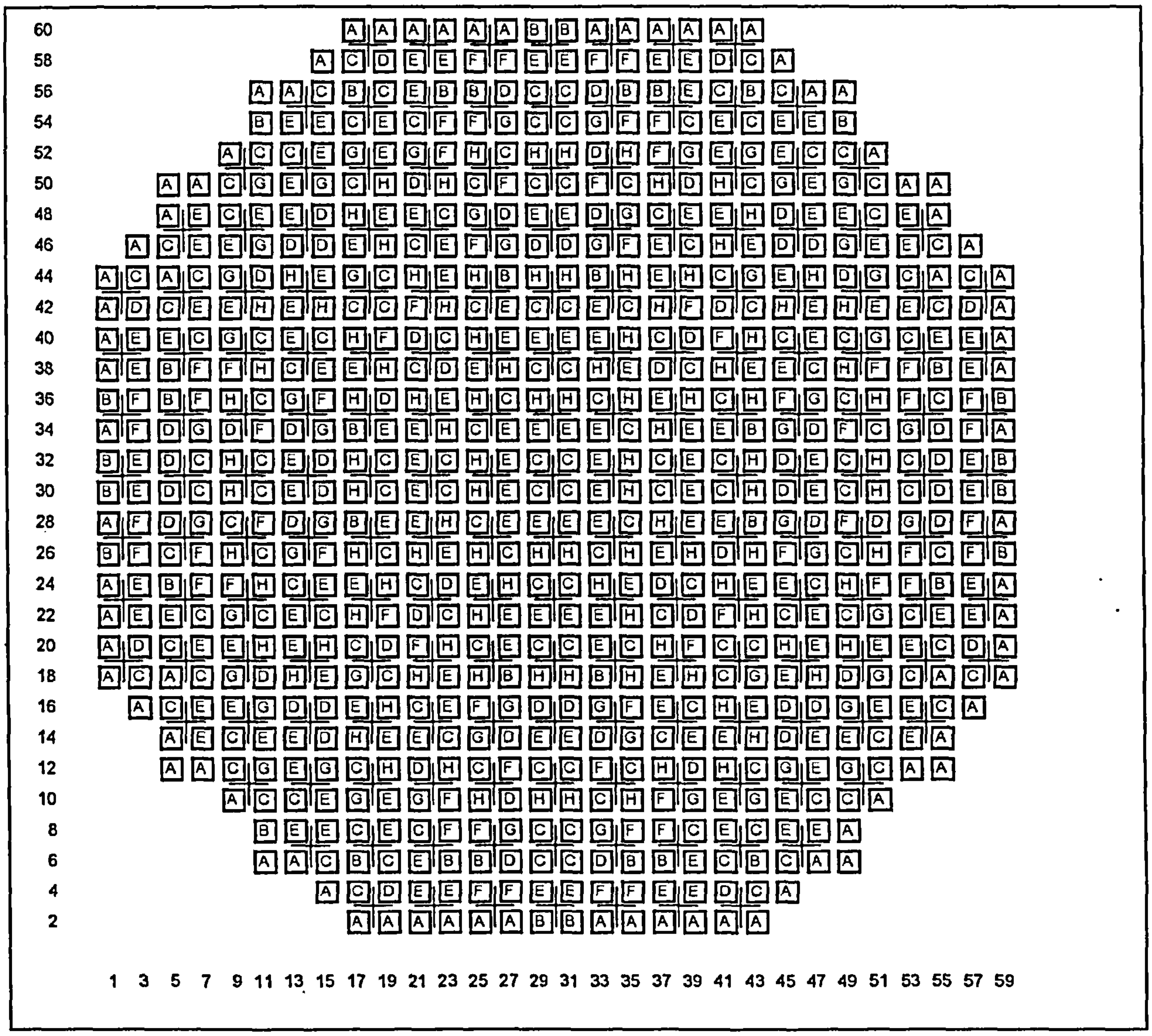
DESCRIPTION	Hope Creek Cycle 13
Non-power Distribution Uncertainties	Revised NEDC-32601P-A
Core flow rate (derived from pressure drop)	2.5 Two Loop 6.0 SLO
Individual channel flow area	[[{3}]]
Individual channel friction factor	5.0
Friction factor multiplier	[[{3}]]
Reactor pressure	0.7
Core inlet temperature	0.2
Feedwater temperature	[[{3}]]
Feedwater flow rate	[[{3}]]
Power Distribution Uncertainties	Reduced NEDC-32694P-A
GEXL R-factor	[[{3}]]
Random effective TIP reading	1.2 Two Loop 2.85 SLO
Systematic effective TIP reading	[[{3}]]
Integrated effective TIP reading	[[{3}]]
Bundle power	[[{3}]]
Effective total bundle power uncertainty	[[{3}]]

Table 2b

Exceptions to the Standard Uncertainties Used in Hope Creek Cycle 13

Reactor pressure	2.04
GEXL R-factor	[[{3}]]

Figure 1
Reference Loading Pattern – Hope Creek Cycle 13



Code	Bundle Name	Number Loaded	Cycle Loaded
A	SVEA96-P10CASB326-11GZ-568U-4WR-150-T6-2654	81	10
B	SVEA96-P10CASB326-11G4.5-568U-4WR-150-T6-2655	40	10
C	SVEA96-P10CASB360-12GZ-568U-4WR-150-T6-2656	167	11
D	SVEA96-P10CASB360-12G5.0-568U-4WR-150-T6-2657	72	11
E	SVEA96-P10CASB361-14GZ-568U-4WR-150-T6-2658	176	12
F	SVEA96-P10CASB360-12G5.5/2G2.5-568U-4WR-150-T6-2659	64	12
G	GE14-P10CNAB402-4G6.0/16G4.0-100T-150-T6-2757	56	13
H	GE14-P10CNAB402-5G6.0/14G4.0-100T-150-T6-2758	108	13