Entergy

**Entergy Nuclear Generation Co.**
Pilgrim Nuclear Power Station
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**Mike Bellamy**
Site Vice President

February 5, 2001
ENGC Ltr. 2.01.004

10 CFR 50.90

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

Docket No. 50-293
License No. DPR-35

Subject: Proposed Change to Pilgrim's Technical Specification Concerning the Safety Limit Minimum Critical Power Ratio

References:

1. GENE Topical Report NEDE-24011-P-A-14 (GESTAR II), GE Standard Application for Reactor Fuels.

2. GENE Topical Report NEDC-32601-P-A, Methodologies and Uncertainties for Safety Limit MCPR Evaluations.

3. GENE Topical Report NEDC-32694-P-A (3D-MONICORE), Power Distribution Uncertainties for Safety Limit MCPR Evaluations.

4. NRC letter dated March 11, 1999 (MFN-003-99) accepting References 1, 2, and 3.

5. Global Nuclear Fuels* letter (REK:00-161), dated October 26, 2000.
Subject: Pilgrim Cycle 14 SLMPR Information Transmittal.

Pursuant to 10 CFR 50.90, Entergy Nuclear Generation Company proposes to amend Facility Operating License DPR-35 for the Pilgrim Nuclear Power Station by modifying Technical Specifications sections 2.1.2 and 5.6.5.b.

Specifically, this application proposes to change the Safety Limit Minimum Critical Power Ratio (SLMCPR) in Technical Specification (TS) 2.1.2 from 1.08 to 1.06. The parenthetical statements after certain of the references listed in TS 5.6.5.b are changed to clarify that the analytical methods described in General Electric Nuclear Energy documents inclusive of the latest amendment or revision are used to determine core operating limits. Also, a new reference is added to T.S. 5.6.5.b.

Attachment 1 contains a description of the proposed changes, a safety evaluation of the changes, the Determination of No Significant Hazards Consideration, and an Environmental Assessment.

*Global Nuclear Fuel – Americas, LLC is a joint venture of General Electric, Toshiba, and Hitachi.

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Attachment 2 contains the current Technical Specification pages marked up with the proposed revisions.

Attachment 3 contains the proposed revised Technical Specification pages.

Attachment 4 contains a copy of Reference 4. The letter documents the acceptance of References 1,2, and 3. The Attachment includes Enclosure 1 (safety evaluation) and Enclosure 2 (technical evaluation report) of the letter.

Attachment 5 contains a copy of Reference 5 of which there are two versions, one containing non-proprietary information and the other containing proprietary information. The attachment addresses the applicability of the SLMCPR methodology and uncertainties, and verifications required.

Thus, there are two versions of this letter, one sent to the Document Control Desk from Pilgrim Station containing non-proprietary information and the other sent to the NRC Project Manager containing proprietary information. The version containing proprietary information in Attachment 5 includes an affidavit supporting the request that the information contained within double brackets in Attachment 5 be considered Global Nuclear Fuel proprietary information as described in 10 CFR 2.790(a)(4). Therefore, it is requested the information within the double brackets in the proprietary version of Attachment 5 be withheld from public disclosure.

Prompt review and approval of this amendment is requested in order to support startup from the 2001 refueling outage (RFO-13). The refueling outage is currently scheduled to commence on April 14, 2001. Therefore, in order to support Cycle 14 operation, it is requested the proposed changes be issued by April 14, 2000, to provide sufficient time to revise affected documents prior to startup from RFO-13.

Following approval of the proposed amendment, the Core Operating Limits Report and applicable operating procedures will be revised prior to start-up from RFO-13.

Please feel free to contact Mr. Douglas Ellis of my staff at (508) 830-8160 if you have questions regarding this subject.

M. Bellamy

Mike Bellamy

Commonwealth of Massachusetts)
County of Plymouth )

Then personally appeared before me, Mike Bellamy, who being duly sworn, did state that he is Pilgrim Station Site Vice President and that he is duly authorized to execute and file the submittal contained herein in the name and on behalf of Entergy Nuclear Generation Company and that the statements in said submittal are true to the best of his knowledge and belief.

My commission expires: September 20, 2002 DATE

NOTARY PUBLIC

Attachments (as stated)
DWE/slmcprTS.doc

Attachments:

1. Description and Evaluation of Proposed Technical Specification Change to Minimum Critical Power Ratio Safety Limit
2. Technical Specification Marked Up Pages
3. Technical Specification Revised Pages
4. NRC letter dated March 11, 1999 (MFN-003-99)
5. Global Nuclear Fuel letter (REK:00-161), dated October 26, 2000

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**ATTACHMENT 1**

Description and Evaluation of Proposed Technical Specifications Change to Minimum Critical Power Ratio Safety Limit

## DESCRIPTION OF PROPOSED CHANGES

Change 1 – Revise SLMCPR Value

Change the value of Safety Limit Minimum Critical Power Ratio (SLMCPR) in Section 2.1.2 from "1.08" to "1.06".

Change 2 – Clarify Reference to Analytical Methods

Revise the parenthetical statement in Section 5.6.5.b.1 from "the approved version...in the COLR" to "through the latest approved amendment at the time the reload analyses are performed as specified in the COLR".

Revise the parenthetical statements in Sections 5.6.5.b.2 and 5.6.5.b.3 from "the approved version.....in the COLR" to "through the latest revision at the time the reload analyses are performed as specified in the COLR".

Change 3 – Add New Reference

Add a new Reference as Section 5.6.5.b.4.

## REASONS FOR THE PROPOSED CHANGES

Change 1

The current required safety limit MCPR (SLMCPR) for Pilgrim Station is 1.08. Calculations performed by Global Nuclear Fuel for Pilgrim Station resulted in a minimum calculated Cycle 14 SLMCPR value of 1.03.

ENGC is proposing to operate with the more conservative SLMCPR of 1.06 to account for SLMCPR variances in fuel cycle(s) subsequent to Cycle 14.

Change 2

Technical Specification 5.6.5.b lists the analytical methods used to determine core operating limits.

The first document listed is NEDE-24011-P-A. The clarifying note states "the approved version at the time the reload analyses are performed shall be identified in the (Core Operating Limits Report)." In practice, GENE submits revisions to this document for NRC review and approval. Once approved, these amendments are not necessarily incorporated into the current version of NEDE-24011-P-A. Several amendments may exist before NEDE-24011-P-A is revised. Thus, this change clarifies that the latest amendment specified in the Core Operating Limits Report is used at the time the reload analyses (GESTAR) are performed.

The second and third documents listed are NEDC-31852P and NEDC-31312-P. The clarifying note states "the approved version at the time the reload analyses are performed shall be identified in the COLR". These documents are not reviewed by the NRC. Thus, this change clarifies that the latest revision at the time the reload analyses are performed as specified in the COLR are used at the time the reload analyses are performed.

Change 3

The fourth (new) reference, GE-NE-J1103808-08-02P, is added and the accompanying clarifying note states the latest revision at the time the reload analyses are performed as specified in the COLR is used at the time the reload analyses are performed. This document contains the LOCA evaluation for GE14 fuel that is being introduced in the core during the 2001 refueling outage (RFO-13).

## SAFETY EVALUATION

### Change 1

The Fuel Cladding Integrity Safety Limit is set such that no mechanistic 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. The uncertainties, however, 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 minimum critical power ratio (MCPR) in the limiting fuel assembly for which more the 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

Global Nuclear Fuel's calculation (Attachment 5) of the revised plant-specific SLMCPR value for Pilgrim's Cycle 14 was performed as part of the Reload Licensing Analysis for Pilgrim Cycle 14 and is based upon NRC approved methods (References 1, 2, and 3). The new Pilgrim Station SLMCPR is 1.06 until a more conservative SLMCPR becomes necessary.

Summary of Change 1.

Based on the above, it is concluded that the proposed SLMCPR value of 1.06 is appropriate for the Pilgrim Cycle 14 core.

### Change 2

The proposed amendment also contains a change to the parenthetical statement in Technical Specifications 5.6.5.b.1, 5.6.5.b.2, and 5.6.5.b.3. Currently, these sections specify that the analytical methods described in the "approved version" be used to determine the core operating limits. Specification 5.6.5.b.1 is changed to clarify that the latest version of NEDE-24011-P-A "through the latest approved amendment at the time the reload analyses are performed as specified in the COLR" is used.

Specifications 5.6.5.b.2 and 5.6.5.b.3 are changed to clarify that the latest version of the respective document (NEDC-31852P, NEDC-31312-P) "through the latest revision at the time the reload analyses are performed as specified in the COLR" is used.

Summary of Change 2.

This is an administrative change and has no impact on plant safety.

Summary of Change 3

This is an administrative change that documents the application of the LOCA methodology referenced in Section 5.6.5.b.1 for the introduction of new fuel (GE14) during RFO-13 (GE-NE-J1103808-08-02P).

Summary of Bases Changes.

The changes to Bases section 2.0 (Safety Limits) provide references for: (1) ranges of validity of GEXL correlation for GE11 and GE14 fuel types specified in References 5 and 6; and, (2) uncertainties in the calculation of the Safety Limit MCPR. Moreover, the change in references, from Reference 1 to References 3 and 4, is due to the use of the revised uncertainty methodology using the uncertainties specified in References 3 and 4. The methodology has been reviewed and approved by the NRC (Reference 4, NRC letter dated March 11, 1999). Relative to Reference 2 (GETAB), the method and default uncertainties in GETAB were used for the SLMCPR calculation prior to this proposed change. References 3 and 4 change the uncertainty inputs used with Reference 2 in the calculation of the Safety Limit MCPR.

The changes to Bases section 3.11 (Reactor Fuel Assembly) corrects the reference to Specification 6.9.A.4 that was previously changed to Specification 5.6.5.b (COLR) in License Amendment 177 dated July 31, 1998 (Ltr. 1.98.091).

## DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS

The Code of Federal Regulations, 10 CFR 50.91, requires licensees requesting an amendment to provide an analysis, using the standards in 10 CFR 50.92, that determines whether a significant hazards consideration exists. The following analysis is provided in accordance with 10 CFR 50.91 and 10 CFR 50.92 for the proposed amendment.

1. **The proposed changes to technical specification do not involve a significant increase in the probability of an accident previously evaluated.**

   The proposed Safety Limit MCPR (SLMCPR), and its use to determine the Cycle 14 thermal limits, have been derived using NRC approved methods (References 1, 2, and 3). These methods do not change the method of operating the plant and have no effect on the probability of an accident initiating event or transient.

The basis of the SLMCPR is to ensure no mechanistic fuel damage is calculated to occur if the limit is not violated. The new SLMCPR preserves the margin to transition boiling, and the probability of fuel damage is not increased.

Therefore, the proposed changes to technical specifications do not involve an increase in the probability or consequences of an accident previously evaluated.

2. **The proposed changes to technical specifications do not create the possibility of a new or different kind of accident from any accident previously evaluated.**

   The proposed changes result only from revised methods of analysis for the Cycle 14 core reload. These methods have been reviewed and approved by the NRC, do not involve any new or unapproved method for operating the facility, and do not involve any facility modifications. No new initiating events or transients result from these changes.

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

3. **The proposed changes to technical specifications do not involve a significant reduction in a margin of safety.**

   The margin of safety as defined in the TS bases will remain the same. The new SLMCPR was derived using NRC approved methods which are in accordance with the current fuel design and licensing criteria. The SLMCPR remains high enough to ensure that greater than 99.9% of all fuel rods in the core will avoid transition boiling if the limit is not violated, thereby preserving the fuel cladding integrity.

   Therefore, the proposed changes to technical specifications do not involve a significant reduction in the margin of safety.

The proposed changes have been reviewed and recommended for approval by the Pilgrim Station Operations Review Committee and the Nuclear Safety Review and Audit Committee.

## ENVIRONMENTAL IMPACT

The proposed technical specification changes were reviewed against the criteria of 10 CFR 51.22 for environmental considerations. The proposed changes do not involve a significant hazards consideration, a significant increase in the amounts of effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Based on the foregoing, Entergy Nuclear Generation Company concludes the proposed Technical Specifications meet the criteria in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.

**ATTACHMENT 2**

Marked-up Pages of Technical Specifications

# 2.0 SAFETY LIMITS

## 2.1 Safety Limits

2.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% of rated core flow:

THERMAL POWER shall be ≤ 25% of RATED THERMAL POWER.

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

MINIMUM CRITICAL POWER RATIO shall be ≥ ~~1.08~~. 1.06

2.1.3 Whenever the reactor is in the cold shutdown ~~condition~~ with irradiated fuel in the reactor vessel, the water level shall not be less than 12 inches above the top of the normal active fuel zone.

2.1.4 Reactor steam dome pressure shall be ≤ 1325 psig at any time when irradiated fuel is present in the reactor vessel.

## 2.2 Safety Limit Violation

With any Safety Limit not met the following actions shall be met:

2.2.1 Within one hour notify the NRC Operations Center in accordance with 10CFR50.72.

2.2.2 Within two hours:

A. Restore compliance with all Safety Limits, and

B. Insert all insertable control rods.

2.2.3 The Station Director and Senior Vice President - Nuclear and the Nuclear Safety Review and Audit Committee (NSRAC) shall be notified within 24 hours.

2.2.4 A Licensee Event Report shall be prepared pursuant to 10CFR50.73. The Licensee Event Report shall be submitted to the Commission, the Operations Review Committee (ORC), the NSRAC and the Station Director and Senior Vice President - Nuclear within 30 days of the violation.

2.2.5 Critical operation of the unit shall not be resumed until authorized by the Commission.

Instead of the standard GETAB model uncertainties, revised uncertainties in accordance with References 3 and 4 were used to calculate the SLMCPR

**FUEL CLADDING INTEGRITY (2.1.1) (Cont)**

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.

**MINIMUM CRITICAL POWER RATIO (2.1.2)**

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB (2), which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) - Boiling Length (L), GEXL, correlation.

The range of validity of ~~T~~he GEXL correlation is specified in References 5 and 6. ~~valid over the range of conditions used in the tests of the data used to develop the correlation. These conditions are:~~

| | |
|---|---|
| Pressure: | 800 to 1300 psia |
| Mass Flux: | 0.1 to 1.5 Mlb/hr-ft$^2$ |
| Inlet Subcooling: | 0 to 70 Btu/lb |
| Axial Profile: | 1.5 chopped cosine |
| | 1.7 inlet peaked |
| | 1.7 outlet peaked |
| R-Factor | 0.95 to 1.20 |
| Rod Array | 9X9 GE 11 array |

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 result in damage to BWR fuel rods, the critical power at

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**MINIMUM CRITICAL POWER RATIO (2.1.2) (Cont)**

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 the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity Safety Limit calculation are given in Reference 1. References 3 and 4 ~~Reference 1~~ include~~s~~ a tabulation of the uncertainties used in the determination of the Safety Limit MCPR and of the nominal values of the parameters used in the Safety Limit MCPR statistical analysis.

**REACTOR WATER LEVEL (Shutdown Condition) (2.1.3)**

With fuel in the reactor vessel during periods when the reactor is shutdown, consideration must be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the fuel provides adequate margin. This level will be continuously monitored.

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**REACTOR STEAM DOME PRESSURE (2.1.4)**

The Safety Limit for the reactor steam dome pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME Boiler and Pressure Vessel Code (1965 Edition, including the January 1966 Addendum), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The Safety Limit of 1325 psig, as measured by the reactor steam dome pressure indicator, is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The reactor coolant system is designed to the USAS Nuclear Power Piping Code, Section B31.1.0 for the reactor recirculation piping, which permits a maximum pressure transient of 120% of design pressures of 1148 psig at 562°F for suction piping and 1241 psig at 562°F for discharge piping. The pressure Safety Limit is selected to be the lowest transient overpressure allowed by the applicable codes.

**REFERENCES**


1) "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A ~~(Applicable Amendment specified in the CORE OPERATING LIMITS REPORT).~~ (through the latest approved amendment at the time the reload analyses are performed as specified in the CORE OPERATING LIMITS REPORT).

2) General Electric Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, General Electric Co. BWR Systems Department, January 1977, NEDE-10958-PA and NEDO-10958-A.

3) "Methodology & Uncertainties for SLMCPR Evaluations," NEDC-32601-P-A (August 1999).

4) "Power Distribution Uncertainties for Safety Limit MCPR Evaluations," NEDC-32694-P-A (August 1999).

5) "GE11 Compliance with Amendment 22 of GESTAR II," NEDE-31917P (April 1991).

6) GE 14 Compliance with Amendment 22 of GESTAR II," NEDC-32868P (December 1998).

## 3.11 REACTOR FUEL ASSEMBLY

### A. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10CFR50, Appendix K.

The analytical method used to determine the APLHGR limiting values is described in the topical reports listed in Specification ~~6.9.A.4~~ 5.6.5.b.

### B. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation rate.

The analytical method used to determine the LHGR limiting value is described in the topical reports listed in Specification ~~6.9.A.4.~~ 5.6.5.b.

### C. Minimum Critical Power Ratio (MCPR)

#### Operating Limit MCPR

For any abnormal operating transient analysis with the initial condition of the reactor 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 the instrument trip settings given in Tables 3.1.1, 3.2.A and 3.2.B.

The analytical method used to determine the Operating Limit MCPR values in the CORE OPERATING LIMITS REPORT is described in the topical reports listed in Specification ~~6.9.A.4~~ 5.6.5.b. By maintaining MCPR greater than or equal to the Operating Limit MCPR, the Safety Limit MCPR specified in Specification 2.1.2 is maintained in the event of the most limiting abnormal operating transient.

### D. Power/Flow Relationship During Power Operation

The power/flow curve is the locus of core thermal power as a function of flow from which the occurrence of abnormal operating transients will yield results within defined plant safety limits. Each transient and postulated accident applicable to operation of the plant was analyzed along the power/flow line. The analysis justifies the operating envelope bounded by the power/flow curve as long as other operating limits are satisfied. Operation under the power/flow line is designed to enable the direct ascension to full power within the design basis for the plant.

## 5.6 Reporting Requirements

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a by May 15th of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and process control procedures and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 Core Operating Limits Report (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Table 3.1.1 – APRM High Flux trip level setting
2. Table 3.2.C – APRM Upscale trip level setting
3. 3.11.A – Average Planar Linear Heat Generation Rate (APLHGR)
4. 3.11.B – Linear Heat Generation Rate (LHGR)
5. 3.11.C – Minimum Critical Power Ratio (MCPR)
6. 3.11.D – Power/Flow Relationship During Power Operation
7. 4.2 – Reactor Core

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," ~~(the approved version at the time the reload analyses are performed shall be identified in the COLR).~~

through the (latest approved amendment at the time the reload analyses are performed as specified in the COLR).

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## 5.6 Reporting Requirements

5.6.5 (continued)

2. NEDC-31852P, "Pilgrim Nuclear Power Station SAFER/GESTR-LOCA Loss of Coolant Accident Analysis", dated September, 1990 ~~(the approved version at the time the reload analyses are performed shall be identified~~ in the COLR), and

3. NEDC-31312-P, "ARTS Improvement Program Analyses for Pilgrim Nuclear Power Station", dated September 1987, ~~(the approved version at the time the reload analyses are performed shall be identified~~ in the COLR), And

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

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(through the latest revision at the time the reload analyses are performed as specified

4. GE-NE-J1103808-08-02P, "Pilgrim Nuclear Power Station ECCS-LOCA Evaluation," dated January, 2001 (through the latest revision at the time the reload analyses are performed as specified in the COLR).

# ATTACHMENT 3

Revised Pages of Technical Specifications

## 2.0 SAFETY LIMITS

2.1 Safety Limits

2.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% of rated core flow:

THERMAL POWER shall be $\leq$ 25% of RATED THERMAL POWER.

2.1.2 With the reactor steam dome pressure $\geq$ 785 psig and core flow $\geq$ 10% of rated core flow:

MINIMUM CRITICAL POWER RATIO shall be $\geq$ 1.06.

2.1.3 Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 12 inches above the top of the normal active fuel zone.

2.1.4 Reactor steam dome pressure shall be $\leq$ 1325 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 Safety Limit Violation

With any Safety Limit not met the following actions shall be met:

2.2.1 Within one hour notify the NRC Operations Center in accordance with 10CFR50.72.

2.2.2 Within two hours:

A. Restore compliance with all Safety Limits, and

B. Insert all insertable control rods.

2.2.3 The Station Director and Senior Vice President - Nuclear and the Nuclear Safety Review and Audit Committee (NSRAC) shall be notified within 24 hours.

2.2.4 A Licensee Event Report shall be prepared pursuant to 10CFR50.73. The Licensee Event Report shall be submitted to the Commission, the Operations Review Committee (ORC), the NSRAC and the Station Director and Senior Vice President - Nuclear within 30 days of the violation.

2.2.5 Critical operation of the unit shall not be resumed until authorized by the Commission.

FUEL CLADDING INTEGRITY (2.1.1) (Cont)

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.

MINIMUM CRITICAL POWER RATIO (2.1.2)

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB (2), which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. Instead of the standard GETAB model uncertainties, revised uncertainties in accordance with references 3 and 4 were used to calculate the SLMCPR. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) - Boiling Length (L), GEXL, correlation. The range of validity of the GEXL correlation is specified in References 5 and 6.

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 result in damage to BWR fuel rods, the critical power at

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| MINIMUM CRITICAL POWER RATIO (2.1.2) (Cont) | 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 the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity Safety Limit calculation are given in Reference 1. References 3 and 4 include a tabulation of the uncertainties used in the determination of the Safety Limit MCPR and of the nominal values of the parameters used in the Safety Limit MCPR statistical analysis. |
| REACTOR WATER LEVEL (Shutdown Condition) (2.1.3) | With fuel in the reactor vessel during periods when the reactor is shutdown, consideration must be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the fuel provides adequate margin. This level will be continuously monitored. |

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**BASES:**

## 2.0 SAFETY LIMITS (Cont)

REACTOR STEAM DOME PRESSURE (2.1.4)

The Safety Limit for the reactor steam dome pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME Boiler and Pressure Vessel Code (1965 Edition, including the January 1966 Addendum), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The Safety Limit of 1325 psig, as measured by the reactor steam dome pressure indicator, is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The reactor coolant system is designed to the USAS Nuclear Power Piping Code, Section B31.1.0 for the reactor recirculation piping, which permits a maximum pressure transient of 120% of design pressures of 1148 psig at 562°F for suction piping and 1241 psig at 562°F for discharge piping. The pressure Safety Limit is selected to be the lowest transient overpressure allowed by the applicable codes.

REFERENCES

1) "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (through the latest approved amendment at the time the reload analyses are performed as specified in the CORE OPERATING LIMITS REPORT).

2) General Electric Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, General Electric Co. BWR Systems Department, January 1977, NEDE-10958-PA and NEDO-10958-A.

3) "Methodology & Uncertainties for SLMCPR Evaluations," NEDC-32601-P-A (August 1999).

4) "Power Distribution Uncertainties for Safety Limit MCPR Evaluations," NEDC-32694-P-A (August 1999).

5) "GE 11 Compliance with Amendment 22 of GESTAR II," NEDE-31917P (April 1991).

6) "GE 14 Compliance with Amendment 22 of GESTAR II," NEDC-32868P (December 1998).

# BASES

## 3.11 REACTOR FUEL ASSEMBLY

### A. Average Planar Linear Heat Generation Rate (APLGHR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10CFR50, Appendix K.

The analytical method used to determine the APLHGR limiting values is described in the topical reports listed in Specification 5.6.5.b.

### B. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation rate.

The analytical method used to determine the LHGR limiting value is described in the topical reports listed in Specifications 5.6.5.b.

### C. Minimum Critical Power Ratio (MCPR)

Operating Limit MCPR

For any abnormal operating transient analysis with the initial condition of the reactor 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 the instrument trip settings given in Tables 3.1.1, 3.2.A and 3.2.B.

The analytical method used to determine the Operating Limit MCPR values in the CORE OPERATING LIMITS REPORT is described in the topical reports listed in Specification 5.6.5.b. By maintaining MCPR greater than or equal to the Operating Limit MCPR, the Safety Limit MCPR specified in Specification 2.1.2 is maintained in the event of the most limiting abnormal operating transient.

### D. Power/Flow Relationship During Power Operation

The power/flow curve is the locus of core thermal power as a function of flow from which the occurrence of abnormal operating transients will yield results within defined plant safety limits. Each transient and postulated accident applicable to operation of the plant was analyzed along the power/flow line. The analysis justifies the operating envelope bounded by the power/flow curve as long as other operating limits are satisfied. Operation under the power/flow line is designed to enable the direct ascension to full power within the design basis for the plant.

5.6 Reporting Requirements

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a by May 15th of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and process control procedures and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 Core Operating Limits Report (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Table 3.1.1 – APRM High Flux trip level setting
2. Table 3.2.C –APRM Upscale trip level setting
3. 3.11.A – Average Planar Linear Heat Generation Rate (APLHGR)
4. 3.11.B – Linear Heat Generation Rate (LHGR)
5. 3.11.C –Minimum Critical Power Ratio (MCPR)
6. 3.11.D – Power/Flow Relationship During Power Operation
7. 4.2 – Reactor Core

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (through the latest approved amendment at the time the reload analyses are performed as specified in the COLR).

5.6.5 (continued)

2. NEDC-31852P, "Pilgrim Nuclear Power Station SAFER/GESTR-LOCA Loss of Coolant Accident Analysis", dated September, 1990 (through the latest revision at the time the reload analyses are performed as specified in the COLR), and

3. NEDC-31312-P, "ARTS Improvement Program Analyses for Pilgrim Nuclear Power Station", dated September 1987 (through the latest revision at the time the reload analyses are performed as specified in the COLR), and

4. GE-NE-J1103808-08-02P, "Pilgrim Nuclear Power Station ECCS-LOCA Evaluation", dated January 2001 (through the latest revision at the time the reload analyses are performed as specified in the COLR).

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

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

**ATTACHMENT 4**

NRC letter dated March 11, 1999 (MFN-003-99)

Frank Akstulewicz (NRC) to Glen Watford (General Electric)

Subject: Acceptance of Referencing of Licensing Topical Reports NEDC-32601P, Methodology and Uncertainties for Safety Limit MCPR Evaluations; NEDC-32694P, Power Distribution and Uncertainties for Safety Limit MCPR Evaluation; and Amendment 25 to NEDE-24011-P-A on Cycle-Specific Safety Limit MCPR (TAC NOS. M97490, M99069, M97491)

UNITED STATES
**NUCLEAR REGULATORY COMMISSION**
WASHINGTON, D.C. 20555-0001

March 11, 1999

*MFN-003-99*

Mr. Glen A. Watford, Manager
General Electric Company
P.O. Box 780
Wilmington, NC 28402

SUBJECT: 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)

Dear Mr. Watford:

The staff has reviewed the subject reports submitted by GE Nuclear Energy (GENE) by letters dated December 13, 1996, for NEDC-32601P; June 10, 1997, for NEDC-32694P; and December 13, 1996, for Amendment 25 to NEDE-24011P. These submittals provide (1) the description of the procedures used to account for the reload-specific core design and operation in determining the cycle-specific safety limit minimum critical power ratio (SLMCPR) in NEDC-32601P; (2) the power distribution uncertainty for the new GE 3D-MONICORE core surveillance system in NEDC-32694P; and (3) the methodology and uncertainties required for the implementation of cycle-specific SLMCPR in Amendment 25 to NEDE-24011-P-A. The staff has found the subject reports to be acceptable for referencing in license applications to the extent specified and under the limitations stated in the GENE letter dated March 1, 1999, the enclosed report, and the U. S. Nuclear Regulatory Commission (NRC) technical evaluation. The evaluation defines the basis for acceptance of the report.

The staff will not repeat its review of the matters described in the GENE Topical Reports NEDC-32601P, NEDC-32694, and Amendment 25 to NEDE-24011-P-A and found acceptable when this letter request appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved. NRC acceptance applies only to the matters described in the GENE Topical Reports NEDC-32601P, NEDC-32694P, and Amendment 25 to NEDE-24011-P-A. In accordance with procedures established in NUREG-0390, the NRC requests that GE publish accepted versions of the submittals, proprietary and non-proprietary, within 3 months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract and an -A (designating accepted) following the report identification symbol.

If the NRC's criteria or regulations change so that its conclusions that the submittal is acceptable are invalidated, GE and/or the applicant referencing the submittal will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the submittal without revision of the respective documentation.

Sincerely,

Frank Akstulewicz, Acting Chief
Generic Issues and Environmental Project Branch
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Enclosures:
NEDC-32601P, NEDC-32694P, and Amendment 25 to NEDE-24011-P-A Evaluation

UNITED STATES
**NUCLEAR REGULATORY COMMISSION**
WASHINGTON, D.C. 20555-0001

ENCLOSURE 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATING TO GENERAL ELECTRIC LICENSING TOPICAL REPORTS
NEDC-32601P, NEDC-32694P, AND AMENDMENT 15 to NEDE-24011-P-A

## 1. INTRODUCTION

By letters dated December 13, 1996, June 10, 1997, and December 13, 1996, from R. J. Reda (GE) to USNRC, General Electric Nuclear Energy (GENE) submitted licensing topical reports: NEDC-32601P, 'Methodology and Uncertainties for Safety Limit MCPR Evaluation" (Reference 1); NEDC-32694P, "Power Distribution Uncertainties for Safety Limit MCPR Evaluation," (Reference 2); and Amendment 25 to NEDE-24011-P-A, Proposed Amendment 25 to GE Licensing Topical Report NEDE-24011-P-A (GESTAR II) on Cycle-Specific Safety Limit MCPR," (Reference 3), respectively. The purpose of the submittal is (1) for NEDC32601P to update values of the CPR correlation uncertainties contained in NEDE-10958-P-A (GETAB, Reference 4) based on the most recent analysis of available data; (2) for NEDC32694P to update values of the power distribution uncertainties contained in NEDE-31152P, Revision 5 based on the most recent analysis of available data, and (3) for Amendment 25 to NEDE-24011-P-A to provide for cycle-specific Safety Limit Minimum Critical Power Ratios (MCPRs).

The NRC staff was assisted in this review by its consultant, Brookhaven National Laboratory (BNL). The NRC staff's evaluation includes those three topical reports and the responses to staff's Request for Additional Information (RAI) dated January 8,1998 (GA W-98-002, MFN-004-98, Reference 5), January 9,1998 (GAW-98-003, MFN-005-98, Reference 6), January 28, 1998 (GAW-98-005, MFN-008-98, Reference 7), April 17, 1998 (GAW-98-009, Reference 8), and July 29, 1998 (GAW-98-012, MFN-017-98, Reference 9). The staff adopted the findings recommended in our consultant's Technical Evaluation Report (Enclosure 2).

## 2 EVALUATION

This review includes three topical reports involving the Safety Limit Minimum Critical Power Ratio (SLMC PR) methodology and input uncertainties described in NEDC-32601 P, the methodology for constructing the bounding statepoint power distribution described in NEDC-32694P, and the overall procedures for determining the cycle-specific SLMCPR described in Amendment 25 to GESTAR II. The details of the evaluation are provided in Enclosure 2.

### 2.1 Methodology and Uncertainties for Safety Limit MCPR Evaluation (NEDC-32601P)

The topical report, NEDC-32601 P, provides an update to the Safety Limit MCPR methodology and inputs to be used in the evaluation of the Safety Limit MCPR for BWRs (GETAB, Reference 4) including plant surveillance measurement uncertainties and local R-Factor uncertainties. The plant surveillance component uncertainties include the reactor pressure, feedwater temperature and flow, core inlet temperature and flow, and channel flow area and friction factors. The plant surveillance uncertainty revisions are based on current BWR practice, and are generally evaluated using the error methodology (Reference 10). The

R-Factor provides the critical power dependence on the local pin power distribution (References 4 and 11) in the GEXL correlation. The R-factor uncertainty analysis includes an allowance for power peaking modeling uncertainty, manufacturing uncertainty and channel bow uncertainty.

Based on the review of the NEDC-32601 P topical report and the responses to the staff's request for additional information (RAI) (References 5, 8, and 9), we find the SLMCPR methodology and associated uncertainties to be acceptable, however, actions should be taken as follows:

(1) The TGBLA fuel rod power calculational uncertainty should be verified when applied to fuel designs not included in the benchmark comparisions of Table 3.1 of Reference 1, since changes in fuel design can have a significant effect on calculation accuracy.

(2) The effect of the correlation of rod power calculation uncertainties should be reevaluated to insure the accuracy of the R-Factor uncertainty when the methodology is applied to a new fuel lattice.

(3) In view of the importance of MIP criterion and its potential sensitivity to changes in fuel bundle designs, core loading and operating strategies, the MIP criterion should be reviewed periodically as part of the procedural review process to insure that the specific value recommended in NEDC-32601p is applicable to future designs and operating strategies.

2.2 Power Distribution Uncertainties for Safety Limit MCPR Evaluations (NEDC-32694P)

The power distribution uncertainty topical report NEDC-32694P provides a description of the 3D-MONICORE core surveillance system and the determination of the associated bundle power uncertainty for use in SLMCPR calculation. The 3D-MONCORE system uses three-dimensional coarse-mesh diffusion theory methods, together with models for interfacing with the incore TIP and LPRM instrumentation, to determine the detailed core statepoint. The uncertainty in the 3D-MON ICORE prediction of bundle power was determined by comparisions of measured and calculated TIP integrals and gamma scanned bundle powers.

Based on the review of Reference 2 and the responses to the staff's RAI (References 6 and 8) we have found that the 3D-MONICORE power distribution uncertainties are acceptable for determining the SLMCPR, MAPLHGR and LHGR core limits, however, the 3D-MONICORE bundle power calculational uncertainty should be verified when applied to fuel and core designs not included in the benchmark comparisons of Tables *3.1* and 3.2 of Reference 2.

2.3 Amendment 25 to GE Licensing Topical Report NEDE-24011-P-A (GESTAR II) on Cycle - Specific Safety Limit MCPR

Amendment 25 to GESTAR II provides the methodology and uncertainties required for the implementation of cycle-specific Safety Limit MCPRs that replace the former generic, bounding SLMCPR. General procedures are given describing the analysis to be used in determining the cycle-specific SLMCPR. These procedures require that the analysis be performed for the specific fuel bundle design and core loading used in the cycle reload design.

Based on the review of References 3 and 7, we have found that the proposed methodology to be acceptable for performing cycle-specific SLMCPR analyses.

## 3 CONCLUSION

Based on our review of Topical Reports NEDC-32601P, NEDC-32694P, and Amendment 25 to NEDE-2401 1-P-A (GESTAR II), the staff concludes that the input plant system uncertainties, the power distribution uncertainties associated with the application of 3D-MON ICORE, and the proposed cycle-specific determination of the SLMCPR are acceptable. In letter dated PM TO SUPPLY, GENE has stated that they will take the following actions whenever a new fuel design is introduced.

(1) The TGBLA fuel rod power calculational uncertainty should be verified when applied to fuel designs not included in the benchmark comparisions of Table 3.1 of NEDC-32601P, since changes in fuel design can have a significant effect on calculation accuracy.

(2) The effect of the correlation of rod power calculation uncertainties should be reevaluated to insure the accuracy of R-Factor uncertainty when the methodology is applied to a new fuel lattice.

(3) In view of the importance of MIP criterion and its potential sensitivity to changes in fuel bundle designs, core loading and operating strategies, the MIP criterion should be reviewed periodically as part of the procedural review process to insure that the specific value recommended in NEDC-32601 P is applicable to future designs and operating strategies.

(4) The 3D-MON ICORE bundle power calculational uncertainty should be verified when applied to fuel and core designs not included in the benchmark comparisions in Tables 3.1 and 3.2 of NEDC-32694P.

## 4 REFERENCES


1. GE Letter RJR-96-139 MFN-185-96 dated December 13, 1996 from R. J. Reda to USNRC transmitting a topical report, NEDC-32601 P, "Methodology and Uncertainties for Safety Limit MCPR Evaluations," December 1996.

2. GE Letter RJR-97-074 MFN-022-97 dated June 10, 1997 form R. J. Reda to USNRC transmitting a topical report, NEDC-32694P, "Power Distribution Uncertainties for Safety Limit MCPR Evaluations," January 1997.

3. GE Letter RJR-96-133 MFN-179-96 from R. J. Reda to USNRC, "Proposed Amendment 25 to GE Licensing Topical Report NEDE-2401 1-P-A (GESTAR II) on Cycle Specific Safety Limit MCPR," December 13, 1996.

4. General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, NEDE-10958-PA, January 1977.

5. GE Letter GAW-98-002 MFN-004-98 from Glen A. Watford to USNRC, Responses to Request for Additional Information for GE Topical Report NEDC-32601 P, Methodology and Uncertainties for Safety Limit MCPR Evaluations, January 8,1998.

6. GE Letter GAW-98-003 MFN-005-98 from Glen A. Watford to USNRC, Responses to Request for Additional Information for GE Topical Report NEDC-32694P, Power Distribution Uncertainties for Safety Limit MCPR Evaluations, January 9,1998.

7. GE Letter GAW-98-005 MFN-008-98 from G. A. Waterford to USNRC, Responses to Request for Additional Information for Amendment 25 to GE Topical Report NEDE-24011-*P-A* (GESTAR II) on Cycle-Specific Safety Limit MCPR (TAC No. M97491), January 28, 1998.

8. GE Letter GAW-98-009 MFN-014-98 from Glen A. Watford to USNRC, Responses to NRC Request for Additional Information associated with SLMCPR Methodology and Uncertainty Topical Reports NEDC-32601P and NEDC-32694P, April 17, 1998.

8. GE Letter GAW-98-012 MFN-017-98 from Glen A. Watford to USNRC, Additional Information Associated with SLMCPR Methodology and Uncertainty Topical Report NEDC-32601P, July 29, 1998.

10. "Recommended Practice - Setpoint Methodologies," Part II, ISA-RP 67.04, Instrument Society of America, September 1994.

11. NEDC-32505, Revision 1, "R-Factor Calculation Method for GEII, GEI2, and GEI3 Fuel," June 1997.

ENCLOSURE 2

# TECHNICAL EVALUATION REPORT

| | | |
|---|---|---|
| **Report Titles:** | 1) | Power Distribution Uncertainties for Safety Limit MCPR Evaluations |
| | 2) | Methodology and Uncertainties for Safety Limit MCPR |
| | 3) | Amendment 25 to GE Licensing Topical Report NEDE-24011-P-A (GESTAR) on Cycle-Specific Safety Limit MCPR |
| **Report Numbers:** | 1) | NEDC-32694P |
| | 2) | NEDC-32601P |
| | 3) | NEDE-24011-P-A |
| **Report Dates:** | 1) | January 1997 |
| | 2) | December 1996 |
| | 3) | December 1996 |

**Originating Organization:** General Electric Company

## 1.0 INTRODUCTION

In Reference-1, the General Electric Company (GE) has submitted the proposed GESTAR modifications for including the cycle-specific Safety Limit MCPR (SLMCPR), replacing the generic bounding SLMCPR methodology included in GESTAR, for NRC review and approval. These modifications provide the licensing methods to be used in determining the cycle-specific SLMCPR for each plant reload. In support of these modifications, GE has submitted the two additional licensing topical reports: (1) NEDC-32601P (Reference-2), "Methodology and Uncertainties for Safety Limit MCPR," and (2) NEDC-32694P (Reference-3), "Power Distribution Uncertainties for Safety Limit MCPR Evaluations." The NEDC-32601P Topical Report describes the procedures used to account for the reload-specific core design and operation in determining the cycle-specific

SLMCPR. In this topical report, the values of the plant monitoring uncertainties and local R-Factor uncertainty used in the SLMCPR determination are also reviewed and updated to reflect current recommended practices.

The NEDC-32694P Topical Report provides the power distribution uncertainty for the new GE 3D-MONICORE core surveillance system. The 3D-MONICORE power distribution uncertainties are determined based on an uncertainty propagation analysis and on comparisons with benchmark measurements. The resulting 3D-MONICORE uncertainties are used in the determination of the SLMCPR for the plants employing the 3D-MONJCORE system.

The review of the GE core monitoring and SLMCPR analysis was included in the NRC vendor inspections (Nos. 99900003/95-01 and 99900003/96-01) at the General Electric Nuclear Energy Facility in Wilmington, NC during the weeks of August 14 through September 1, 1995 and May 6 through May 10, 1996. Several important concerns were identified during these reviews including: (1) the level of conservatism in the operating state assumed in the cycle-specific determination of the SLMCPR and (2) the effect of the 3D-MONICORE uncertainties on the SLMCPR uncertainty analysis. These concerns are addressed in the safety limit methodology and uncertainty analysis Topical Report NEDC-32694P and the power distribution uncertainty Topical Report NEDC32694P, respectively.

The purpose of this review was to evaluate these methodology modifications and updates to insure that the changes in the monitoring uncertainties are acceptable and that adequate margin is included in the determination of the SLMCPR. The methodology changes are summarized in Section 2, and the evaluation of the important technical issues raised during this review is presented in Section 3. The Technical Position is given in Section 4.

## 2.0 SUMMARY OF THE REVISED SLMCPR METHODOLOGY

### 2.1 Power Distribution Uncertainties for Safety Limit MCPR Evaluations (NEDC-32694P)

The power distribution uncertainty Topical Report NEDC-32694P provides (1) a description of the 3D-MONICORE core surveillance system and (2) the determination ofthe associated bundle power uncertainty for use in SLMCPR calculations. The 3D-MONICORE system uses three-dimensional coarse-mesh diffusion theory methods, together with models for interfacing with the incore TIP and LPRM instrumentation, to determine the detailed core statepoint. The physics methods used in 3D-MONICORE are identical to those used in BWR fuel design calculations and core management evaluations. 3D-MONICORE solves a modified diffusion theory equation in order to allow the local normalization of the power distribution to the TIP and LPRM incore measurements. However, prior to this normalization, the TIP/LPRM measurements are compared to the instrument responses predicted by 3D-MONICORE. If these comparisons indicate that certain measurements are suspect, this data is rejected and the normalization is performed with the remaining reliable TIP/LPRM measurements.

The uncertainty in the 3D-MONICORE prediction of bundle power was determined by comparisons of measured and calculated TIP integrals and gamma scanned bundle powers. These comparisons included a wide range of fuel enrichments, poison loadings and operating conditions. The increased uncertainty between TIP measurements was determined by comparing LPRM-updated TIPs and TIP measurements taken immediately following the LPRM update. The uncertainty analysis also accounts for TIP and LPRM failures (i.e., measurement rejection). The NEDC-32694P uncertainty analysis indicates that the 3D-MONICORE power distribution uncertainty is less than the value presently used in the GETAB SLMCPR determination.

## 2.2 Methodology and Uncertainties for Safety Limit MCPR (NEDC-32601P)

The NEDC-32601P Topical Report documents the latest updates to the GETAB (Reference-4) (1) plant surveillance measurement uncertainties, (2) local R-Factor uncertainties and (3) SLMCPR methodology. The plant surveillance component uncertainties include the reactor pressure, feedwater temperature and flow, core inlet temperature and flow, and channel flow area and friction factors. The plant surveillance uncertainty revisions are based on current BWR practice, and are generally evaluated using the error methodology of Reference-5. The uncertainty analysis accounts for the overall instrument channel accuracy, drift, calibration, process uncertainty, and plant environmental effects. In most cases, a simple sum-of-the-squares combination of the contributing uncertainties is employed, however, the uncertainty in the inlet subcooling (i.e., core inlet temperature) is determined using the process computer heat balance to propagate the uncertainties.

In most cases, the reevaluation of the plant surveillance uncertainties concluded that the presently accepted GETAB uncertainty values are conservative. A detailed analysis is provided to support the revised values in the cases where the reevaluation results in a reduction in the component uncertainty.

In the GEXL correlation, the R-Factor provides the critical power dependence on the local pin power distribution (References 4 and 6). The R-Factor uncertainty analysis includes an allowance for power peaking modeling uncertainty, manufacturing uncertainty and channel bow uncertainty. The TGBLA (Reference-7) power peaking modeling uncertainty is determined by comparisons of TGBLA with MCNP (Reference-8) and quarter-core benchmark calculations for a range of BWR fuel bundle and core designs. The power peaking uncertainty determined by this analysis was confirmed with gamma scan measurements taken following Cycle-8 of the Duane Arnold Plant (Reference-9).

The uncertainty in the power peaking resulting from channel bow is determined using the procedures of Reference- 10, and the uncertainty introduced by the manufacturing process is baseci on estimated fuel enrichment and density measurements. The R-Factor uncertainty is determined by propagating the resulting local power peaking uncertainty using the R-Factor dependence on peaking factor.

The revised methodology includes updates to the calculation process used to determine the SLMCPR. The operating core statepoint is determined using the PANACEA (Reference-7) 3D-simulator program. The statepoint information used in the SLMCPR calculation includes the channel flows, bundle powers, local void fraction and the TIP detector responses. In addition, the bundle and exposure dependent R-Factors are obtained from the PANACEA statepoint data and used to determine the critical power. The SLMCPR is determined by randomizing the statepoint surveillance input and correlation data to determine the MCPR margin required to insure that 99.9 % of the rods avoid boiling transition.

The SLMCPR is sensitive to the assumed statepoint radial power distribution. In the cycle-specific methodology, the power distribution is selected to provide a reasonable bound on the number of rods expected to experience boiling transition. This selection is made subject to the condition that the core is critical and within thermal limits. For current BWR reload designs the limiting radial power distribution includes a centrally located high powered region which is either circular or annular in shape. Control rod patterns which provide these limiting power distributions are described and recommended. In order to quantify the severity of power distributions with respect to the number of rods in boiling transition a core weighting parameter is defined. The frequency distribution of this parameter is used to compare and select the limiting power distribution.

The determination of the SLMCPR using the revised methodology and input uncertainties is compared to the presently accepted GETAB methodology for several plants. For the cases evaluated, the effect of the changes in methodology and uncertainties is small ~.01 ΔSLMCPR.

### 2.3 GESTAR II Amendment 25 on Cycle-Specific Safety Limit MCPR (NEDE-24011-P-A)

Amendment 25 to GESTAR II provides the methodology and uncertainties required for the implementation of the cycle-specific Safety Limit MCPR. A set of general procedures are given describing the analysis to be used in determining the cycle-specific SLMCPR. These procedures require that the analysis be performed for the specific fuel bundle design and core loading used in the cycle reload design. The core radial power distribution must represent a reasonable bound on the number of fuel bundles at or near thermal limits, and the fuel assembly local power distribution must be based on the actual bundle design. The cycle-specific analysis is performed at multiple exposure points throughout the cycle, and either the most limiting or an exposure-dependent SLMCPR is used in determining the Operating Limit MCPR (OLMCPR). The cycle-specific procedures require that the SLMCPR be recalculated or' reconfirmed for each plant operating cycle.

In the reload process, the final core loading plan is evaluated relative to the reference design criteria including the OLMCPR. If the cycle-specific determination results in an increased SLMCPR, the final core loading plan may fail to satisfy the specified acceptance criteria. In this case, calculations of the sensitivity of the OLMCPR to changes in the SLMCPR are used to determine the acceptability of the calculated cycle-specific SLMCPR.

While Amendment 25 provides the overall procedures for determining the cycle-specific SLMCPR, the detailed SLMCPR methodology and input uncertainties are described in NEDC-32601P and the methodology for constructing the bounding statepoint power distribution is described in NEDC-32694P.

## 3.0 SUMMARY OF THE TECHNICAL EVALUATION

The GE Topical Reports NEDC-32694P, NEDC-32601P and Amendment 25 to NEDE-24011-P-1 (GESTAR II) provide the basis for the cycle-specific determination of the SLMCPR, input plant system uncertainties and the power distribution uncertainties associated with the application of 3D-MONICORE. The review of the GE methodology focused on: (1) the assumptions made in the cycle-specific SLMCPR methodology and the changes relative to the presently approved generic SLMCPR approach and (2) the basis for the changes in the SLMCPR uncertainty values. As a result of the review of the methodology, several important technical issues were identified which required additional information and clarification from GE. This information was requested in References-11 and 12 and was provided in the GE responses included in References 13-16. This evaluation is based on the material presented in the topical reports (References 1-3) and in References 13-19. The evaluation of the major issues raised during this review are summarized in the following.

### 3.1 Power Distribution Uncertainties for Safety Limit MCPR Evaluations (NEDC-32694P)

The 3D-MONICORE system is used to perform the steady-state on-line core performance evaluation. The 3D-MONLCORE models are based on accepted BWR calculational methods. The neutronics model is essentially the same as that described in Reference-7 and the thermal-hydraulics model is the same as presently used in the P-1 Process Computer Analysis (Reference- 13, Response 11.4).

The 3D-MONICORE power distribution uncertainties are required for determining the SLMCPR, LHGR and MAPLHGR limits. The (axially integrated) bundle power uncertainty is required for the SLMCPR and the nodal power uncertainty is required for detenmining MAPLHGR and LHGR. The radial bundle power uncertainty is considered to be a statistical combination of: (1) the uncertainty in the four-bundle power associated with the TIP location and (2) the uncertainty in the allocation of the four-bundle power to the individual bundles. The four-bundle power uncertainty is determined by a comparison of the predicted and measured TIP responses, and the uncertainty in the power allocation is determined by comparisons of calculated and measured (gamma-scanned) bundle powers.

While the calculated bundle powers were determined with the "core tracking" system, rather than with 3D-MONICORE, GE has indicated (in Reference- 13, Responses 1.2 and 1.6) that the difference in these codes has no effect on the uncertainty estimates. The TIP comparisons include cores with both part length fuel rods and axially zoned gadolinium, and all current fuel product lines except for GE13. However, in view of the similarity of the GE13, GE11, and GE12 fuel designs, this is considered acceptable (Reference- 13, Response-II.4). In addition, GE has indicated that the core follow calculations employed the same methods to process and accumulate the void-history and fuel exposure as used in the on-line core surveillance (Reference-1 3, Response-II.8). However, it is concluded that since changes in the fuel and core design can have a significant effect on the calculation accuracy, the 3D-MONICORE bundle power calculational uncertainty should be verified when applied to fuel and core designs not included in the benchmark comparisons of Tables-3.1 and 3.2 of NEDC-32694P.

The review of the calculation-to-measurement (C/M) comparisons indicated an increased uncertainty at end-of-cycle. However, the cycle-average four-bundle power uncertainty is considered acceptable since the uncertainty estimate does not take credit for the uncertainty increase due to TIP measurement uncertainty. The nodal power uncertainty is determined by a statistical combination of the 3D-MONICORE bundle power uncertainty and an accepted TIP axial power uncertainty. The TIP uncertainty is measured once per cycle to ensure that it satisfies the specified acceptance criteria.

The 3D-MONICORE system allows rejection of the TIP measurement data based on a specified acceptance criteria. During the review it was noted that the 3D-MONICORE acceptance criteria will, under certain conditions, reject good TIP measurement data. However, in Responses-I.7 and I.10 (Reference- 13), GE has indicated that the rejection of TIP data is very rare. In addition, in most cases TIP rejection is due to poor agreement between measured and calculated data and, when the acceptance criteria results in rejection of measurements which are in good agreement with the calculations, the effect on the core power distribution uncertainty is negligible.

The uncertainty methodology determines the effect of TIP and LPRM rejection and the LPRMupdate of the power distribution using comparisons of calculations and measurements. In these comparisons the recommended value for the rejection criteria parameter is used. It is noted that after ten years of operation, no correlation has been observed between the rejected TIP locations and the core locations that are difficult to calculate, such as the peripheral core locations, part-length fuel bundles and partially controlled fuel bundles. It is concluded that the TIP rejections are generally a result of erroneous measurement data rather than miscalculation. It is also noted that the TIP rejection only affects the 3D-MONICORE system and the other BWR surveillance systems use the measured TIP/LPRM data.

The process computer monitors kw/ft and LHGR as well as the SLMCPR. The uncertainty analysis for the 3D-MONICORE LHGR evaluation is provided in Response-II.5 (Reference- 14) and accounts for the effect of both the TIP and LPRM update uncertainties on the nodal power calculation.

Based on the review of the NEDC-32694P topical report and supporting documentation provided in References 13 and 14, it is concluded that the 3D-MONICORE power distribution uncertainties are acceptable for determining the SLMCPR, MAPLLIGR and LHGR core limits subject to the condition identified above (in the third paragraph of this section).

### 3.2 Methodology and Uncertainties for Safety Limit MCPR (NEDC-32601P)

#### 3.2.1 Process Computer Uncertainties

The reevaluation of the process computer uncertainties provided in the NEDC-32601 P topical report were reviewed in detail. The topical report provides a description of both the instrumentation and modeling uncertainties that are required for the SLMCPR analysis. The evaluation of the core inlet subcooling uncertainty employs the heat balance used by the process computer to relate the inlet subcooling to the available instrumentation signals. Using this relation, the inlet subcooling variance is determined by the individual component variances (e.g., feedwater flow and temperature, core flow, steam carry under fraction). While the coefficients that weight the individual uncertainty components depend on the reactor statepoint, the analysis neglects this dependence and assumes constant weighting coefficients. In Response-I. 1 (Reference-15), GE has shown using a Monte Carlo procedure that these constant weighting coefficients are conservative.

The calculation of the bundle critical power is sensitive to the channel flow area and friction factors. The two-phase friction factor is based on measurements made at the full scale ATLAS test facility covering a range of power and flow The uncertainty in the two-phase friction factor is based on the comparisons with test data. The uncertainty in the single-phase friction factor is determined by comparison of the calculations to total pressure drop measurements made at the ATLAS facility. In Response-II.6 (Reference-15), it is noted that, since the total pressure drop measurement includes both the single-phase and two-phase losses, the inferred single-phase loss coefficient is conservative.

The channel flow area is subjected to random variations due to non-uniform channel bulge and crud/corrosion buildup which result in channel-to-channel variations in flow. The SLMCPR uncertainty analysis accounts for the effect of these variations by increasing the uncertainty in the channel-to-channel friction factor multiplier (Response-I.2, Reference- 15).

### 3.2.2 R-Factor Uncertainties

The fuel rod power calculational uncertainty determines the GEXL R-Factor uncertainty and is separated into three components; modeling, manufacturing and bowing. The modeling uncertainty is determined by comparison of the TGBLA calculation to MCNP benchmark lattice calculations. The Table-3.1 TGBLAIMCNP rod power comparisons include all GE fuel designs which are currently loaded in operating BWRs (Response-II.1, Reference-iS). A range of gadolinium rods is included in the comparisons in order to simulate the effects of depleted fuel rods (Response-II.2, Reference- *15)*. The fuel rod power peaking uncertainty is determined by weighting the variance for each fuel design by the number of rods in the lattice (Response-II.3, Reference-iS). However it is concluded that since changes in the fuel lattice design can have a significant effect on the calculation accuracy, the TGBLA fuel rod power calculational uncertainty should be verified when applied to fuel designs not included in the benchmark comparisons of Table-3.1 of NEDC-32601P.

In addition to the TGBLA!MCNP comparisons, GE has evaluated the effect of void fraction uncertainty on the fuel rod power calculation (Response-II.10, Reference-15). Estimates of the lattice-average void fraction uncertainty were determined by comparison with measurement. The local void fraction uncertainty was determined by comparison with detailed subchannel calculations. The effect of the lattice-average and local void fraction uncertainties on the fuel rod power calculation was determined by sensitivity calculations and found to be negligible.

The fuel rod manufacturing uncertainty includes the effects of fuel enrichment, density and rod position uncertainty. The uncertainty in fuel enrichment and density was determined from measurements on a large number of fuel rods performed as part of manufacturing studies. The fuel rod position uncertainty was determined from a series of rod spacing measurements performed on a high bumup fuel bundle. In Responses 11.4, 11.5, and 11.9 of References 14 and *15,* GE has shown that the effects of these uncertainties are conservatively included in the R-Factor analysis. In Response-II.10 (Reference- 14), the effect of local fuel bundle exposure uncertainty on rod power is shown to be negligible. It is important to note that the power peaking uncertainty is determined using a components of uncertainty approach and then independently confirmed by a comparison with gamma scan measurements.

In the approved GETAB methodology of Reference-4, the power peaking calculation errors in neighboring fuel rods are assumed to be correlated so that each of the fuel rods has exactly the same calculational error. In the proposed methodology, the modeling errors in neighboring fuel rods are assumed to be uncorrelated. As a result, the uncertainty in the R-Factor is reduced significantly in the proposed methodology. In Response-II.13 of Reference 14 and in References 17-19, GE has evaluated this effect for the 8x8, 9x9 and 10x10 lattices and has indicated that the R-Factor uncertainty will be increased (relative to the presently approved value of Reference-4) to account for the correlation of rod power uncertainties. However, in References-18 and 19 (Table-1), it is noted that the effect of the rod-to-rod correlation has a significant dependence on the fuel lattice (e.g., 9x9 versus 10x10). Therefore, in order to insure the adequacy of the R-Factor uncertainty, the effect of the correlation of rod power calculation uncertainties should be reevaluated when the NEDC-32601P methodology is applied to a new fuel lattice.

### 3.2.3 SLMCPR Evaluation Methodology

The SLMCPR is sensitive to the "flatness" of the bundle power distribution of the initial reactor statepoint. GE has defined a MCPR Importance Parameter (MIP) to allow a quantitative assessment of the flatness of the power distribution and identify limiting statepoints for SLMCPR analysis. In Response-III.2 (Reference- 15*)*, the expression for determining MIP is derived and shown to provide a quantification of the effect of the bundle power distribution on the SLMCPR. In Reference- 15, GE provides the specific MIP criterion (Response-III.5) and the thermal limits and reactivity constraints (Response-III.6) for selecting the bundle power distribution to be used in the SLMCPR analysis.

The determination of the selected MIP criterion is based on an extensive evaluation of operating reactor statepoints. In view of the importance of this MIP criterion and its potential sensitivity to changes in fuel bundle designs, core loadings and operating strategies, there is a need to insure that the specific value recommended in NEDC-32601P is applicable to future designs and operating strategies. In response to this concern, GE has indicated that the MIP criterion will be reviewed periodically as part of the procedural review process (Response III.6, Reference-15).

In the presently approved GETAB methodology (Reference-4), the bundle power calculation error is assigned to the four bundles surrounding the TIP in a correlated manner so that each of the four bundles is perturbed simultaneously by the same amount. In the proposed methodology, the calculational error is assumed to be uncorrelated and the individual bundle powers are varied independently in the Monte Carlo uncertainty propagation. The increased variability in the proposed methodology results in a (non conservative) reduction in the SLMCPR. In Response-III.1 of Reference-14, GE has revised the NEDC-32601P methodology to allow for the correlation of the bundle power calculation modeling errors.

Based on the review of the NEDC-32601P topical report and supporting documentation provided in References 14 and 15, we find the SLMCPR methodology and associated uncertainties to be acceptable subject to the conditions identified in Sections-3.2.2 and 3.2.3.

### 3.3 GESTAR II Amendment 25 on Cycle-Specific Safety Limit MCPR (NEDE-24011-P-A)

Amendment 25 to GESTAR II provides the modifications required for performing the cycle-specific SLMCPR analysis. In the cycle-specific analysis, a search is performed to determine the initial reactor statepoint for use in the Monte Carlo statistical analysis. The purpose of this search is to determine a reactor statepoint that satisfies both (1) the operations criteria required for operating statepoints and (2) the MIP flatness criterion to insure that the statepoint provides a bounding SLMCPR. In the information provided in support of Amendment 25 (Reference-1, Corrective

Action-4, Item-3), it is noted that this search may be terminated before all criteria are satisfied. However, in Responses 2 and 3 (Reference- 16), GE has indicated that if the MIP criterion is not initially satisfied the search will be expanded, by relaxing the operations criteria, to insure that the MIP criterion is satisfied.

In the presently approved GETAB methodology, the limiting power shape is assumed to include a centrally located annular ring of high-powered fuel bundles. While the proposed cycle-specific methodology does not require the power distribution to include this high-powered annular zone, it is indicated in Response-4 (Reference- 16) that the control rods are selected so that this power shape is not precluded from the search for the bounding statepoint.

Based on the review of Amendment 25 and the supporting information provided in Reference-16, we find the proposed methodology to be acceptable for performing cycle-specific SLMCPR analyses.

## 4.0 TECHNICAL POSITION

The Topical Reports NEDC-32694P, NEDC-32601P and Amendment 25 to NEDE-24011-P-A (GESTAR II) and supporting documentation provided in References 13-16 have been reviewed in detail. Based on this review, it is concluded that the proposed cycle-specific determination of the SLMCPR, the input plant system uncertainties, and the power distribution uncertainties associated with the application of 3D-MONICORE are acceptable subject to the conditions stated in Section 3 of this evaluation and summarized in the following.

1) Since changes in the fuel and core design can have a significant effect on the calculation accuracy, the 3D-MONICORE bundle power calculational uncertainty should be verified when applied to fuel and core designs not included in the benchmark comparisons of Tables-3.1 and 3.2 of NEDC-32694P (Section 3.1).

2) Since changes in fuel design can have a significant effect on the calculation accuracy, the TGBLA fuel rod power calculational uncertainty should be verified when applied to fuel designs not included in the benchmark comparisons of Table-3.1 of NEDC-32601P (Section-3.2.2).

3) In order to insure the adequacy of the R-Factor uncertainty, the effect of the correlation of rod power calculation uncertainties should be reevaluated when the NEDC-32601P methodology is applied to a new fuel lattice (Section-3.2.2).

4) In view of the importance of the MIP criterion and its potential sensitivity to changes in fuel bundle designs, core loadings and operating strategies, the MIP criterion should be reviewed periodically as part of the procedural review process to insure that the specific value recommended in NEDC-32601P is applicable to future designs and operating strategies (Section-3.2.3).

## 5.0 REFERENCES


1. "Proposed Amendment 25 to GE Licensing Topical Report NEDE-24011-P-A (GES TAR II) on Cycle-Specific Safety Limit MCPR," RJR-96-133, Letter R. J. Reda (GE) to U.S. NRC, dated December 13, 1996.

2. "GE Licensing Topical Report, Methodology and Uncertainties for Safety Limit MCPR Evaluations," RJR-96-139, Letter, R. J. Reda (GE) to U.S. NRC, dated December 13, 1996.

3. "GE Licensing Topical Report, Power Distribution Uncertainties for Safety Limit MCPR Evaluations," RJR-97-074, Letter, R. J. Reda (GE) to U.S. NRC, dated June 10, 1997.

4. "General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application," NEDO-10958-A, January 1977.

5. "Recommended Practice - Setpoint Methodologies," Part II, ISA-PP 67.04, Instrument Society of America, September 1994.

6. "R-Factor Calculation Method for GE11, GE12, and GE13 Fuel," NEDC-32505P, November 1995.

7. "Steady State Nuclear Methods," NEDE-30130-P-A, April 1985.

8. "MCNP - A General Monte Carlo - N-Particle Transport Code, Version 4a," LA-12625, J. F. Breismeister, Ed., Los Alamos National Laboratory (1993).

9. "Gamma Scan Measurements of the Lead Test Assembly at the Duane Arnold Energy Center Following Cycle-8," NEDC-3 1569-P. April 1988.

10. "Fuel Channel Bow Assessment," GENE Report MFN086-89, Letter, J. S. Charnley (GE) to R. C. Jones (NRC), dated November 15, 1989.

11. "Request for Additional Information for GE Topical Reports NEDC-32601 P and NEDC-32694P," Letter, J. H. Wilson (NRC) to R. J. Reda (GE), dated August 20, 1998.

12. "Request for Additional Information for Amendment 25 to GE Topical Report NEDE-24011-P-A (GESTAR II) on Cycle-Specific Safety Limit MCPR (TAC No. M97491)," Letter, J. H. Wilson (NRC) to R. J. Reda (GE), dated October 21, 1998.

13. "Responses to Request for Additional Information for GE Topical Report NEDC-32694P," "Power Distribution Uncertainties for Safety Limit MCPR Evaluations," GAW-98-003, Letter, G. A. Watford (GE) to U.S. NRC, dated January 9, 1998.

14. "Responses to NRC Request for Additional Information Associated with SLMCPR Methodology and Uncertainty Topical Reports NEDC-32601P and NEDC-32694P," GAW 98-009, Letter, G. A. Watford (GE) to U.S. NRC, dated April 17, 1998.

15. "Responses to Request for Additional Information for GE Topical Report NEDC-32601P, Methodology and Uncertainties for Safety Limit MCPR Evaluations," GAW-98-002, Letter, G. A. Wafford (GE) to U.S. NRC, dated January 8, 1998.

16. "Responses to NRC Request for Additional Information for Amendment *25* to GE Topical Report NEDE-2401 1-P-A (GESTAR II) on Cycle-Specific Safety Limit MCPR (TAC No. M97491)," GAW-98-005, Letter, G. A. Watford (GE) to U.S. NRC, dated January 28, 1998.

17. "Additional Information Associated with SLMCPR Methodology and Uncertainty Topical Report NEDC-32601P," GAW-98-012, Letter, G. A. Watford (GE) to U.S. NRC, dated June 12, 1998.

18. "Additional Information Associated with SLMCPR Methodology and Uncertainty Topical Report NEDC-32601P," GAW-98-012, Letter, G. A. Watford (GE) to U.S. NRC, dated July 29, 1998.

19. "Additional Information Associated with SLMCPR Methodology and Uncertainty Topical Report NEDC-32601P," GAW-98-017, Letter, G. A. Watford (GE) to U.S. NRC, dated September 9, 1998.

## ATTACHMENT 5

Global Nuclear Fuel letter REK:00-161 dated October 26, 2000.
Subject: Pilgrim Cycle 14 SLMCPR Information Transmittal.

# (Non-Proprietary Version)

**Global Nuclear Fuel**
A Joint Venture of GE, Toshiba, & Hitachi

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October 26, 2000
REK:00-161

cc: G.T. James
K.R. Nicholas*
ecc: S. B. Shelton
J.P. Riley
* w/o enclosure

Mr. S. Paranjape
Entergy Nuclear Generation Company
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth MA 02360-5599

Subject: Pilgrim Cycle 14 SLMCPR Information Transmittal

Dear Raja:

GNF has completed calculation of the Safety Limit MCPR for the Pilgrim Cycle 14 core using the reduced GETAB uncertainties. The calculated limits are 1.05 for single loop operation and 1.03 for dual loop operation. Per ENGC request, the Tech Spec limit will be set at 1.06 for both single and dual loop operation.

Information regarding the calculation of the SLMCPR, in both proprietary and non-proprietary versions, and an affidavit for NRC submittal are enclosed.

Please call Steve Shelton (910) 675-5591 if you have questions regarding this matter.

Very truly yours,

Rick Kingston

R. E. Kingston
Fuel Project Manager
M/C A33, (910) 675-6192

rk
enclosure

## 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 No. M99070 and M95081), January 11, 1999.

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


## Comparison of Pilgrim Cycle 14 SLMCPR Value

Table 1 summarizes the relevant input parameters and results of the SLMCPR determination for the Pilgrim Cycle 14 and 13 cores. The SLMCPR evaluations were performed using NRC approved methods and uncertainties[1]. These evaluations yield different calculated SLMCPR values because different inputs were used. The quantities that have been shown to have some impact on the determination of the safety limit MCPR (SLMCPR) are provided.

In comparing the Pilgrim Cycle 14 and 13 SLMCPR values it is important to note the impact of the differences in the core and bundle designs. These differences are summarized in Table 1. The Cycle 13 column and the GETAB power distribution uncertainty column for Cycle 14 are both provided for comparison to the Cycle 14 revised power distribution uncertainty column.

[[ ]]

[[ ]].

The uncontrolled bundle pin-by-pin power distributions were compared between the Pilgrim Cycle 14 bundles and the Cycle 13 bundles. Pin-by-pin power distributions are characterized in terms of R-factors using the NRC approved methodology[2]. [[ ]]

## Comparison of the GETAB and Revised Uncertainties

The power distribution and other uncertainties that are the bases for the current TS safety limit for Pilgrim, Cycle 14 are identified in Table 2. Column 2 of Table 2 shows the power distribution and other uncertainties that are the bases for the current TS safety limit for Pilgrim, Cycle 13. The revised bases to support the proposed TS change in safety limit for Pilgrim, Cycle 14 are identified in column 3b of Table 2. The GETAB bases and values for Cycle 14 are provided for comparison purposes in column 3a. By comparing the values from columns 2 for Cycle 13 and column 3a for Cycle 14, one may see that the calculated SLMCPR for Pilgrim, Cycle 14 is slightly higher [[ ]] than the value for Cycle 13 when using the GETAB model and uncertainties for both calculations. [[ ]]

Next let us shift the focus of our discussion of Table 2 on how the revised model and reduced power distribution uncertainties affect the calculated SLMCPR for Pilgrim, Cycle 14. Bases that have not changed are not reported in either table except where it is important to indicate that the bases have not changed. For these exceptions, the impact on the SLMPCR is indicated as "none" in the rightmost column of Table 2. For the other items where a change in basis is indicated, the calculated impact that each item has on the calculated SLMCPR is indicated.

The impacts from the changes in bases have been grouped into four categories. In each category the shaded cells contain values that sum to produce the total impact for that category indicated in the cell immediately below the shaded cells.

In Section 1 of Table 2 the impact of using the "revised uncertainties not related to power distribution" is indicated as "None" since the same revised uncertainties were used for both the GETAB calculation (Column 3a) and the revised calculation (Column 3b).

[[ ]]

For Pilgrim, Cycle 14 the GETAB calculation is more limiting at EOC whereas for the reduced power uncertainties the PHE point ends up being more limiting. [[ ]] For a particular cycle exposure, the same limiting rod pattern was used for both the GETAB and the reduced uncertainties calculations and thus no credit was assumed for a lower OLMCPR that can be obtained when the SLMCPR is reduced. That is why the impact on SLMCPR of a potentially lower OLMCPR is reported as "None" in Section 3 of Table 2.

The impact from using different exposure points (Section 4 of Table 2) was determined by the difference in the calculated SLMCPR for the GETAB results. [[ ]] Therefore, removing this difference from the limiting GETAB results allows one to determine the changes that are due only to the values and the modeling of the power distribution uncertainties as was done in Section 2 of Table 2.

The total impact is that the SLMCPR as calculated using NRC-approved methods, inputs and procedures decreases by [[ ]]. Similar calculated reductions are seen for the SLO SLMCPR. These reductions are consistent with those presented to the NRC in Table 4.3 of NEDC-32694P-A [[ ]].

Reduction in the Tech Spec SLMCPRs by these calculated amounts is warranted since the old GETAB value is overly conservative. The excessive conservatism in the GETAB model and inputs is primarily due to the higher TIPSYS uncertainty that is needed to account for monitoring limitations of the P1 process computer. These limitations are not applicable to the 3D MONICORE (3DM) monitoring system. The revised power distribution model and reduced uncertainties associated with 3DM have been justified, reviewed and approved by the NRC (ref. NEDC-32601P-A and NEDC-32694P-A). The conservatism that remains even when applying the revised model and reduced uncertainties to calculate a lower SLMCPR was documented as part of the NRC review and approval. [[ ]]

**Summary**

[[ ]] have been used to compare quantities that impact the calculated SLMCPR value. Based on these comparisons, the conclusion is reached that the Pilgrim Cycle 13 core/cycle has a flatter core MCPR distribution [[ ]] and flatter in-bundle power distributions [[ ]] than what was used to perform the Cycle 14 SLMCPR evaluation. These facts alone account for a reduction in the calculated GETAB safety limit by 0.02.

The calculated 1.03 Monte Carlo SLMCPR for Pilgrim Cycle 14 is consistent with what one would expect [[ ]] the 1.03 SLMCPR value is 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 facts, observations and arguments presented above, it is concluded that the calculated SLMCPR value of 1.03 for the Pilgrim Cycle 14 core is appropriate. It is reasonable that this value is smaller than the 1.06 value calculated for the previous cycle.

For single loop operations (SLO) the calculated safety limit MCPR for the limiting case is 1.05 as determined by specific calculations for Pilgrim Cycle 14.

**Supporting Information**

The following information is provided in response to NRC questions on previous submittals containing GE14 fuel designs:

*1. Provide the fuel types and numbers of assemblies used in Pilgrim Cycle 14 operation and identify if they are fresh or irradiated fuel (once or twice burned, etc.). Also, provide the fuel loading pattern for Cycle 14 and identify its difference from Cycle 13 and the impact on the SLMCPR calculation.*

Response:

The requested core loading information is provided as Figures 1 and 2. The impact of the fuel loading pattern differences on the calculated SLMCPR is correlated to the values of [[ ]]

*2. The approved methodologies used include NEDC-32694P, NEDC-32601P, Amendment 25 to NEDE-24011P-A, and NEDC-32505P, Revision 1. Please identify which power distribution uncertainties and SLMCPR uncertainties for SLMCPR are used to support this amendment request.*

Response:

For both cycles 13 and 14, a line has been added to Table 1 to indicate which reference is used for the power distribution uncertainties of each analysis. The NRC staff has taken the position in their SER dated March 11, 1999 that the non-power distribution uncertainties reported in NEDC-32601P are "revisions" or "updates" to the GETAB values. GE (GNF) has accepted this position so that the revised non-power distribution uncertainties are used for all SLMCPR calculations performed after June 1999 regardless of which approved methodology is used for the power distribution uncertainties. A line has been added to Table 1 to indicate that the revised non-power distribution uncertainties from NEDC-32601P-A Table 2.1 were used for Pilgrim, Cycle 14.

*3. Provide the details for R-Factor calculation for GE14 fuel and provide the data bases to justify that the approach is conservative with respect to the approved method stated in NEDC-32505P, Revision 1.*

Response:

Calculation of GE14 R-factors follows the approved methodology of NEDC-32505P Rev. 1. The R-factor calculations consist of three essential components: the weight scheme for combining rod peaking factors, the additive constants for adjusting individual position performance and the behavior for partially controlled conditions. The weighting scheme of GE14 is identical to that of GE12 because the two bundles are identical in the lattice geometry. The GE14 bundle is similar to the GE12 bundle. It is a 10x10 design with 78 full length rods, 14 part length rods and 2 large central water rods. The location of the part length rods and the water rods are identical. The main difference is that the length of the part length rods and the spacer locations are slightly different. The additive constants are derived from the test data along with the GEXL coefficients. For partially controlled conditions, the bundle R-factors are calculated based on the prescribed axial power shapes that corresponds to the specific GEXL correlation. [[ ]] The process used for GE14 is the same as the approved methodology in NEDC-32505PA Rev. 1 and the recommendations in the SER.

*4. Provide the details for GEXL14 correlation including its development and verification process, and data bases, and justify that the GEXL14 correlation is conservative.*

Response:

Section 1.2.7 of NEDE-24011-P-A (GESTAR II) provides the conditions by which a GEXL correlation may be developed and documented. Explicit NRC approval of the "GEXL topical report" is not required under the NRC-approved provisions of Amendment 22 to GESTAR II.

An overview of the evaluations performed for GE14 fuel was provided previously in NEDC-32868P, Revision 0, December 1998 titled "GE14 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR II)". This document was transmitted by G. A. Watford (GE) letter MFN-045-98 to the attention of M. J. Davis at the NRC Document Control Desk dated December 11, 1998. Section 2.8.3 of this document describes the GEXL14 correlation.

Additional supporting details were provided previously by separate transmittal of "GEXL14 Correlation for GE14 Fuel", NEDC-32851P, Revision 1, September 1999. This document was transmitted by G.A. Watford (GE) letter FLN-2000-12 dated August 8, 2000 to the NRC Document Control Desk and to the attention of Tai L. Huang (NRC). Section 3 of NEDC-32851P, Rev. 1 describes the database used to develop the GEXL14 correlation for GE14 fuel.

GEXL14 correlation is developed based on the full scale ATLAS test data. The full scale test data were used to generate the GEXL coefficients as well as the additive constants for R-factor calculations to accurately predict the data points over the application range. The report "GE14 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR II)" documents the GEXL14 data and verification base. The database used to develop the GEXL14 correlation consists of [[ ]] different test assemblies. This correlation development database consisted of a total of [[ ]] critical power data points. The database used to verify the GEXL14 correlation consists of [[ ]] different test assemblies. The correlation verification database consisted of a total of [[ ]] data points. [[ ]]

The GEXL14 correlation is valid for GE14 fuel over the following range of state points:

| | Database range | Correlation application range |
|---|---|---|
| Pressure: | [[ | ]] |
| Mass Flux: | [[ | ]] |
| Inlet Subcooling: | [[ | ]] |
| R-factor: | [[ | ]] |
| *exception | | [[ ]] |

[[ ]]

The GEXL14 correlation like previous GEXL correlations is derived as a best fit to the ATLAS critical power data. The GEXL correlation is not intended to be conservative. The GEXL correlation is derived following the process described in GESTAR II (NEDE-24011-P-A-14) Section 1.1.7.C.iv "Correlation fit to data shall be best fit". The bias and uncertainty in the correlation is determined as specified in GESTAR Section 1.1.7. The overall GEXL14 uncertainty is [[ ]]. This uncertainty is an explicit input to the approved SLMCPR methodology.

*5. Provide justification that the impacts of low R-factor and low subcooling are reflected in developing the overall bias and uncertainty, inaccuracies associated with the GEXL correlation are accounted for in the SLMCPR calculation. Also, identify the analysis and the data bases available in the approved topical report.*

Response:

The "GEXL14 Correlation for GE14 Fuel", NEDC-32851P, Revision 1, September 1999 was transmitted by G.A. Watford (GE) letter FLN-2000-12 dated August 8, 2000 to the NRC Document Control Desk and to the attention of Tai L. Huang (NRC). Section 3 of NEDC-32851P, Rev. 1 describes the database used to develop the GEXL14 correlation for GE14 fuel.

[[ ]]

It is difficult to predict and therefore detect the rod location of the boiling transition in a bundle with low R-factor because many rods show the same vulnerability to boiling transition; nevertheless, the critical power value itself is well-predicted. This fact is supported by the lack of any trend in the correlation error as the lower R-factor values are approached. The second point is that the GEXL14 correlation exhibits the typical almost-linear behavior in the critical quality for low R-factor values that one would expect [[ ]]

*6. The staff approved those methodologies cited in Question 2 with one condition that the 3D-MONICORE bundle power calculational uncertainty should be verified when applied to fuel and core designs not included in the benchmark comparisons in Tables 3.1 and 3.2 of NEDC-32694P, and three actions should be taken for application of NEDC-32601P for a new fuel. GE14 is considered a new fuel at the time the staff approved those licensing topical reports, therefore, provide the details of the actions taken and verification for Pilgrim Cycle 14 operation.*

Response:

The referenced requirement for 3D-MONICORE and the three actions pertaining to NEDC-32601P correspond to the four items listed as the NRC's Technical Position in Enclosure 2 accompanying their SER dated March 11, 1999 approving NEDC-32601P and NEDC-32694P. The NRC positions are quoted here together with the actions taken to satisfy each item. Item (a) is the specific requirement from NEDC-32694P that pertains to 3D-MONICORE. Items (b), (c) and (d) are the three actions pertaining to NEDC-32601P referred to in the question.

*Item (a): Since changes in the fuel and core design can have a significant effect on the calculation accuracy, the 3D-MONICORE bundle power calculational uncertainty should be verified when applied to fuel and core designs not included in the benchmark comparisons of Tables-3.1 and 3.2 of NEDC-32694P.*

Changes in the fuel bundle (lattice) design primarily impact the rod-by-rod R-factor uncertainty which is addressed under items (b) and (c). Details of the lattice design also impact the GEXL uncertainty for critical power correlation which is evaluated for each product line, i.e., GE11, GE12, GE13, GE14, etc. A new GEXL correlation and its associated uncertainty and bias are derived and verified whenever a new product line is introduced. Uncertainties associated with the loading and operation of the core are addressed under item (d).

The remaining uncertainty, the bundle power calculational uncertainty, relates to assumptions made regarding how the core is monitored. The bundle power calculational uncertainty for 10x10 (GE12,GE14) designs has been evaluated and found to be represented by the benchmark comparisons for 8x8 and 9x9 designs previously documented in Tables-3.1 and 3.2 of NEDC-32694P-A. Measured core power distributions based on TIP detector readings were compared with the PANACEA predicted core power distributions for cores containing 10x10 fuel. The radial TIP RMS value for the 10x10 cores was consistent with the values in Table 3-1 of NEDC-32694P-A. Thus for cores monitored with 3D-MONICORE in the shape-adaptive mode, the reduced bundle power uncertainty documented in NEDC-32694P-A has been verified to be applicable for all the current GE 8x8 (GE9,GE10), 9x9 (GE11,GE13) and 10x10 (GE12,GE14) bundle designs.

*Item (b): Since changes in fuel design can have a significant effect on the calculation accuracy, the TGBLA fuel rod power calculational uncertainty should be verified when applied to fuel designs not included in the benchmark comparisons of Table 3.1 of NEDC-32601P.*

The fidelity of the TGBLA lattice physics calculations for fuel rod powers depend on the lattice designs. The key considerations are the lattice geometry, the location of the water rods, the location of the gadded rods and for vanished-rod lattices the location of the part-length rods. All these characteristics are identical for GE12 and GE14. See the response to question (3) above. Although the length of the part-length rods is different between GE12 and GE14, this has no impact on the lattice calculations which are performed either for a fully-rodded or partially-rodded lattice. Table 3.1 of NEDC-32601P includes several 10x10 lattices. The values given in Table 3.1 for GE12 are representative of the values being calculated for GE14, thus there is no impact.

*Item (c): The effect of the correlation of rod power calculation uncertainties should be reevaluated to insure the accuracy of R-Factor uncertainty when the methodology is applied to a new fuel lattice.*

The R-factor uncertainty is dominated by the same factors that influence the rod powers as described above for item (b). The uncertainty is the same for GE12 and GE14. The derivation of the uncertainty value is presented for GE 10x10 lattices (i.e., GE12 and GE14) in Appendix C of NEDC-32601P-A.

*Item (d): In view of the importance of MIP criterion and its potential sensitivity to changes in fuel bundle designs, core loading and operating strategies, the MIP criterion should be reviewed periodically as part of the procedural review process to insure that the specific value recommended in NEDC-32601 P is applicable to future designs and operating strategies.*

The calculated value of MIP depends only on two things: [[ ]] The GEXL correlation for GE14 was provided in the Amendment 22 submittal for GE14 together with the uncertainty [[ ]] that is needed for the SLMCPR analyses and the calculation of MIP. See also the response to question (4) above. GE (GNF) continues to monitor MIP and periodically assess it as part of their procedural review process. Specific scoping analyses preformed for cores partially and fully-loaded with GE14 fuel have given no indications that suggests that the MIP values from these calculations are statistically distinct from historical data. [[ ]] Thus there is no indication that the MIP criteria should be changed.

Prepared by:

Technical Project Manager
Pilgrim Project

Verified by:

Nuclear Fuel Engineering

Table 1

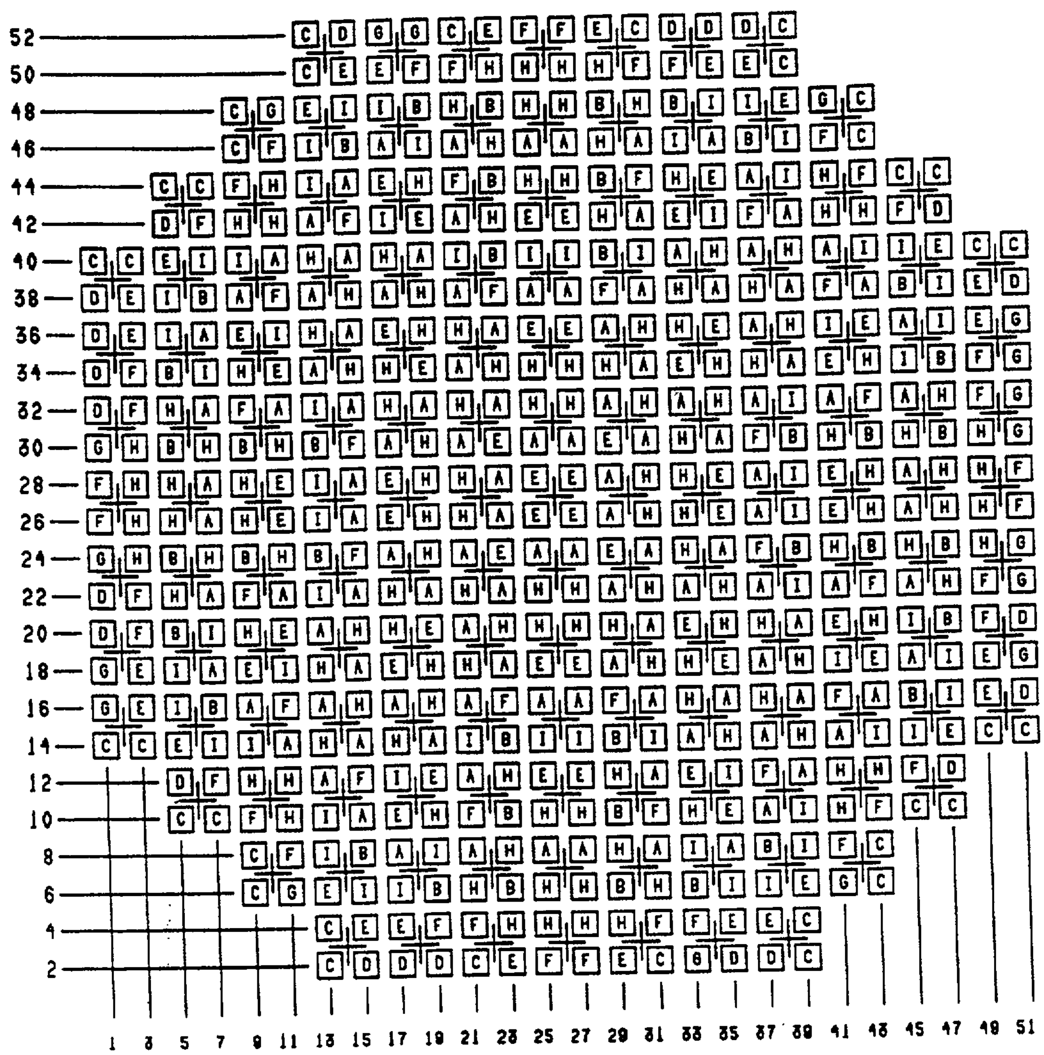
Comparison of the Pilgrim Cycle 14 and Cycle 13 SLMCPR

[[
]]

Table 2 Pilgrim, Cycles 13 and 14

[[ ]]

## Figure 1 Reference Core Loading Pattern – Cycle 13



| Fuel Type | | | |
|---|---|---|---|
| A=GE11–P9DUB407–14GZ–100T–141–T | (Cycle 13) | F=GE11–P9HUB378–15GZ–100T–141–T | (Cycle 11) |
| B=GE11–P9DUB408–6G5.0/7G4.0–100T–141–T | (Cycle 13) | G=GE10–P8HXB355–11GZ–100M–145–T | (Cycle 10) |
| C=GE10–P8HXB355–11GZ–100M–145–T | (Cycle 10) | H=GE11–P9DUB408–16GZ1–100T–141–T | (Cycle 12) |
| D=GE10–P8HXB355–11GZ–100M–145–T | (Cycle 10) | I=GE11–P9DUB408–6G5.0/7G4.0–100T–141–T | (Cycle 12) |
| E=GE11–P9HUB378–15GZ–100T–141–T | (Cycle 11) | | |

## Figure 2 Reference Core Loading Pattern – Cycle 14

| | 1 | 2 | 3 | 4 | 5 | 6 | 7 | 8 | 9 | 10 | 11 | 12 | 13 | 14 | 15 | 16 | 17 | 18 | 19 | 20 | 21 | 22 | 23 | 24 | 25 | 26 | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | | | | | | | 9 | 9 | 9 | 8 | 18 | 18 | 18 | 18 | 18 | 18 | 8 | 9 | 9 | 9 | | | | | | | 52 |
| 2 | | | Bundle IAT | | | | 9 | 15 | 18 | 18 | 18 | 19 | 18 | 18 | 19 | 18 | 18 | 18 | 15 | 9 | | | | | | | 50 |
| 3 | | | | | 8 | 8 | 15 | 19 | 2 | 2 | 15 | 3 | 2 | 2 | 3 | 15 | 2 | 2 | 19 | 15 | 8 | 8 | | | | | 48 |
| 4 | | | | | 9 | 18 | 2 | 2 | 11 | 11 | 11 | 2 | 11 | 11 | 2 | 11 | 11 | 11 | 2 | 2 | 18 | 9 | | | | | 46 |
| 5 | | | 9 | 9 | 15 | 16 | 2 | 11 | 19 | 3 | 19 | 11 | 15 | 15 | 11 | 19 | 3 | 19 | 11 | 2 | 16 | 15 | 9 | 9 | | | 44 |
| 6 | | | 9 | 18 | 16 | 18 | 11 | 2 | 2 | 18 | 11 | 2 | 2 | 2 | 2 | 11 | 18 | 2 | 2 | 11 | 18 | 16 | 18 | 9 | | | 42 |
| 7 | 9 | 8 | 18 | 2 | 3 | 11 | 16 | 11 | 18 | 11 | 2 | 11 | 15 | 15 | 11 | 2 | 11 | 18 | 11 | 16 | 11 | 3 | 2 | 18 | 8 | 9 | 40 |
| 8 | 9 | 19 | 19 | 2 | 11 | 2 | 11 | 2 | 11 | 3 | 11 | 19 | 11 | 11 | 19 | 11 | 3 | 11 | 2 | 11 | 2 | 11 | 2 | 19 | 19 | 9 | 38 |
| 9 | 9 | 18 | 2 | 11 | 19 | 2 | 18 | 11 | 2 | 15 | 2 | 11 | 16 | 16 | 11 | 2 | 15 | 2 | 11 | 18 | 2 | 19 | 11 | 2 | 18 | 9 | 36 |
| 10 | 9 | 18 | 2 | 11 | 3 | 18 | 11 | 3 | 15 | 3 | 11 | 2 | 3 | 3 | 2 | 11 | 3 | 15 | 3 | 11 | 18 | 3 | 11 | 2 | 18 | 9 | 34 |
| 11 | 8 | 18 | 15 | 11 | 19 | 11 | 2 | 11 | 2 | 11 | 18 | 11 | 19 | 19 | 11 | 18 | 11 | 2 | 11 | 2 | 11 | 19 | 11 | 15 | 18 | 8 | 32 |
| 12 | 18 | 19 | 3 | 2 | 11 | 2 | 11 | 19 | 11 | 2 | 11 | 2 | 11 | 11 | 2 | 11 | 2 | 11 | 19 | 11 | 2 | 11 | 2 | 3 | 19 | 18 | 30 |
| 13 | 18 | 15 | 2 | 11 | 18 | 2 | 18 | 11 | 18 | 3 | 18 | 11 | 15 | 15 | 11 | 18 | 3 | 18 | 11 | 18 | 2 | 18 | 11 | 2 | 15 | 18 | 28 |
| 14 | 18 | 15 | 2 | 11 | 18 | 2 | 18 | 11 | 18 | 3 | 18 | 11 | 15 | 15 | 11 | 18 | 3 | 18 | 11 | 18 | 2 | 18 | 11 | 2 | 15 | 18 | 26 |
| 15 | 18 | 19 | 3 | 2 | 11 | 2 | 11 | 19 | 11 | 2 | 11 | 2 | 11 | 11 | 2 | 11 | 2 | 11 | 19 | 11 | 2 | 11 | 2 | 3 | 19 | 18 | 24 |
| 16 | 8 | 18 | 15 | 11 | 19 | 11 | 2 | 11 | 2 | 11 | 18 | 11 | 19 | 19 | 11 | 18 | 11 | 2 | 11 | 2 | 11 | 19 | 11 | 15 | 18 | 8 | 22 |
| 17 | 9 | 18 | 2 | 11 | 3 | 18 | 11 | 3 | 15 | 3 | 11 | 2 | 3 | 3 | 2 | 11 | 3 | 15 | 3 | 11 | 18 | 3 | 11 | 2 | 18 | 9 | 20 |
| 18 | 9 | 18 | 2 | 11 | 19 | 2 | 18 | 11 | 2 | 15 | 2 | 11 | 16 | 16 | 11 | 2 | 15 | 2 | 11 | 18 | 2 | 19 | 11 | 2 | 18 | 9 | 18 |
| 19 | 9 | 19 | 19 | 2 | 11 | 2 | 11 | 2 | 11 | 3 | 11 | 19 | 11 | 11 | 19 | 11 | 3 | 11 | 2 | 11 | 2 | 11 | 2 | 19 | 19 | 9 | 16 |
| 20 | 9 | 8 | 18 | 2 | 3 | 11 | 16 | 11 | 18 | 11 | 2 | 11 | 15 | 15 | 11 | 2 | 11 | 18 | 11 | 16 | 11 | 3 | 2 | 18 | 8 | 9 | 14 |
| 21 | | | 9 | 18 | 16 | 18 | 11 | 2 | 2 | 18 | 11 | 2 | 2 | 2 | 2 | 11 | 18 | 2 | 2 | 11 | 18 | 16 | 18 | 9 | | | 12 |
| 22 | | | 9 | 9 | 15 | 16 | 2 | 11 | 19 | 3 | 19 | 11 | 15 | 15 | 11 | 19 | 3 | 19 | 11 | 2 | 16 | 15 | 9 | 9 | | | 10 |
| 23 | | | | | 9 | 18 | 2 | 2 | 11 | 11 | 11 | 2 | 11 | 11 | 2 | 11 | 11 | 11 | 2 | 2 | 18 | 9 | | | | | 08 |
| 24 | | | | | 8 | 8 | 15 | 19 | 2 | 2 | 15 | 3 | 2 | 2 | 3 | 15 | 2 | 2 | 19 | 15 | 8 | 8 | | | | | 06 |
| 25 | | | | | | | 9 | 15 | 18 | 18 | 18 | 19 | 18 | 18 | 19 | 18 | 18 | 18 | 15 | 9 | | | | | | | 04 |
| 26 | | | | | | | 9 | 9 | 9 | 8 | 18 | 18 | 18 | 18 | 18 | 18 | 8 | 9 | 9 | 9 | | | | | | | 02 |
| | 01 | 03 | 05 | 07 | 09 | 11 | 13 | 15 | 17 | 19 | 21 | 23 | 25 | 27 | 29 | 31 | 33 | 35 | 37 | 39 | 41 | 43 | 45 | 47 | 49 | 51 | |

| Bundle Name | IAT | # in Core | # Fresh | Cycle Loaded |
|---|---|---|---|---|
| GE11-P9DUB407-14GZ-100T-141-T6 | 2 | 120 | 0 | 13 |
| GE11-P9DUB408-6G5.0/7G4.0-100T-141-T6 | 3 | 40 | 0 | 13 |
| GE11-P9HUB378-15GZ-100T-141-T6 | 8 | 20 | 0 | 11 |
| GE11-P9HUB378-15GZ-100T-141-T6 | 9 | 48 | 0 | 11 |
| GE14-P10DNAB412-16GZ-100T-145-T6 | 11 | 144 | 144 | 14 |
| GE11-P9DUB408-16GZ-100T-141-T6 | 15 | 44 | 0 | 12 |
| GE11-P9DUB408-6G5.0/7G4.0-100T-141-T6 | 16 | 16 | 0 | 12 |
| GE11-P9DUB408-16GZ-100T-141-T6 | 18 | 100 | 0 | 12 |
| GE11-P9DUB408-6G5.0/7G4.0-100T-141-T6 | 19 | 48 | 0 | 12 |
| | Total | 580 | 144 | |