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DEC 0 1 2005

U. S. Nuclear Regulatory Commission
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SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED AMENDMENT NO. 283 TO UNIT 1
LICENSE NPF-14: MCPR SAFETY LIMITS
AND REFERENCE CHANGES
PLA-5990 **Docket No. 50-387**

In accordance with the provisions of 10 CFR 50.90, PPL Susquehanna, LLC is submitting a request for an amendment to the Technical Specifications for Susquehanna Unit 1.

The purpose of this letter is to propose changes to the Susquehanna Steam Electric Station Unit 1 Technical Specifications. Included is a revision to Section 2.1.1.2 which reflects the Unit 1 Cycle 15 (U1C15) Minimum Critical Power Ratio (MCPR) Safety Limits for single-loop operation. (Note that there is no change required to the MCPR Safety Limit for two-loop operation.) Additionally, Section 5.6.5.b is revised to include NRC approved methodology used in the MCPR Safety Limit Analysis.

The enclosure to this letter contains PPL's evaluation of this proposed change. Included are a description of the proposed change, technical analysis of the change, regulatory analysis of the change (No Significant Hazards Consideration and the Applicable Regulatory Requirements), and the environmental considerations associated with the change.

Attachment 1 to this letter contains the applicable pages of the Susquehanna SES Unit 1 Technical Specifications, marked to show the proposed change.

Attachment 2 is included which identifies that there are no regulatory commitments associated with this change.

A001

Attachment 3 contains the applicable pages of the Susquehanna SES Unit 1 Technical Specifications Bases, marked to show the proposed changes (Provided for Information).

Attachment 4 has been provided as a description of the U1C15 core composition to assist in your review.

Attachment 5 provides the preliminary U1C15 Core Loading Pattern.

Attachment 6 contains preliminary descriptions of the reload bundles utilized in U1C15.

Attachment 7 contains the relationship between Framatome-ANP (FANP) references and Technical Specification LCO's.

Attachment 8 provides a diagram of the NRC approved MCPR Safety Limit Methodology.

The proposed changes have been approved by the Susquehanna SES Plant Operations Review Committee and reviewed by the Susquehanna Review Committee.

PPL plans to implement the proposed changes in the spring of 2006 to support the startup of U1C15 operation. Therefore, we request NRC complete its review of this change by February 17, 2006 with the changes effective upon startup following the Unit 1 14th Refueling and Inspection Outage.

Any questions regarding this request should be directed to Mr. Duane L. Filchner at (610) 774-7819.

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

Executed on: 12-1-05

B. T. McKinney

B. T. McKinney

Enclosure: PPL Susquehanna Evaluation of the Proposed Changes

Attachments:
Attachment 1 - Proposed Technical Specification Changes Unit 1, (Mark-ups)
Attachment 2 - List of Regulatory Commitments
Attachment 3 - Proposed Technical Specification Bases Changes Unit 1, (Mark-ups)
Attachment 4 - Description of U1C15 Core Composition
Attachment 5 - Preliminary U1C15 Core Loading Pattern
Attachment 6 - Preliminary descriptions of Reload Bundles for U1C15
Attachment 7 - Relationship between FANP references and TS LCO's
Attachment 8 - Diagram of the NRC approved MCPR Safety Limit Methodology

cc: NRC Region I
Mr. B. A. Bickett, NRC Sr. Resident Inspector
Mr. R. V. Guzman, NRC Project Manager
Mr. R. Janati, DEP/BRP

ENCLOSURE to PLA-5990

PPL Evaluation

UNIT 1 CYCLE 15 MCPR SAFETY LIMIT AND COLR REFERENCES

PPL EVALUATION

Subject: Unit 1 Cycle 15 MCPR Safety Limit and COLR References: TS Sections 2.1.1.2 and 5.6.5.b.

1.0 DESCRIPTION

This letter is a request to amend Operating License NPF-14 for PPL Susquehanna, LLC (PPL), Susquehanna Steam Electric Station Unit 1 (SSES).

The proposed changes would revise the Susquehanna Unit 1 Technical Specifications (TS) Section 2.1.1.2 to reflect the Unit 1 Cycle 15 (U1C15) Minimum Critical Power Ratio (MCPR) Safety Limit for single-loop operation. The change to Section 2.1.1.2 is necessary because, as a result of U1C15 cycle specific calculations, the single-loop operation MCPR Safety Limit needs to be increased. The proposed changes also would revise Susquehanna Unit 1 TS Section 5.6.5.b. TS 5.6.5.b lists the analytical methods used to determine the core operating limits contained in the unit / cycle specific Core Operating Limits Report (COLR). The proposed change to TS 5.6.5.b replaces PPL's analytical methods with Framatome-ANP's (FANP) NRC approved analytical methods. FANP's analytical methods will be used to develop the core operating limits documented in the COLR.

The changes are described in detail in Section 4.0.

The requested approval date (February 17, 2006) will allow time for the Core Operating Limits Report to be prepared and reviewed by the Plant Operation Review Committee (PORC) prior to the Spring 2006 Unit 1 outage.

2.0 PROPOSED CHANGE

Specifically the proposed changes would revise the following:

2.1 <u>TS 2.1.1.2</u>

The Minimum Critical Power Ratio (MCPR) Safety Limit, single-loop operation, is revised from 1.10 to 1.12 to reflect results of the cycle specific MCPR Safety Limit analysis for Unit 1 Cycle 15.

2.2 TS 5.6.5.b

Core Operating Limits Report (COLR) references are revised to delete PPL's analytical methods and add FANP's NRC approved analytical methods that are not already contained in Section 5.6.5.b. The references were reordered to correspond to the list provided as Attachment 7.

In summary, the proposed changes would revise the Susquehanna Unit 1 Technical Specifications (TS) Sections 2.1.1.2 and 5.6.5.b. TS Section 2.1.1.2 is revised to reflect the Unit 1 Cycle 15 (U1C15) MCPR Safety Limit for single-loop operation.
TS Section 5.6.5.b is revised to remove references applicable to PPL's analytical methods and add FANP's analytical methods. The TS Bases changes corresponding to the proposed TS changes are included for information.

3.0 BACKGROUND

3.1 MCPR SAFETY LIMIT CHANGE

Excessive thermal overheating of the fuel rod cladding can result in cladding damage and the release of fission products. In order to protect the cladding against thermal overheating due to boiling transition, Safety Limits (Section 2.1.1.2 of the Susquehanna SES Unit 1 Technical Specifications) were established. The change to Section 2.1.1.2 reflects the change from the previous Unit 1 MCPR Safety Limits to the U1C15 MCPR Safety Limits.

NUREG-0800, Standard Review Plan Section 4.4, specifies an acceptable, conservative approach to define this Safety Limit. Specifically, a Minimum Critical Power Ratio (MCPR) value is specified such that at least 99.9% of the fuel rods are expected to avoid boiling transition during normal operation or Anticipated Operational Occurrences (AOOs). Boiling transition is predicted using a correlation based on test data (i.e., a Critical Power Correlation). The Safety Limit MCPR calculation accounts for various uncertainties such as feedwater flow, feedwater temperature, pressure, power distribution uncertainties (including the effects of fuel channel bow), and uncertainty in the Critical Power Correlation.

FANP calculated both two-loop and single-loop Safety Limit MCPR values for Unit 1 Cycle 15 using NRC approved analytical methods with the SPCB critical power correlation for ATRIUM™-10 fuel. Input to the U1C15 MCPR Safety Limit analysis, provided by PPL, assumed the rated core thermal power of 3489 MWt. The Safety Limit

MCPR values assure that at least 99.9% of the fuel rods are expected to avoid boiling transition during normal operation or anticipated operational occurrences. The calculation results demonstrated that the two-loop Safety Limit MCPR did not change from the value contained in the current Unit 1 TS, and the current value of 1.09 remains applicable for Unit 1 Cycle 15. However, the single-loop Safety Limit MCPR increased compared to the current Unit 1 TS value (1.10 to 1.12).

The MCPR Safety Limit analysis is the first in a series of analyses that assure the new core loading for U1C15 is operated in a safe manner. Prior to the startup of U1C15, other licensing analyses are performed (using NRC approved methodology referenced in Technical Specification Section 5.6.5.b) to determine changes in the critical power ratio as a result of anticipated operational occurrences. These results are combined with the MCPR Safety Limit values to generate the MCPR operating limits in the U1C15 COLR. The COLR operating limits thus assure that the MCPR Safety Limit will not be exceeded during normal operation or anticipated operational occurrences, thus providing the required protection for the fuel rod cladding. Postulated accidents are also analyzed prior to the startup of U1C15 and the results shown to be within the NRC approved criteria.

3.2 CHANGES TO COLR REFERENCES

Core operating limits are established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and are documented in the Core Operating Limits Report (COLR). Technical Specification Section 5.6.5.b contains the NRC approved methodology used to determine the core operating limits.

References pertaining to the PPL's analytical methods were removed and FANP's NRC approved analytical methods not already incorporated in Section 5.6.5.b were added. These changes are necessary since the Unit 1 Cycle 15 core operating limits were developed using FANP's NRC approved analytical methods.

The references in Section 5.6.5.b were reordered to be consistent with Attachment 7. Attachment 7 provides the relationship between the references generated by FANP in Section 5.6.5.b and the applicable Technical Specification Limiting Condition for Operation.

4.0 TECHNICAL ANALYSIS

4.1 MCPR SAFETY LIMIT CHANGE

This Technical Specification change increases the single-loop MCPR Safety Limit from 1.10 (current Unit 1 TS) to 1.12 (Unit 1 Cycle 15). The two-loop MCPR Safety Limit (1.09) did not change as a result of the Unit 1 Cycle 15 analysis and remains applicable. The single-loop MCPR Safety Limit change occurs due to cycle-to-cycle variation and changing the critical power correlation from ANFB-10 to SPCB. A more detailed description of the reasons for the 0.02 increase in the single-loop MCPR Safety Limit is provided below in sections "Cycle-to-Cycle Variation" and "SPCB Critical Power Correlation." In addition, due to recent control cell interference issues due to channel bow on Unit 1, the channel bow assumptions for the Unit 1 Cycle 15 MCPR Safety Limit are discussed below ("Channel Bow" section). The channel bow assumptions used in the Unit 1 Cycle 15 MCPR Safety Limit did not contribute to the increase in the single-loop MCPR Safety Limit.

Cycle-to-Cycle Variation

The preliminary Unit 1 Cycle 15 core consists of a full core of FANP's ATRIUM™-10 fuel design. The preliminary core composition is provided as Attachment 4 and the corresponding core loading pattern is provided as Attachment 5. The fresh fuel for Unit 1 Cycle 15 is split into three different assembly types as described in Attachment 6. The descriptions of the Unit 1 Cycle 14 core loading and reload fuel assemblies can be found in the Susquehanna FSAR Section 4.3.

As described previously in PPL correspondence with the NRC (e.g., PLA-5702 dated December 22, 2003) changes in both the two-loop and single-loop MCPR Safety Limits due solely to cycle-to-cycle variation are estimated to range from -0.01 to +0.01.

SPCB Critical Power Correlation

The current Unit 1 MCPR Safety Limit is based on FANP's NRC approved ANFB-10 Critical Power Correlation, EMF-1997(P)(A), which is referenced in TS 5.6.5.b. Beginning with Unit 1 Cycle 15, PPL plans on using FANP's NRC approved SPCB Critical Power Correlation, EMF-2209 (P)(A), for MCPR Safety Limit determination, reload licensing analyses, and MCPR monitoring. SPCB will be included in TS 5.6.5.b per Section 4.2.

Single-loop safety analyses including the MCPR Safety Limit are performed at the highest rod-line that is within the analyzed operating domain for single-loop operation; 76% of rated thermal power and 52 Mlbm/hr core flow. The core flow value of 52 Mlbm/hr is the maximum that can be achieved due to recirculation pump speed limitations in single-loop operation specified in TS 3.4.1. Due to a flow discontinuity in the ANFB-10 Critical Power Correlation near 52 Mlbm/hr core flow, the previous Unit 1 MCPR Safety Limit calculation was performed at a core flow of 55 Mlbm/hr. FANP had previously performed sensitivities for ANFB-10 at various single-loop core flows and demonstrated that 55 Mlbm/hr core flow provides a conservative single-loop MCPR Safety Limit. The SPCB Critical Power Correlation does not have a flow discontinuity at 52 Mlbm/hr. Therefore, FANP performed the U1C15 single-loop MCPR Safety Limit at 52 Mlbm/hr core flow and also at 55 Mlbm/hr core flow. The results of the analyses are provided in the following table:

Percentage of Pins in Boiling Transition
at Single-Loop Conditions for SPCB

	% of Pins in Boiling Transition 52 Mlbm/hr Core Flow	% of Pins in Boiling Transition 55 Mlbm/hr Core Flow
1.10 MCPR SL	0.198	Calculation Not Performed
1.11 MCPR SL	0.131	0.095
1.12 MCPR SL	0.087	Calculation Not Performed

The preceding table shows that changing the core flow from 55 to 52 Mlbm/hr increases the MCPR Safety Limit by 0.01 for the SPCB Critical Power Correlation. Analyzing core flows less than 52 Mlbm/hr is not necessary since a smaller core flow at 76% rated thermal power is outside the analyzed operating domain, and a reduction of both core flow and core thermal power would result in a smaller MCPR Safety Limit.

Channel Bow

The impact of channel bow on the MCPR Safety Limit is included due to recent fuel channel / control rod interference observed during previous Unit 1 operating cycles. The channel bow assumptions used for Unit 1 Cycle 15 are consistent with the current Unit 1 assumptions. Therefore, the change in the single-loop MCPR Safety Limit is not the result of a change in channel bow assumptions.

NRC Bulletin 90-02 was issued to ensure that the effects of channel box bow on the critical power ratio (CPR) calculations are properly taken into account. In response to NRC Bulletin 90-02, FANP issued Supplement 1 to their CPR Methodology, ANF-524(P)(A). The methodology described in ANF-524 has been reviewed and approved by the NRC and incorporated in Section 5.6.5.b. The ANF-524(P)(A) methodology incorporates the effects of channel bow on CPR through the MCPR Safety Limit (SL) calculation.

Based on fuel channel / control rod interference observed during previous Unit 1 operating cycles which may indicate that fuel channel bow is larger than the FANP nominal database, PPL requested FANP to increase the amount of channel bow assumed in the Unit 1 Cycle 15 MCPR Safety Limit calculation. PPL requested that FANP use a mean channel bow for the highly exposed Unit 1 Cycle 15 fuel assemblies of 122.6 mils (from an initial value of 61.3 mils) which is consistent with mean channel bow assumed for the current Unit 1 operating cycle following the recently completed Unit 1 maintenance outage. As part of the continuous validation of safety analyses assumptions, PPL will confirm that the actual Unit 1 Cycle 15 mean channel bow is less than or equal to the mean channel bow assumed. The confirmation will rely on performance data from previously measured fuel channels that were operated in a manner consistent with projected Unit 1 Cycle 15 operation and a potential channel measurement and re-channeling campaign during the refueling and inspection outage preceding Unit 1 Cycle 15 operation. PPL will continue to monitor fuel channel performance in conformance with PPL's fuel channel monitoring program.

Additional Discussion for MCPR SL Change

The proposed change to the MCPR Safety Limits does not directly or indirectly affect any plant system, equipment, component, or change the processes used to operate the plant. As discussed above, the reload analyses performed prior to U1C15 startup will meet all applicable acceptance criteria. Therefore, the proposed changes do not affect the failure modes of any systems or components. Thus, the proposed change does not create the possibility of a previously unevaluated operator error or a new single failure. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Since the proposed change does not alter any plant system, equipment, or component, the proposed change will not jeopardize or degrade the function or operation of any plant system or component governed by Technical Specifications. The proposed MCPR Safety Limits do not involve a significant reduction in the margin of safety as currently defined in the Bases of the applicable Technical Specification sections, because the MCPR Safety Limits calculated for U1C15 preserve the required margin of safety.

Operator performance and procedures are unaffected by these proposed changes since the changes are essentially transparent to the operators and plant procedures, and do not change the way in which the plant is operated. The MCPR Operating Limits to be incorporated in the Core Operating Limits Report (determined from the MCPR Safety Limits and U1C15 transient analysis results) may be different from the current Unit 1 limits. Following use of the methodology to analyze the Unit 1 Cycle 15 core design and future Unit 1 reloads, the reload cycle specific results are incorporated into the FSAR via inclusion of the COLR in the Technical Requirements Manual (TRM).

4.2 CHANGES TO COLR REFERENCES

References pertaining to PPL's analytical methods are removed and FANP's NRC approved analytical methods not already incorporated in Section 5.6.5.b are added. These changes are necessary since the Unit 1 Cycle 15 core operating limits are developed using FANP's NRC approved analytical methods and will allow the use of FANP's NRC approved analytical methods for future reloads.

The references in Section 5.6.5.b are reordered to be consistent with Attachment 7. Attachment 7 provides the relationship between the references generated by FANP in Section 5.6.5.b and the applicable Technical Specification Limiting Condition for Operation. Attachment 8 provides the calculational flowpath and methodology reports used in the MCPR Safety Limit. The attachment shows that the entire MCPR Safety Limit calculation is based on FANP analytical methods.

4.3 CONCLUSION

The changes to Section 5.6.5.b references reflect the NRC approved methodology which will be used to generate Core Operating Limits for Unit 1 Cycle 15.

The proposed change to the MCPR Safety Limit does not affect any plant system, equipment, or component. Therefore, the proposed change will not jeopardize or degrade the function or operation of any plant system or component governed by Technical Specifications. The proposed MCPR Safety Limit change does not involve a significant reduction in the margin of safety as currently defined in the Bases of the applicable Technical Specification sections, because the MCPR Safety Limits calculated for U1C15 preserve the required margin of safety.

Licensing analyses will be performed (using methodology referenced in Technical Specification Section 5.6.5.b) to determine changes in the critical power ratio as a result of anticipated operational occurrences. These results are added to the MCPR Safety

Limit values proposed herein to generate the MCPR operating limits in the U1C15 COLR. Thus, the MCPR operating limits assure that the MCPR Safety Limits will not be exceeded during normal operation or anticipated operational occurrences, providing the required protection for the fuel rod cladding. The proposed change to the MCPR Safety Limits will have a negligible impact on the results of postulated accident analyses.

Therefore, the proposed action does not involve an increase in the probability or an increase in the consequences of an accident previously evaluated in the SAR. Thus, the proposed changes are in compliance with applicable regulations. The health and safety of the public are not adversely impacted by operation of SSES as proposed.

5.0 REGULATORY SAFETY ANALYSIS

5.1 NO SIGNIFICANT HAZARDS CONSIDERATION

The proposed changes would revise the following:

TS 2.1.1.2

The Minimum Critical Power Ratio (MCPR) Safety Limit, single-loop operation, is revised from 1.10 to 1.12 to reflect results of the cycle specific MCPR Safety Limit analysis for Unit 1 Cycle 15.

TS 5.6.5.b

Core Operating Limits Report (COLR) references are revised to delete PPL's analytical methods and add FANP's NRC approved analytical methods that are not already contained in Section 5.6.5.b. The references were reordered to correspond to the list provided as Attachment 7.

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

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

   Response: No.

   The proposed change to the single-loop MCPR Safety Limit does not directly or indirectly affect any plant system, equipment, component, or change the processes used to operate the plant. Further, the proposed U1C15 MCPR Safety Limit was generated using NRC approved methodology and meets the applicable acceptance criteria. Thus, this proposed amendment does not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated.

   Prior to the startup of U1C15, licensing analyses are performed (using NRC approved methodology referenced in Technical Specification Section 5.6.5.b) to determine changes in the critical power ratio as a result of anticipated operational occurrences. These results are added to the MCPR Safety Limit values to generate the MCPR operating limits in the U1C15 COLR. These limits could be different from those specified for the current Unit 1 COLR. The COLR operating limits thus assure that the MCPR Safety Limit will not be exceeded during normal operation or anticipated operational occurrences. Postulated accidents are also analyzed prior to the startup of U1C15 and the results shown to be within the NRC approved criteria.

   The changes to the references in Section 5.6.5.b were made to properly reflect the NRC approved methodology used to generate the U1C15 core operating limits. The use of this approved methodology does not increase the probability of occurrence or consequences of an accident previously evaluated.

   Therefore, this proposed amendment does not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated.

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

   Response: No.

   The change to the single-loop MCPR Safety Limit does not directly or indirectly affect any plant system, equipment, or component and therefore does not affect the failure modes of any of these items. Thus, the proposed change does not create the possibility of a previously unevaluated operator error or a new single failure.

The changes to the references in Section 5.6.5.b were made to properly reflect the NRC approved methodology used to generate the U1C15 core operating limits. The use of this approved methodology does not create the possibility of a new or different kind of accident.

Therefore, this proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

Since the proposed changes do not alter any plant system, equipment, component, or the processes used to operate the plant, the proposed change will not jeopardize or degrade the function or operation of any plant system or component governed by Technical Specifications. The proposed single-loop MCPR Safety Limit does not involve a significant reduction in the margin of safety as currently defined in the Bases of the applicable Technical Specification sections, because the MCPR Safety Limits calculated for U1C15 preserve the required margin of safety.

The changes to the references in Section 5.6.5.b were made to properly reflect the NRC approved methodology used to generate the U1C15 core operating limits. This approved methodology is used to demonstrate that all applicable criteria are met, thus, demonstrating that there is no reduction in the margin of safety.

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

Based upon the above, PPL Susquehanna, LLC (PPL) concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

Title 10 of the Code of Federal Regulations (10 CFR) establishes the fundamental regulatory requirements with respect to reactivity control systems. Specifically, General Design Criterion 10 (GDC-10), "Reactor design," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 states, in part, that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded.

The proposed MCPR Safety Limit values in TS Section 2.1.1.2 will ensure that 99.9% of the fuel rods in the core are not expected to experience boiling transition. This satisfies the requirements of GDC-10 regarding acceptable fuel design limits.

NRC Generic Letter 88-16 (GL 88-16), "Removal of Cycle-Specific Parameter Limits from Technical Specifications," provides guidance on modifying cycle-specific parameter limits in TS. The proposed changes to TS Section 5.6.5.b are in compliance with the guidance specified in GL 88-16.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

10 CFR 51.22(c)(9) identifies certain licensing and regulatory actions, which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility does not require an environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; or (3) result in a significant increase in individual or cumulative occupational radiation exposure. PPL Susquehanna, LLC has evaluated the proposed changes and has determined that the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Accordingly, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with issuance of the amendment. The basis for this determination, using the above criteria, follows:

BASIS

As demonstrated in the No Significant Hazards Consideration Evaluation, the proposed amendment does not involve a significant hazards consideration.

There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite. The proposed change does not involve any physical alteration of the plant (no new or different type of equipment will be installed) or change in methods governing normal plant operation.

There is no significant increase in individual or cumulative occupational radiation exposure. The proposed change does not involve any physical alteration of the plant (no new or different type of equipment will be installed) or change in methods governing normal plant operation.

7.0 REFERENCES

1) PLA-5702, B. L. Shriver (PPL) to USNRC, "Request for Additional Information Regarding Proposed Amendment No. 256 to Unit 1 License NPF-14: MCPR Safety Limits and Reference Changes." dated December 22, 2003.

Attachment 1 to PLA-5990

Proposed Technical Specification Changes (Markups)

(Unit 1)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10 million lbm/hr:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10 million lbm/hr:

1.12

MCPR shall be ≥ 1.09 for two recirculation loop operation or ≥ ~~1.10~~ for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

5.6 Reporting Requirements

5.6.5 COLR (continued)

(102% of 3441 MWt), remains the initial power level for the bounding licensing analysis.

Future revisions of approved analytical methods listed in this Technical Specification that are currently referenced to 102% of rated thermal power (3510 MWt) shall include reference that the licensed RTP is actually 3489 MWt. The revisions shall document that the licensing analysis performed at 3510 MWt bounds operation at the RTP of 3489 MWt so long as the LEFM✓™ system is used as the feedwater flow measurement input into the core thermal power calculation.

The approved analytical methods are described in the following documents, the approved version(s) of which are specified in the COLR.

INSERT 1

1. ~~PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis."~~
2. ~~XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors," Exxon Nuclear Company, Inc.~~
3. ~~XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, Inc.~~
4. ~~ANF-524(P)(A), "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors"~~
5. ~~NE-092-001A, "Licensing Topical Report for Power Uprate With Increased Core Flow," Pennsylvania Power & Light Company.~~
6. ~~ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation.~~

(continued)

5.6 Reporting Requirements

5.6.5 COLR (continued)

7. ~~ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model."~~

8. ~~XN-NF-79-71(P)(A), "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors."~~

9. ~~EMF-1997(P)(A), "ANFB-10 Critical Power Correlation."~~

10. ~~Caldon, Inc., "TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Engineering Report - 80P.~~

11. ~~Caldon, Inc., "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus™ System,", Engineering Report ER -160P.~~

12. ~~EMF-85-74(P)(A), "RODEX 2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model."~~

13. ~~EMF-CC-074 (P)(A), Volume 4, "BWR Stability Analysis: Assessment of STAIF with Input from MICROBURN-B2."~~

14. ~~EMF-2158(P)(A), "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2."~~

15. ~~NEDO-32465-A, "BWROG Reactor Core Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications."~~

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 SDM, 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.

(continued)

INSERT 1:

1. XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company.

2. XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet pump BWR Reload Fuel," Exxon Nuclear Company.

3. EMF-85-74(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation.

4. ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation.

5. XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors," Exxon Nuclear Company.

6. EMF-2158(P)(A), "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," Siemens Power Corporation.

7. EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP.

8. EMF-2292(P)(A), "ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients," Siemens Power Corporation.

9. XN-NF-84-105(P)(A), "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," Exxon Nuclear Company.

10. ANF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors," Advanced Nuclear Fuels Corporation.

11. ANF-913(P)(A), "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation.

12. ANF-1358(P)(A), "The Loss of Feedwater Heating Transient in Boiling Water Reactors," Advanced Nuclear Fuels Corporation.

13. EMF-2209(P)(A), "SPCB Critical Power Correlation," Siemens Power Corporation.

14. EMF-CC-074(P)(A), "BWR Stability Analysis - Assessment of STAIF with Input from MICROBURN-B2," Siemens Power Corporation.

15. NE-092-001A, "Licensing Topical Report for Power Uprate With Increased Core Flow," Pennsylvania Power & Light Company.

16. Caldon, Inc., "TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety while Increasing Operating Power Level Using the LEFTM√™ System," Engineering Report - 80P.

17. Caldon, Inc., "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM√™ or LEFM CheckPlus™ System," Engineering Report ER-160P.

18. NEDO-32465-A, "BWROG Reactor Core Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications."

Attachment 2 to PLA-5990

List of Regulatory Commitments

(Unit 1)

LIST OF REGULATORY COMMITMENTS

REGULATORY COMMITMENTS	Due Date/Event
There are no new commitments associated with this submittal.	NA

Attachment 3 to PLA-5990

Changes to TS Bases Pages
(Markup)

(Unit 1)

BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that an MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with the other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR limit.

2.1.1.1 Fuel Cladding Integrity

The use of the ~~ANFB-10 (~~ SPCB (Reference 4) correlation is valid for critical power calculations at pressures ~~> 571~~ ≥ 571.4 psia and bundle mass fluxes ≥ ~~0.115~~ 0.087 x 10^6 lb/hr-ft^2. For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

> Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to ensure a minimum bundle flow for all fuel assemblies that have a relatively high power and potentially can approach a critical heat flux condition.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

For the SPC ATRIUM-10 design, the minimum bundle flow is > 28×10^3 lb/hr. For the ATRIUM-10 fuel design, the coolant minimum bundle flow and maximum area are such that the mass flux is always > $.25 \times 10^6$ lb/hr-ft^2. Full scale critical power test data taken from various SPC and GE fuel designs at pressures from 14.7 psia to 1400 psia indicate the fuel assembly critical power at 0.25×10^6 lb/hr-ft^2 is approximately 3.35 MWt. At 25% RTP, a bundle power of approximately 3.35 MWt corresponds to a bundle radial peaking factor of approximately 3.0, which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 25% RTP for reactor pressures < 785 psig is conservative.

2.1.1.2 MCPR

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition
(i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty in the critical power correlation. References 2, 4, and 5 describe the methodology used in determining the MCPR SL.

SPCB

The ~~ANFB-10~~ critical power correlation is based on a significant body of practical test data. As long as the core pressure and flow are within the range of validity of the correlations (refer to Section B.2.1.1.1), the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. These conservatisms and the inherent accuracy of the ~~ANFB-10~~ correlation provide a reasonable degree of assurance that during sustained operation at the MCPR SL there would be no transition boiling in the core.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not be compromised.

Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicate that BWR fuel can survive for an extended period of time in an environment of boiling transition.

SPCB

SPC Atrium -10 fuel is monitored using the ~~ANFB-10~~ Critical Power Correlation. The effects of channel bow on MCPR are explicitly included in the calculation of the MCPR SL. Explicit treatment of channel bow in the MCPR SL addresses the concerns of NRC Bulletin No. 90-02 entitled "Loss of Thermal Margin Caused by Channel Box Bow."

Monitoring required for compliance with the MCPR SL is specified in LCO 3.2.2, Minimum Critical Power Ratio.

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.3 Reactor Vessel Water Level (continued)

monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 3). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. ANF 524 (P)(A), Revision 2, "Critical Power Methodology for Boiling Water Reactors," Supplement 1 Revision 2 and Supplement 2, November 1990.

3. 10 CFR 100.

INSERT TSB2.1.1-1

4. ~~EMF-1997 (P)(A), Revision 0, "ANFB-10 Critical Power Correlation," July 1998 and EMF-1997 (P)(A) Supplement 1 Revision 0, "ANFB-10 Critical Power Correlation: High Local Peaking Results," July 1998.~~

5. EMF-2158(P)(A), Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/Microburn-B2," October 1999.

(continued)

BASES

ACTIONS

E.1, E.2, E.3, E.4, and E.5 (continued)

assumed to be isolated to mitigate radioactivity releases. This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances as needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, SRs may need to be performed to restore the component to OPERABLE status. Action must continue until all required components are OPERABLE.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

SDM must be verified to be within limits to ensure that the reactor can be made subcritical from any initial operating condition. Adequate SDM is demonstrated by testing before or during the first startup after fuel movement, control rod replacement, or shuffling within the reactor pressure vessel. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain the SDM, the initial measured value must be increased by an adder, "R", which is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated BOC core reactivity. If the value of "R" is zero (that is, BOC is the most reactive point in the cycle), no correction to the BOC measured value is required (Ref. ~~7~~). 6 For the SDM demonstrations that rely solely on calculation of the highest worth control rod, additional margin (0.10% $\Delta k/k$) must be added to the SDM limit of 0.28% $\Delta k/k$ to account for uncertainties in the calculation.

The SDM may be demonstrated during an in sequence control rod withdrawal, in which the highest worth control rod is analytically determined, or during local criticals, where

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

the highest worth control rod is determined by analysis or testing.

Local critical tests require the withdrawal of control rods in a sequence that is not in conformance with BPWS. This testing would therefore require re-programming or bypassing of the rod worth minimizer to allow the withdrawal of control rods not in conformance with BPWS, and therefore additional requirements must be met (see LCO 3.10.7, "Control Rod Testing - Operating").

The Frequency of 4 hours after reaching criticality is allowed to provide a reasonable amount of time to perform the required calculations and have appropriate verification.

During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each planned in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.

2. FSAR, Section 15.

INSERT TSB 3.1.1-1

3. ~~PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," Sections 2.2 and 2.8, July 1992, Supplement 1-A, August 1995, Supplement 2-A, July 1996 and Supplement 3-A, March 2001.~~

4. FSAR, Section 15.4.1.1.

BASES

REFERENCES (continued)

5. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

6. FSAR, Section 4.3.

DELETE →

7. ~~PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," Section 2.4, July 1992, Supplement 1-A, August 1995, Supplement 2-A, July 1996, and Supplement 3-A, March 2001.~~

BASES

REFERENCES (continued)

4. FSAR, Section 15.0.

5. ~~PL-NF-90-001-A, Applicability of Reactor Analysis Methods for BWR Design and Analysis," Section 4.1.2, July 1992, and Supplement 1-A, August 1995, Supplement 2-A, July 1996, and Supplement 3-A, March 2001.~~

FSAR, SECTION 15.4.9

6. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

7. Letter from R.F. Janecek (BWROG) to R.W. Starostecki (NRC), "BWR Owners Group Revised Reactivity Control System Technical Specifications," BWROG-8754, September 17, 1987.

BASES

ACTIONS

B.1 and B.2 (continued)

of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified to be in compliance with the BPWS at a 24 hour Frequency to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at ≤ 10% RTP.

REFERENCES

INSERT TSB3.1.6-1

1. ~~"PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," Section 2.8, July 1992, Supplement 1-A, August 1995, Supplement 2-A, July 1996, and Supplement 3-A, March 2001.~~

2. "Modifications to the Requirements for Control Rod Drop Accident Mitigating System," BWR Owners Group, July 1986.

3. NUREG-0979, Section 4.2.1.3.2, April 1983.

4. NUREG-0800, Section 15.4.9, Revision 2, July 1981.

5. 10 CFR 100.11.

6. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.

7. ASME, Boiler and Pressure Vessel Code.

8. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

9. NEDO 33091-A, Revision 2, "Improved BPWS Control Rod Insertion Process," April 2003.

BASES

ACTIONS (continued)

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 24 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. Additionally, APLHGRs must be calculated prior to exceeding 50% RTP unless performed in the previous 24 hours. APLHGRs are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 24 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels and because the APLHGRs must be calculated prior to exceeding 50% RTP.

REFERENCES

1. Not used.

2. Not used.

INSERT TSB 3.2.1-1

3. ~~ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," January 1993.~~

4. ANF-CC-33(P)(A) Supplement 2, "HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option," January 1991.

5. XN-CC-33(P)(A) Revision 1, "HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option Users Manual," November 1975.

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in References 2 through ~~8~~ 10. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

state to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency. These analyses may also consider other combinations of plant conditions (i.e., control rod scram speed, bypass valve performance, EOC-RPT, cycle exposure, etc.). Flow dependent MCPR limits are determined by analysis of slow flow runout transients. ~~using the methodology of Reference 2.~~

The MCPR satisfies Criterion 2 of the NRC Policy Statement (Ref. ~~10~~ 11).

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the flow dependent MCPR and power dependent MCPR limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to 25% RTP. This trend is expected to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 (continued)

COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 24 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels and because the MCPR must be calculated prior to exceeding 50% RTP.

SR 3.2.2.2

INSERT TS83.2.2-5

~~Because the transient analysis takes credit for conservatism in the scram time performance, it must be demonstrated that the specific scram time is consistent with those used in the transient analysis. SR 3.2.2.2 determines the scram time fraction which is a measure of the actual scram time compared with the assumed scram time. The COLR contains a table of scram time fractions based on the LCO 3.1.4 "Control Rod Scram Times" and the realistic scram times used in the transient analysis. The MCPR operating limit is then determined based on an interpolation between the applicable limits for scram times of LCO 3.1.4, "Control Rod Scram Times" and realistic scram time analyses using the scram time fraction. The scram time fraction and corresponding MCPR operating limit must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3 and SR 3.1.4.4 because the effective scram times may change during the cycle. The 72 hour Completion Time is acceptable due to the relatively minor changes in the scram time fraction expected during the fuel cycle.~~

REFERENCES

1. NUREG-0562, June 1979.

2. ~~PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," July 1992, Supplement 1-A, August 1995, Supplement 2-A, July 1996, and Supplement 3-A, March 2001.~~

INSERT TS83.2.2-1

(continued)

BASES

REFERENCES (continued)

INSERT TSB3.2.2-2

3. ~~PL-NF-87-001-A, "Qualification of Steady State core Physics Methods for BWR Design and Analysis," April 28, 1988.~~

4. ~~PL-NF-89-005-A, "Qualification of Transient Analysis Methods for BWR Design and Analysis," July 1992, including Supplements 1 and 2.~~

5. XN-NF-80-19 (P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986.

6. NE-092-001, Revision 1, "Susquehanna Steam Electric Station Units 1 & 2: Licensing Topical Report for Power Uprate with Increased Core flow," December 1992, and NRC Approval Letter: Letter from T. E. Murley (NRC) to R. G. Byram (PP&L), "Licensing Topical Report for Power Uprate With Increased Core Flow, Revision 0, Susquehanna Steam Electric Station, Units 1 and 2 (PLA-3788) (TAC Nos. M83426 and M83427)," November 30, 1993.

INSERT TSB3.2.2-3

7. ~~EMF-1997 (P)(A) Revision 0, ANFB-10 Critical Power Correlation," July 1998, and EMF-1997 (P)(A) Supplement 1 Revision 0, "ANFB-10 Critical Power Correlation : High Local Peaking Results," July 1998.~~

8. XN-NF-79-71(P)(A) Revision 2, Supplements 1, 2, and 3, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," March 1986.

9. XN-NF-84-105(P)(A), Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," February 1987.

11 ~~10~~. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

INSERT TSB3.2.2-4

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.10.7.2

When the RWM provides conformance to the special test sequence, the test sequence must be verified to be correctly loaded into the RWM prior to control rod movement. This Surveillance demonstrates compliance with SR 3.3.2.1.8, thereby demonstrating that the RWM is OPERABLE. A Note has been added to indicate that this Surveillance does not need to be performed if SR 3.10.7.1 is satisfied.

REFERENCE

1. FSAR 15.4.9

INSERT TSB3.10.7-1

2. ~~PL-NF-90-001-A "Application of Reactor Analysis Methods for BWR Design and Analysis," July 1992 and Supplement 1-A, August 1995, Supplement 2-A, July 1996, and Supplement 3-A, March 2001.~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.10.8.4

Periodic verification of the administrative controls established by this LCO will ensure that the reactor is operated within the bounds of the safety analysis. The 12 hour Frequency is intended to provide appropriate assurance that each operating shift is aware of and verifies compliance with these Special Operations LCO requirements.

SR 3.10.8.5

Coupling verification is performed to ensure the control rod is connected to the control rod drive mechanism and will perform its intended function when necessary. The verification is required to be performed any time a control rod is withdrawn to the "full out" notch position, or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved as well as operating experience related to uncoupling events.

SR 3.10.8.6

CRD charging water header pressure verification is performed to ensure the motive force is available to scram the control rods in the event of a scram signal. A minimum accumulator pressure is specified, below which the capability of the accumulator to perform its intended function becomes degraded and the accumulator is considered inoperable. The minimum accumulator pressure of 940 psig is well below the expected pressure of 1100 psig. The 7 day Frequency has been shown to be acceptable through operating experience and takes into account indications available in the control room.

REFERENCE

1. ~~PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," July 1992 and Supplement 1-A, August 1995, Supplement 2-A, July 1996, and Supplement 3-A, March 2001.~~

INSERT TSB3.10.8-1

INSERT TSB2.1.1-1:

4. EMF-2209(P)(A), Revision 1, "SPCB Critical Power Correlation," Siemens Power Corporation, July 2000.

INSERT TSB3.1.1-1:

XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors," Exxon Nuclear Company, March 1983.

INSERT TSB3.1.6-1:

XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors," Exxon Nuclear Company, March 1983.

INSERT TSB 3.2.1-1

EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP.

INSERT TSB 3.2.2-1

2. XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors," Exxon Nuclear Company, March 1983.

INSERT TSB 3.2.2-2

3. XN-NF-80-19(P)(A) Volume 3 Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description," Exxon Nuclear Company, January 1987.

4. ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," Advanced Nuclear Fuels Corporation, August 1990.

INSERT TSB 3.2.2-3

7. EMF-2209(P)(A), Revision 1, "SPCB Critical Power Correlation," Siemens Power Corporation, July 2000.

INSERT TSB 3.2.2-4

10. ANF-1358(P)(A) Revision 1, "The Loss of Feedwater Heating Transient in Boiling Water Reactors," Advanced Nuclear Fuels Corporation, September 1992.

INSERT TSB 3.2.2-5

Because the transient analysis takes credit for conservatism in the scram time performance, it must be demonstrated that the specific scram time is consistent with those used in the transient analysis. SR 3.2.2.2 compares the average measured

scram times to the assumed scram times documented in the COLR. The COLR contains a table of scram times based on the LCO 3.1.4, "Control Rod Scram Times" and the realistic scram times, both of which are used in the transient analysis. If the average measured scram times are greater than the realistic scram times then the MCPR operating limits corresponding to the Maximum Allowable Average Scram Insertion Time must be implemented. Determining MCPR operating limits based on interpolation between scram insertion times is not permitted. The average measured scram times and corresponding MCPR operating limit must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3 and SR 3.1.4.4 because the effective scram times may change during the cycle. The 72 hour Completion Time is acceptable due to the relatively minor changes in average measured scram times expected during the fuel cycle.

INSERT TSB3.10.7-1:

XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors," Exxon Nuclear Company, March 1983.

INSERT TSB3.10.8-1:

XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors," Exxon Nuclear Company, March 1983.

Attachment 4 to PLA-5990

Preliminary Unit 1 Cycle 15 Core Composition

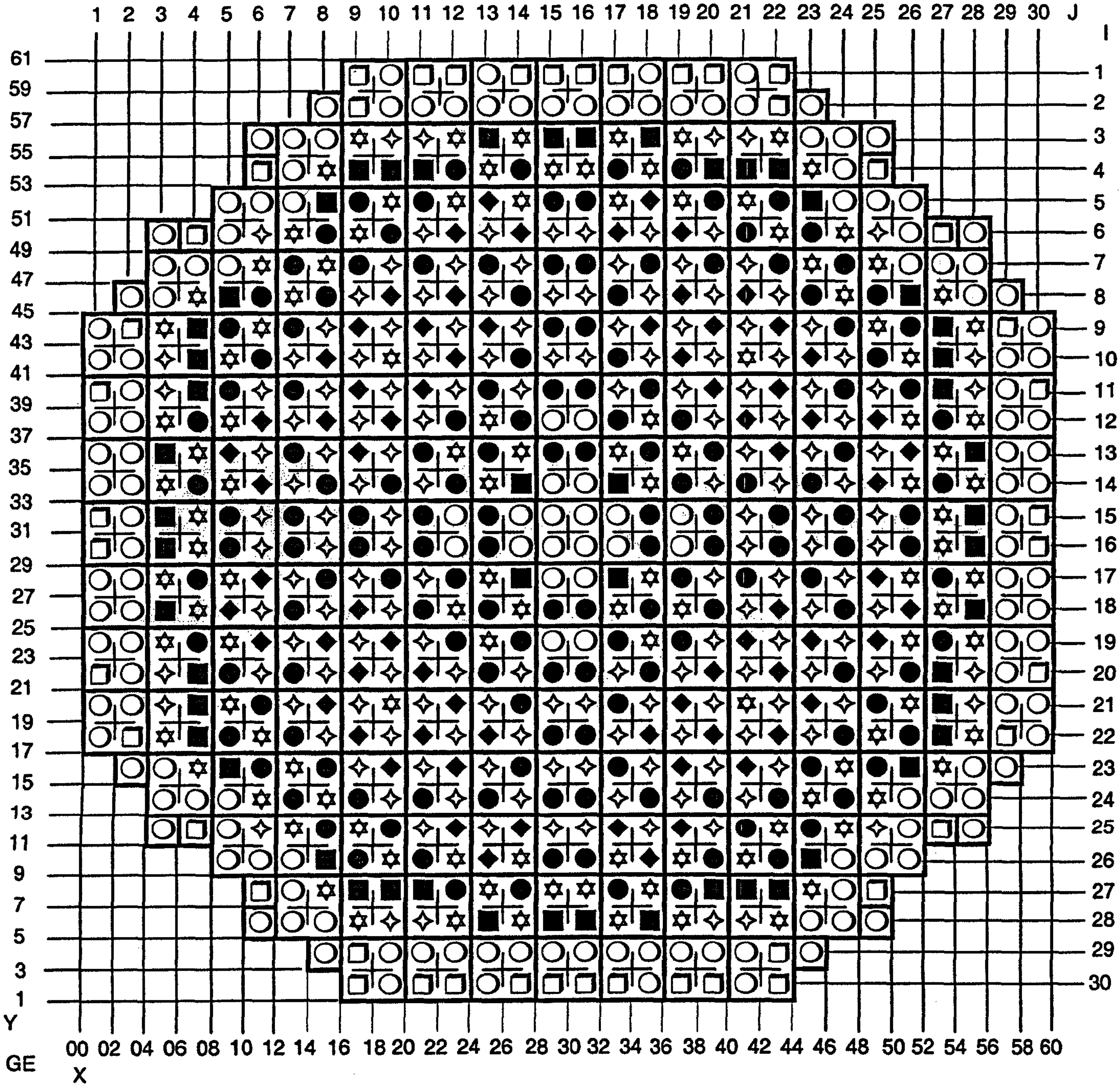
Preliminary Unit 1 Cycle 15 Core Composition

Assembly Type	Operational History	Number of Assemblies
FANP ATRIUM™-10	Fresh	284
FANP ATRIUM™-10	Once-burned	280
FANP ATRIUM™-10	Twice-burned	200

Attachment 5 to PLA-5990

Preliminary Unit 1 Cycle 15
Core Loading Pattern

PRELIMINARY SUSQUEHANNA UNIT 1 CYCLE 15 CORE LOADING PATTERN



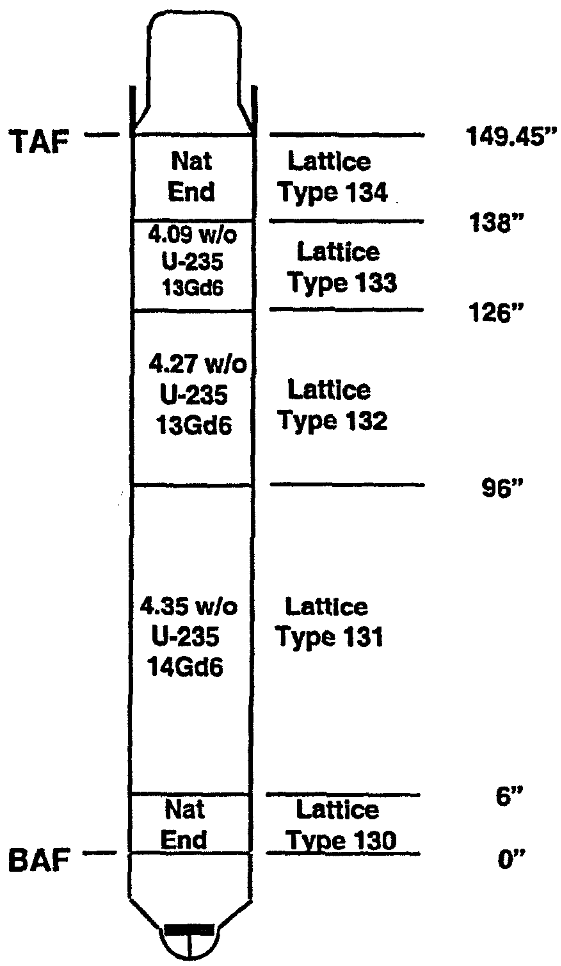
- ◯ SQA-12 12GdZ/1Gd7 TWICE BURNED (3.97)
- □ SQA-12 13Gd6/12GdZ TWICE BURNED (3.75)
- ✡ SQA-13 14GdZ ONCE BURNED (4.12)
- ✧ SQA-13 14GdZ ONCE BURNED (3.90)
- ● SUS1-15 15GdZ FRESH (4.11)
- ■ SUS1-15 14GdZ FRESH (3.91)
- ◆ SUS1-15 16GdZ FRESH (4.09)

Attachment 6 to PLA-5990

Preliminary Unit 1 Cycle 15
Fresh Fuel Description

Preliminary
Assembly Type 60
Reload Bundle Description
(ATRIUM-10, 100mil Channel)

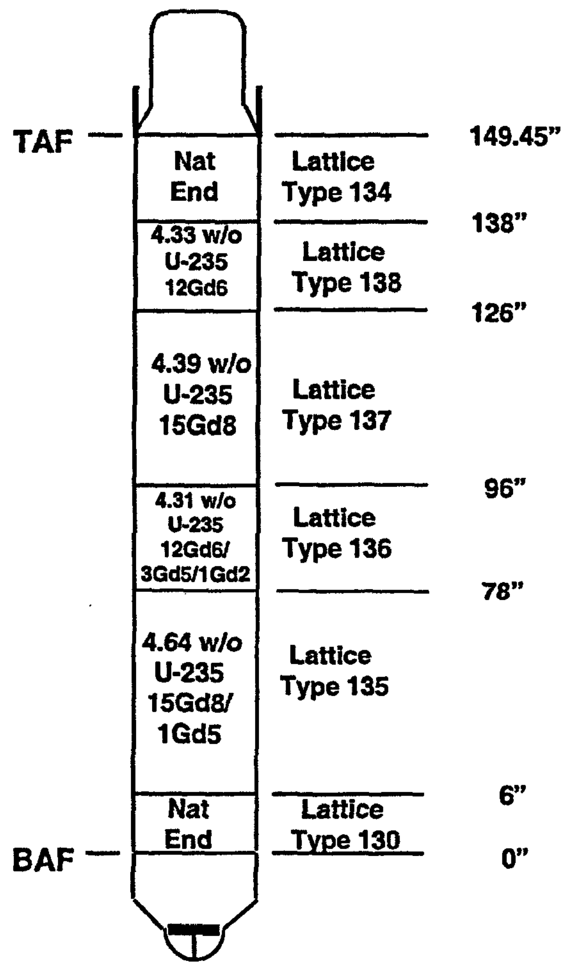
Bundle Average Enrichment = 3.91%



G:\NFE\NFEPKG\U1C15\NFE115001_R0\NQPlots\BUNDLE60.PPT

Preliminary
Assembly Type 61
<u>Reload Bundle Description</u>
(ATRIUM-10, 100mil Channel)

Bundle Average Enrichment = 4.09%



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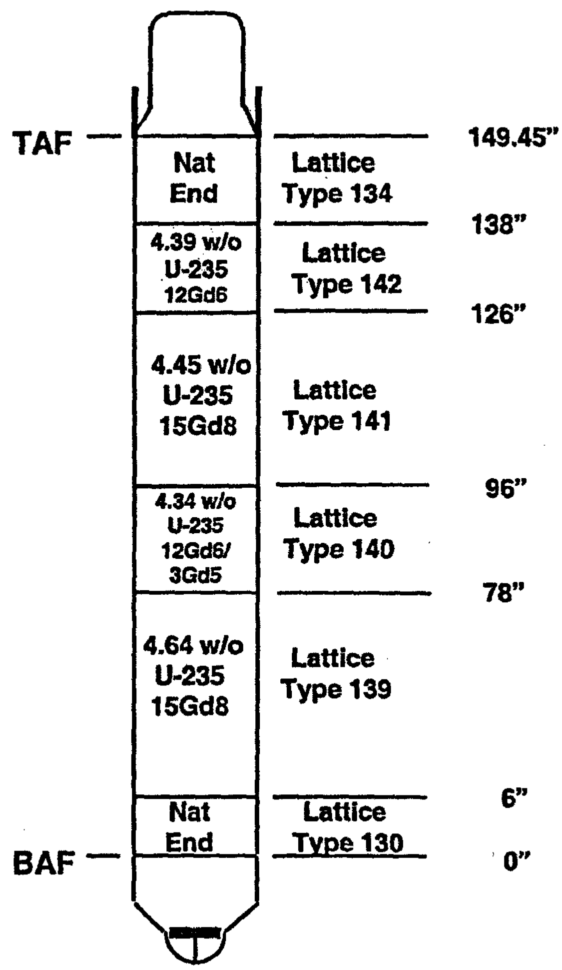
Preliminary
Assembly Type 62
<u>Reload Bundle Description</u>
(ATRIUM-10, 100mil Channel)

Bundle Average Enrichment = 4.11%



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Attachment 7 to PLA-5990

Listing of Framatome-ANP's Approved Methodology and Applicable LCOs

BWR Approved Topical Reports for Susquehanna Nuclear Plant Technical Specifications COLR References

Report	Applicable LCO	Methodology / Justification
XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, *RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model*, Exxon Nuclear Company, March 1984.	3.2.1 3.2.2 3.2.3	Provides an analytical capability to predict BWR fuel thermal and mechanical conditions for normal core operation and to establish initial conditions for power ramping, non-LOCA and LOCA analyses.
XN-NF-85-67(P)(A) Revision 1, *Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel*, Exxon Nuclear Company, September 1986.	3.2.3	Describes the process used to develop linear heat generation rates for fuel designs.
EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), *RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model*, Siemens Power Corporation, February 1998.	3.2.3	Extends the exposure limit of the RODEX2A code which is a version of RODEX2 that includes a fission gas release model specific to BWR fuel designs.
ANF-89-98(P)(A) Revision 1 and Supplement 1, *Generic Mechanical Design Criteria for BWR Fuel Designs*, Advanced Nuclear Fuels Corporation, May 1995.	3.2.3	Establishes a set of design criteria which assures that BWR fuel will perform satisfactorily throughout its lifetime.
XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, *Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis*, Exxon Nuclear Company, March 1983.	3.1.1 3.2.1 3.2.2 3.2.3 3.3.2.1 Table 3.3.2.1-1	Development of BWR core analysis methodology which comprises codes for fuel neutronic parameters and assembly burnup calculations, reactor core simulation diffusion theory calculations, core and channel hydrodynamic stability predictions, and producing input for nuclear plant transients. Subsequently approved codes or methodologies have superceded portions of this report. Applicable portions include CRDA, and methodology to determine neutronic reactivity parameters, void reactivity, Doppler reactivity, scram reactivity, delayed neutron fraction, and prompt neutron lifetime.
XN-NF-80-19(P)(A) Volume 4 Revision 1, *Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads*, Exxon Nuclear Company, June 1986.	3.2.1 3.2.2 3.2.3	Summarizes the types of BWR licensing analyses performed, identifies the methodologies used.
EMF-2158(P)(A) Revision 0, *Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2*, Siemens Power Corporation, October 1999.	3.1.1 3.2.2 3.2.3 3.3.2.1 Table 3.3.2.1-1	Describes the reactor core simulator code MICROBURN-B2 and the lattice physics code CASMO-4.

Report	Applicable LCO	Methodology / Justification
XN-NF-80-19(P)(A) Volumes 2, 2A, 2B and 2C, *Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model,* Exxon Nuclear Company, September 1982.	3.2.1	Describes an evaluation model methodology for licensing analyses of postulated LOCAs in jet pump BWRs. The methodology was developed to comply with 10 CFR 50.46 and Appendix K criteria to 10 CFR 50.
EMF-2361(P)(A) Revision 0, *EXEM BWR-2000 ECCS Evaluation Model,* Framatome ANP, May 2001.	3.2.1	Describes an upgraded evaluation model methodology for licensing analyses of postulated LOCAs in jet pump BWRs. The methodology was developed to comply with 10 CFR 50.46 and Appendix K criteria to 10 CFR 50.
EMF-2292(P)(A) Revision 0, *ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients,* Siemens Power Corporation, September 2000.	3.2.1	Provides measured cladding temperatures from spray heat transfer tests to justify the use of Appendix K coefficients for ATRIUM-10 fuel LOCA analyses.
XN-NF-80-19(P)(A) Volume 3 Revision 2, *Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description,* Exxon Nuclear Company, January 1987.	3.2.2	Provides overall methodology for determining a MCPR operating limit.
XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, *XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis,* Exxon Nuclear Company, February 1987.	3.2.2	Provides a capability to perform analyses of transient heat transfer behavior in BWR assemblies.
ANF-524(P)(A) Revision 2 and Supplements 1 and 2, *ANF Critical Power Methodology for Boiling Water Reactors,* Advanced Nuclear Fuels Corporation, November 1990.	3.2.2	Provides a methodology for the determination of thermal margins, specifically the MCPR safety limit.
ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4, *COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses,* Advanced Nuclear Fuels Corporation, August 1990.	3.2.2	Provides a computer program for analyzing BWR system transients.

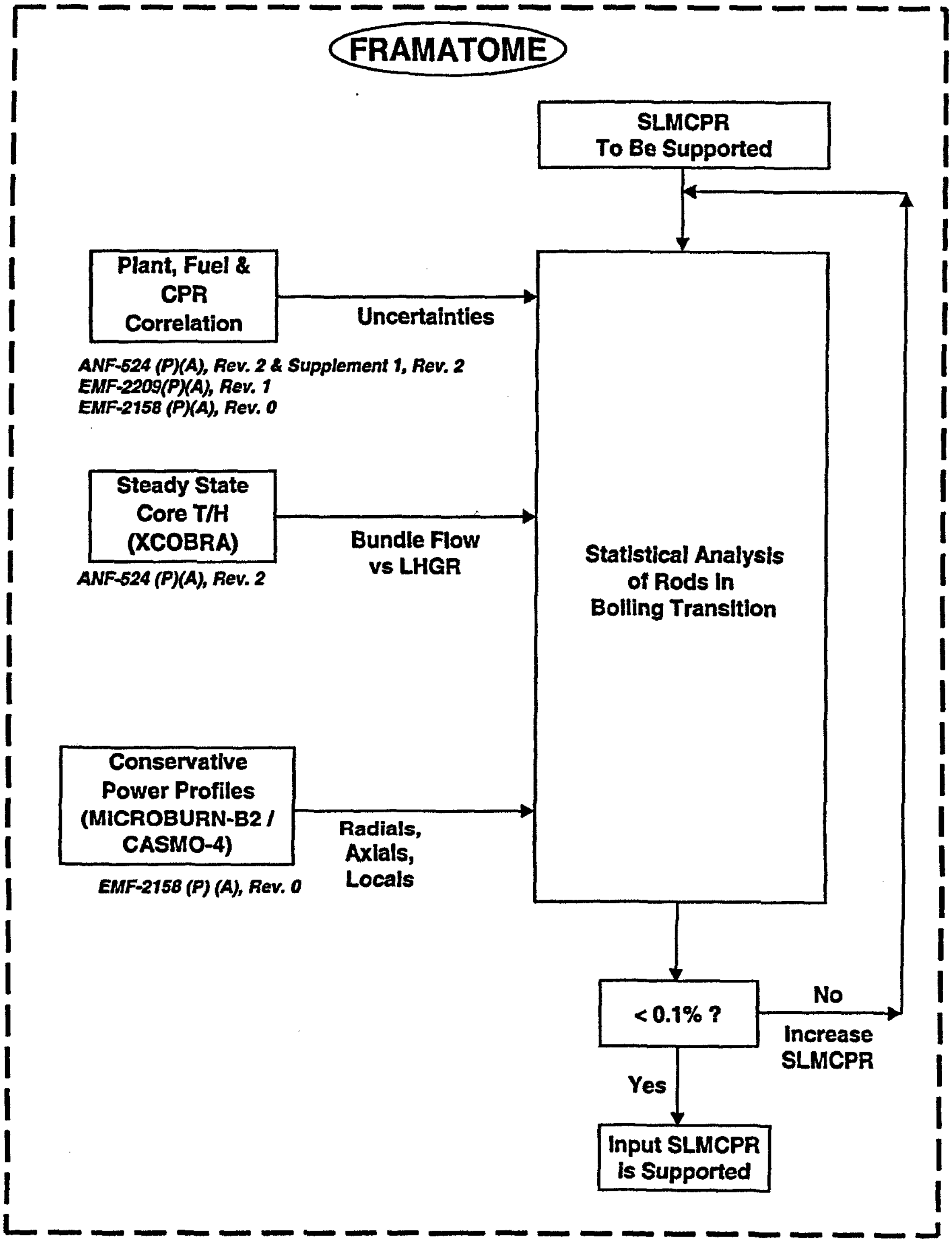
Report	Applicable LCO	Methodology / Justification
ANF-1358(P)(A) Revision 1, *The Loss of Feedwater Heating Transient in Boiling Water Reactors,* Advanced Nuclear Fuels Corporation, September 1992.	3.2.2	Presents a generic methodology for evaluating the loss of feedwater heating event.
EMF-2209(P)(A) Revision 1, *SPCB Critical Power Correlation,* Siemens Power Corporation, July 2000.	3.2.2	Presents a critical power correlation for use with the ATRIUM™-10* fuel designs. This correlation is used in the BWR-2000 LOCA methodology.
EMF-CC-074(P)(A) Volume 4 Revision 0, *BWR Stability Analysis - Assessment of STAIF with Input from MICROBURN-B2,* Siemens Power Corporation, August 2000.	3.4.1	Provides a computer program for performing stability analysis.

* ATRIUM is a trademark of Framatome ANP.

Attachment 8 to PLA-5990

MCPR Safety Limit Methodology

MCPR Safety Limit Methodology



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01/20/2005