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JUL 0 1 2003

U. S. Nuclear Regulatory Commission
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Washington, DC 20555

**SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED AMENDMENT NO. 256 TO UNIT 1
LICENSE NPF-14: MCPR SAFETY LIMITS
AND REFERENCE CHANGES
PLA-5638** **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 14 (U1C14) Minimum Critical Power Ratio (MCPR) Safety Limits for both two-loop and single-loop operation. Section 4.2.1 is also revised to indicate the use of depleted uranium in reload fuel bundles. Additionally, Section 5.6.5.b is revised to include NRC approved methodology that forms the basis for particular uncertainties 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 contains the applicable pages of the Susquehanna SES Unit 1 Technical Specifications Bases, marked to show the proposed change.

Attachment 3 contains the "camera ready" version of the revised Technical Specification pages.

A001

Attachment 4 is included to identify any regulatory commitments associated with this change.

Attachment 5 has been provided as a description of the U1C14 core composition to assist in your review.

The proposed change has been approved by station management as recommended 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 2004 to support the startup of U1C14 operation. Therefore, we request NRC complete its review of this change by January 31, 2004 with the changes effective upon startup following the Unit 1 13th 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: 7/01/03

B. T. Jones
for B. L. Shriver

Enclosure:
PPL Evaluation of the Proposed Change

Attachments:
1. Proposed Technical Specification Changes Unit 1, (Mark-ups)
2. Proposed Technical Specification Bases Changes Unit 1, (Mark-ups)
3. Proposed Technical Specification Pages Unit 1, (Camera Ready)
4. List of Regulatory Commitments
5. Description of U1C14 Core Composition

cc: NRC Region I
Mr. S. L. Hansell, NRC Sr. Resident Inspector
Mr. R. V. Guzman, NRC Project Manager
Mr. R. Janati, DEP/BRP

ENCLOSURE to PLA-5638

PPL Evaluation

UNIT 1 CYCLE 14 MCPR SAFETY LIMIT

PPL EVALUATION

1.0 DESCRIPTION

The proposed changes would revise the Susquehanna Unit 1 Technical Specifications (TS) Section 2.1.1.2 to reflect the Unit 1 Cycle 14 (U1C14) Minimum Critical Power Ratio (MCPR) Safety Limits for both two-loop and single-loop operation. The change to Section 2.1.1.2 is necessary because, as a result of U1C14 cycle specific calculations, the two-loop and single-loop operation MCPR Safety Limits are decreased relative to existing Unit 1 TS values. The proposed changes also would revise Susquehanna Unit 1 TS Section 4.2.1 to indicate the use of depleted uranium in reload fuel bundles. The change to Section 4.2.1 is necessary to reflect the fact that the U1C14 reload utilizes fuel pins that contain depleted uranium. Use of depleted uranium provides an economic advantage, since less uranium is required for the reload, and it also provides MCPR operating margin improvement. Depleted uranium is modeled in the approved design and licensing methodology. Previous industry use of depleted uranium in reloads includes Susquehanna Unit 2, River Bend, and Grand Gulf. The proposed changes also would revise Susquehanna Unit 1 TS Section 5.6.5.b to remove references applicable to Framatome-ANP (FANP) 9x9-2 fuel (which is no longer contained in Unit 1) and the ANFB critical power correlation and add a reference describing (FANP) NRC approved methodology. The addition of the reference to Section 5.6.5.b is needed to include NRC approved methodology that forms the basis for particular uncertainties used in the MCPR Safety Limit Analysis. In addition, this NRC approved methodology may be used to perform certain licensing analyses for U1C14. The removal of various references from Section 5.6.5.b pertaining to 9x9-2 fuel and the ANFB CPR correlation is necessary to reflect the fact that 9x9-2 fuel and the ANFB CPR correlation are not used in U1C14.

2.0 PROPOSED CHANGE

Specifically the proposed changes would revise the following:

2.1 TS 2.1.1.2

The Minimum Critical Power Ratio (MCPR) Safety Limits (two-loop operation and single-loop operation) are revised from 1.12 (two-loop operation) and 1.13 (single loop operation) to 1.08 (two-loop operation) and 1.10 (single loop operation) to reflect results of the cycle specific MCPR Safety Limit analysis for Unit 1 Cycle 14.

2.2 TS 4.2.1

Design Feature 4.2.1 is revised to indicate the use of depleted uranium in reload fuel bundles.

2.3 TS 5.6.5.b

Core Operating Limits Report (COLR) references are revised to delete references to 9x9-2 fuel and the ANFB correlation. A reference is added to describe NRC approved methodology applicable to the MCPR Safety Limit change. This additional reference also enables use of the methodology for core physics analysis to support U1C14 and future reloads.

In summary, the proposed changes would revise the Susquehanna Unit 1 Technical Specifications (TS) Sections 2.1.1.2, 4.2.1, and 5.6.5.b. The TS Bases changes corresponding to the proposed TS changes are also included as Attachment 2 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 U1C13 MCPR Safety Limits to the U1C14 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.

The proposed Safety Limit MCPR values (two-loop and single-loop) were calculated using FANP NRC approved licensing methods with the ANFB-10 critical power correlation for ATRIUM™-10 fuel. Input to the U1C14 MCPR Safety Limit analysis, provided by PPL, assumed the rated core thermal power of 3489 MWt. The proposed Safety Limit MCPR values (two-loop and single-loop) assure that at least 99.9% of the fuel rods are expected to avoid boiling transition during normal operation or anticipated operational occurrences.

The MCPR Safety Limit analysis is the first in a series of analyses that assure the new core loading for U1C14 is operated in a safe manner. Prior to the startup of U1C14, 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 proposed here to generate the MCPR operating limits in the U1C14 COLR. The COLR operating limits 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 U1C14 and the results shown to be within the NRC approved criteria.

3.2 Design Features Change

Section 4.2.1 contains a description of the fuel assemblies contained in the core. The U1C14 reload fuel bundles will utilize a small amount of depleted uranium in the fuel rods, in addition to natural and slightly enriched uranium. Thus, Section 4.2.1 was modified to reflect this change. Depleted uranium is modeled in the approved design and licensing methodology.

3.3 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 analysis methodologies used to analyze FANP 9x9-2 fuel were removed since FANP 9x9-2 fuel is no longer used in Unit 1. In addition, references pertaining to the ANFB critical power correlation methodology were removed since the ANFB critical power correlation is not used for U1C14.

A reference is added to Section 5.6.5.b. This reference documents NRC approved FANP analysis methodology. This reference is added since the methodology provides the basis for power distribution uncertainties applied to the MCPR Safety limit analysis. The reference also enables use of this methodology for core physics analysis to support U1C14 and future reloads.

4.0 TECHNICAL ANALYSIS

4.1 MCPR Safety Limit Change

This Technical Specification change decreases the MCPR safety limits from Unit 1 Cycle 13 (1.12 two-loop and 1.13 single loop) to Unit 1 Cycle 14 (1.08 two-loop and 1.10 single loop). The MCPR safety limit decrease occurs for the following reasons:

1. Removal of an excess conservatism that is not required by NRC approved methodology, and

2. Incorporation of smaller power distribution uncertainties in the MCPR Safety Limit analysis that are based on NRC approved CASMO-4/MICROBURN-B2 methodology (This approved methodology is consistent with implementation of the POWERPLEX®-III Core Monitoring System for U1C14).

Removal of Excess Conservatism

NRC approval of the previously used ANFB critical power correlation required a factor of 2 to be applied to the number of pins calculated to be in boiling transition for the MCPR Safety Limit calculation. The technical basis for this requirement was that the mean of the ANFB correlation (measured over predicted critical power) was very slightly greater than 1.0 (i.e., non-conservative). Unit 1 Cycle 13 and Unit 1 Cycle 14 apply the ANFB-10 critical power correlation. The NRC SER on the ANFB-10 correlation (Ref. 1) does not require use of a factor to be applied to the number of rods calculated to be in boiling transition. It should be noted that the mean of the ANFB-10 correlation is slightly less than 1.0 (i.e., conservative). Thus, there is no technical requirement that the factor of 2 be applied to the ANFB-10 correlation.

Although not required by Reference 1 (NRC SER attached to EMF-1997(P)(A), Rev. 0 and EMF-1997, Supplement 1 (P)(A), Rev. 0, dated July 17, 1998), the factor of 2 was conservatively applied to the Unit 1 Cycle 13 MCPR Safety Limit calculation. Application of the factor of 2 to the Unit 1 Cycle 13 MCPR Safety Limit represented an overly conservative input. For the Unit 1 Cycle 14 MCPR Safety Limit calculation, this factor of 2 is not applied. A similar change was addressed in the NRC SER for the Unit 2 Cycle 12 MCPR Safety Limit (Ref. 2). The NRC SER in Reference 2 concluded that there is no technical basis for applying the multiplier to the ANFB-10 correlation.

Power Distribution Uncertainties

The NRC approved MCPR Safety Limit methodology referenced in T.S. 5.6.5.b uses radial and local power distribution uncertainties that are based on NRC approved statistical methods and code system benchmarks. For the previous Unit 1 Cycle 13 MCPR Safety Limit, radial and local power distribution uncertainties were based on the NRC approved CASMO-3/MICROBURN-B code system that is implemented within the POWERPLEX®-II core monitoring system. The POWERPLEX®-II core monitoring system is used for Unit 1 Cycle 13 operation, thus the CASMO-3/MICROBURN-B based uncertainties are used. For the Unit 1 Cycle 14 MCPR Safety Limit, radial and local power distribution uncertainties are based on the NRC approved CASMO-4/MICROBURN-B2 code system that is implemented within the POWERPLEX®-III core monitoring system. The POWERPLEX®-III core monitoring system will be applied to Unit 1 Cycle 14 operation. Radial and local power distribution uncertainties based on the CASMO-4/MICROBURN-B2 code system are smaller than the corresponding uncertainties based on the CASMO-3/MICROBURN-B code system.

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 U1C14 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 U1C14 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 COLR (determined from the MCPR Safety Limits and U1C14 transient analysis results) may be different from the U1C13 limits. Following use of the methodology to analyze the Unit 1 Cycle 14 core design and future Unit 1 reloads, the reload cycle specific results are incorporated into the FSAR via a licensing document change notice, which also includes use of the POWERPLEX®-III core monitoring system for Unit 1.

4.2 Design Features Change

The U1C14 reload fuel bundles will utilize a small amount of depleted uranium in certain fuel rods, in addition to natural and slightly enriched uranium. There is no change to the composition of the fuel pellets containing depleted uranium material (i.e., UO2) except a slight decrease in the amount of ^{235}U. Therefore, the use of depleted uranium in the fuel rods does not affect the mechanical performance of the fuel rods. The impact of the use of depleted uranium on core performance is included in the reload licensing analysis. Also, NRC approved methods contained in T.S. Section 5.6.5.b do not prohibit or restrict the use of depleted uranium. The use of depleted uranium was previously addressed for Unit 2 in Reference 2.

4.3 Changes to COLR References

The changes to the references in Section 5.6.5.b remove references no longer required since 9x9-2 fuel and the ANFB critical power correlation are no longer used in Unit 1. One reference is added which contains FANP NRC approved methodology. This reference is added since the methodology provides the basis for power distribution uncertainties applied to the MCPR Safety limit analysis. The reference also enables use of this methodology for core physics analysis to support U1C14 and future reloads.

4.4 Conclusion

Unit 1 Cycle 14 will contain depleted uranium as well as natural and slightly enriched uranium in some of the fuel rods. The change to Section 4.2.1 reflects the use of depleted uranium. The use of depleted uranium in the fuel rods does not affect the mechanical performance of the fuel rods and is included in the reload licensing analysis.

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 14.

The proposed change to the MCPR Safety Limits 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 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 U1C14 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 U1C14 COLR. Thus, the COLR 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 is not adversely impacted by operation of SSES as proposed.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

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 MCPR Safety Limits does not directly or indirectly affect any plant system, equipment, component, or change the processes used to operate the plant. Further, the U1C14 MCPR Safety Limits are generated using NRC approved methodology and meet the applicable acceptance criteria. Thus, this proposed amendment does not involve a significant increase in the probability of occurrence of an accident previously evaluated.

Prior to the startup of U1C14, 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 proposed herein to generate the MCPR operating limits in the U1C14 COLR. These limits could be different from those specified for the U1C13 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 U1C14 and the results shown to be within the NRC approved criteria.

The U1C14 reload fuel bundles will utilize a small amount of depleted uranium in certain fuel rods, in addition to natural and slightly enriched uranium. There is no change to the composition of the fuel pellets containing depleted uranium material (i.e., UO_2) except a slight decrease in the amount of ^{235}U. Therefore, the use of depleted uranium in the fuel rods does not affect the mechanical performance of the fuel rods. The depleted uranium was modeled in the approved design and licensing methodology.

The changes to the references in Section 5.6.5.b were made to properly reflect the NRC approved methodology used to generate the U1C14 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 MCPR Safety Limits 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 changes do not create the possibility of a previously unevaluated operator error or a new single failure.

The use of depleted uranium in the fuel rods does not affect the mechanical performance of the fuel rods.

The changes to the references in Section 5.6.5.b were made to properly reflect the NRC approved methodology used to generate the U1C14 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 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 U1C14 preserve the required margin of safety.

The use of depleted uranium in the fuel rods does not affect the mechanical performance of the fuel rods.

The changes to the references in Section 5.6.5.b were made to properly reflect the NRC approved methodology used to generate the U1C14 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. Letter, T. H. Essig (USNRC) to H. D. Curet (SPC), "Acceptance for Referencing of Licensing Topical Reports EMF-1997(P), Revision 0," "ANFB-10 Critical Power Correlation, and EMF-1997(P), Supplement 1, Revision 0," "ANFB-10 Critical Power Correlation: High Local Peaking Results," dated July 17, 1998.

2. Letter, R. V. Guzman (USNRC) to B. L. Shriver (PPL), "Susquehanna Steam Electric Station, Unit 2 – Issuance of Amendment Regarding minimum Critical Power Ratio Safety Limits and Reference Changes (TAC NO. MB5610)" (Issuance of Amendment 184 (Unit 2), dated March 4, 2003.

Attachment 1 to PLA-5638

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:

MCPR shall be ≥ ~~1.12~~ 1.08 for two recirculation loop operation or ≥ ~~1.13~~ 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.

4.0 DESIGN FEATURES

4.1 Site Location

4.1.1 Exclusion Area Boundaries

The exclusion area shall be as shown in Figure 4.1-1.

4.1.2 Low Population Zone

The low population zone shall be as shown in Figure 4.1-2.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 764 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy fuel rods with an initial composition of depleted, natural, or slightly enriched uranium dioxide (UO_2) as fuel material, and water rods or water channels. Limited substitutions of zirconium alloy filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with NRC staff approved codes and methods and have been shown by tests or analyses to comply with all safety design bases. A limited number of lead use assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 185 cruciform shaped control rod assemblies. The control material shall be boron carbide and/or hafnium metal as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

(continued)

5.6 Reporting Requirements (continued)

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.

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. ANF-1125(P)(A), "ANFB Critical Power Correlation."~~

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

~~7. PL-NF-94-005-P-A, "Technical Basis for SPC 9x9-2 Extended Fuel Exposure at Susquehanna SES.";~~

6. ~~8.~~ 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. ~~9.~~ ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model.

~~10. XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors.~~

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

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

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

11. ~~14.~~ 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. ~~15.~~ EMF-85-74(P)(A), "RODEX 2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model."

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

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)

Attachment 2 to PLA-5638

Changes to TS Bases Pages
(Markups)

(Unit 1)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

minimum bundle flow is approximately 30 x 10^3 lb/hr. For the SPC ATRIUM-10 design, the minimum bundle flow is > 28 x 10^3 lb/hr. For both the SPC 9x9-2 and ATRIUM-10 fuel designs, the coolant minimum bundle flow and maximum area are such that the mass flux is always > .25 x 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 x 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 ~~ANFB~~ critical power correlation. References 2, 4 and 5 describe~~s~~ the methodology used in determining the MCPR SL.

The ~~ANFB and~~ ANFB-10 critical power correlation~~s~~ ~~are~~ 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 and~~ ANFB-10 correlation~~s~~ 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.

~~SPC 9x9-2 fuel is monitored using the ANFB Critical Power Correlation, and the~~ 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.
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)

Attachment 3 to PLA-5638

Proposed Technical Specification Changes (Camera Ready)

(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:

MCPR shall be ≥ 1.08 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.

4.0 DESIGN FEATURES

4.1 Site Location

4.1.1 Exclusion Area Boundaries

The exclusion area shall be as shown in Figure 4.1-1.

4.1.2 Low Population Zone

The low population zone shall be as shown in Figure 4.1-2.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 764 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy fuel rods with an initial composition of depleted, natural, or slightly enriched uranium dioxide (UO_2) as fuel material, and water rods or water channels. Limited substitutions of zirconium alloy filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with NRC staff approved codes and methods and have been shown by tests or analyses to comply with all safety design bases. A limited number of lead use assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 185 cruciform shaped control rod assemblies. The control material shall be boron carbide and/or hafnium metal as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

(continued)

5.6 Reporting Requirements (continued)

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.

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

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)

Attachment 4 to PLA-5638

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 5 to PLA-5638

Unit 1 Cycle 14 Core Composition

Unit 1 Cycle 14 Core Composition

Assembly Type	Operational History	Number of Assemblies
FANP ATRIUM™-10	Fresh	276
FANP ATRIUM™-10	Once-burned	316
FANP ATRIUM™-10	Twice-burned	172