

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of

PP&L, INC.

Docket No. 50-387

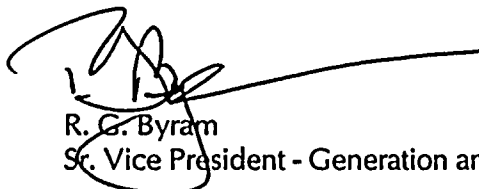
REVISED PROPOSED AMENDMENT NO. 227
FACILITY OPERATING LICENSE NO. NPF-14
SUSQUEHANNA STEAM ELECTRIC STATION
UNIT NO. 1

Licensee, PP&L, Inc., hereby files proposed Amendment No. 227 to its Facility Operating License No. NPF-14 dated July 17, 1982.

This amendment contains a revision to the Susquehanna SES Unit 1 Technical Specifications.

PP&L, INC.

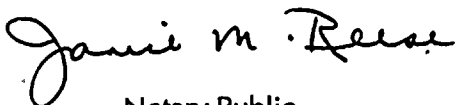
BY:



R. G. Byram

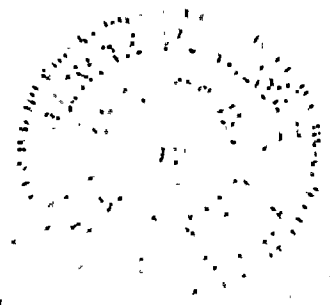
Sr. Vice President - Generation and Chief Nuclear Officer

Sworn to and subscribed before me
this 12th day of March, 1999.

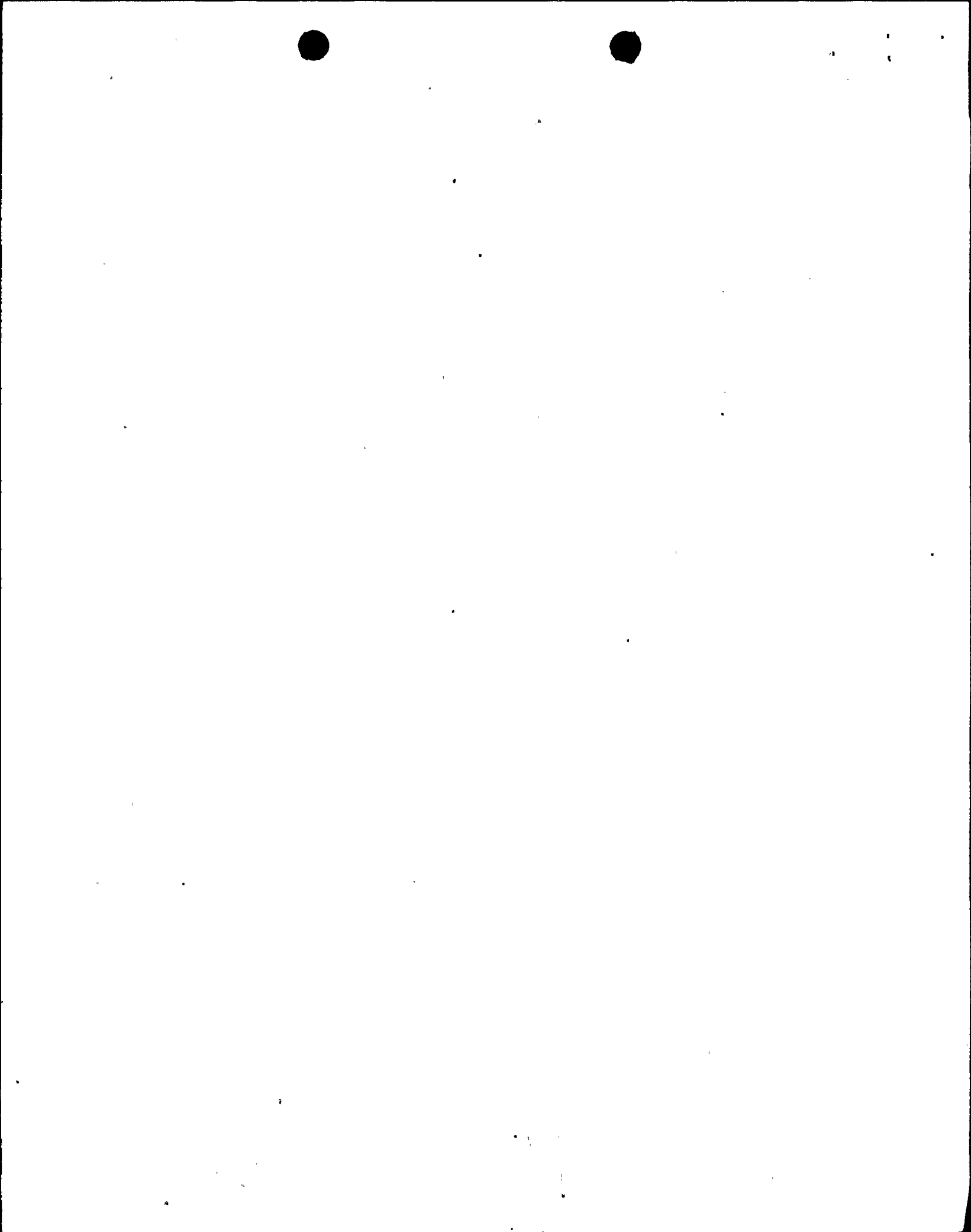


Notary Public

NOTARIAL SEAL
JANICE M. REESE, Notary Public
City of Allentown, Lehigh County, PA
My Commission Expires June 11, 2001

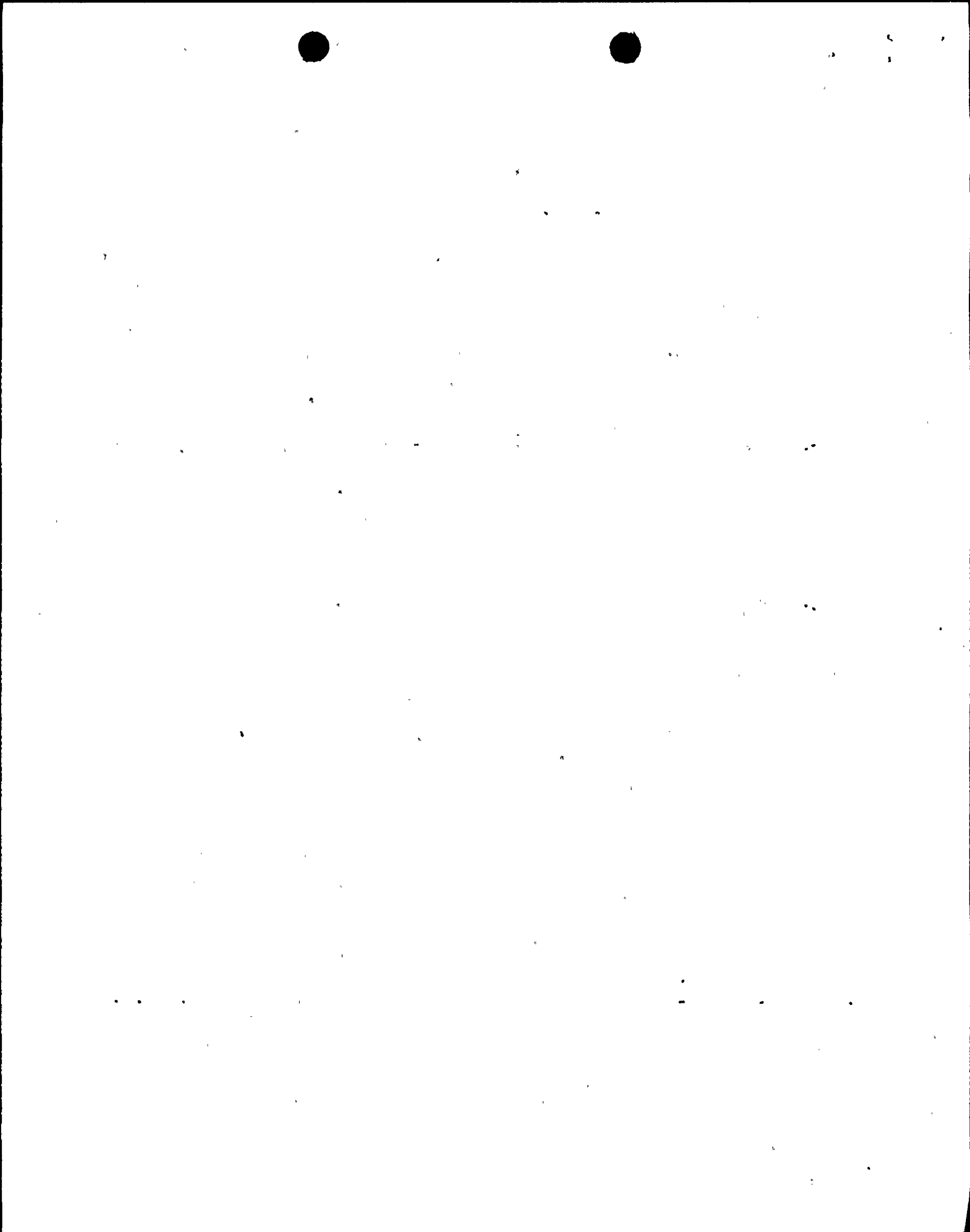


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ENCLOSURE A TO PLA-5040

SAFETY ASSESSMENT



SAFETY ASSESSMENT

ANFB-10 CRITICAL POWER CORRELATION AND MCPR SAFETY LIMITS

BACKGROUND

Susquehanna Steam Electric Station Unit 1, Cycle 12 and future cycles will contain SPC ATRIUM™-10 fuel. The ATRIUM™-10 fuel design is a 10x10 lattice design that contains 83 full length fuel rods, 8 part length fuel rods, and a central water channel.

Siemens Power Corporation has developed the ANFB-10 correlation, which is applicable to the ATRIUM™-10 fuel assemblies (Reference 1). ANFB-10 is based on a large amount of critical power test data on the ATRIUM™-10 design. The ANFB-10 correlation represents the critical power performance of ATRIUM™-10 more accurately than the original ANFB correlation (Reference 2).

Similar to our recent approved amendment on Unit 2, PP&L proposes to replace the ANFB correlation and Reference 4 and 5 methodology with the ANFB-10 correlation for analyzing ATRIUM™-10 fuel. Revised MCPR Safety Limits were generated using SPC's NRC-approved methodology described in Reference 6. These methodologies will be used each cycle to calculate the Unit 1 Safety Limits. MCPR operating limits for each Unit 1 reload cycle will be generated using the ANFB-10 correlation for ATRIUM™-10 fuel. These operating limits will be included in the cycle specific Core Operating Limits Report (COLR). NRC approval of the proposed changes is required to support use of the ANFB-10 correlation for Unit 1.

Description of the Proposed Change

The proposed Unit 1 Technical Specification change consists of:

- (1) Replacement of Figures 2.1.1.2-1 and 2.1.1.2-2, including the footnote indicating that the MCPR Safety Limit is only approved for Unit 1 Cycle 11, with single value MCPR Safety Limits in Section 2.1.1.2,
- (2) Removal of a number of items from Section 5.6.5b. in order to include only those references which directly support the generation of Core Operating Limits,
- (3) Removal of items 4 and 5 from Section 5.6.5b. of the Technical Specifications, which were previously included to address the application of ANFB to ATRIUM™-10 fuel,
- (4) Removal of item 20 for the Lead Use Assemblies from Section 5.6.5b. of the Technical Specifications since the ABB LUAs will be discharged at the end of cycle 11,
- (5) Inclusion of the Siemens Power Corporation (SPC) ANFB-10 topical report (Reference 1) in Section 5.6.5b., and



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- (6) Changes to the BASES section to reflect the inclusion of the ANFB-10 critical power correlation.

Reference 1, which describes the approved ANFB-10 methodology, plus the other NRC approved topical reports contained in Section 5.6.5 of the Technical Specifications contain methodology which will be used to ensure safe operation of Unit 1 with ATRIUM™-10 fuel.

SAFETY ANALYSIS

This section discusses the safety implications of the proposed action.

Changes to MCPR Safety Limits (Section 2.1.1.2)

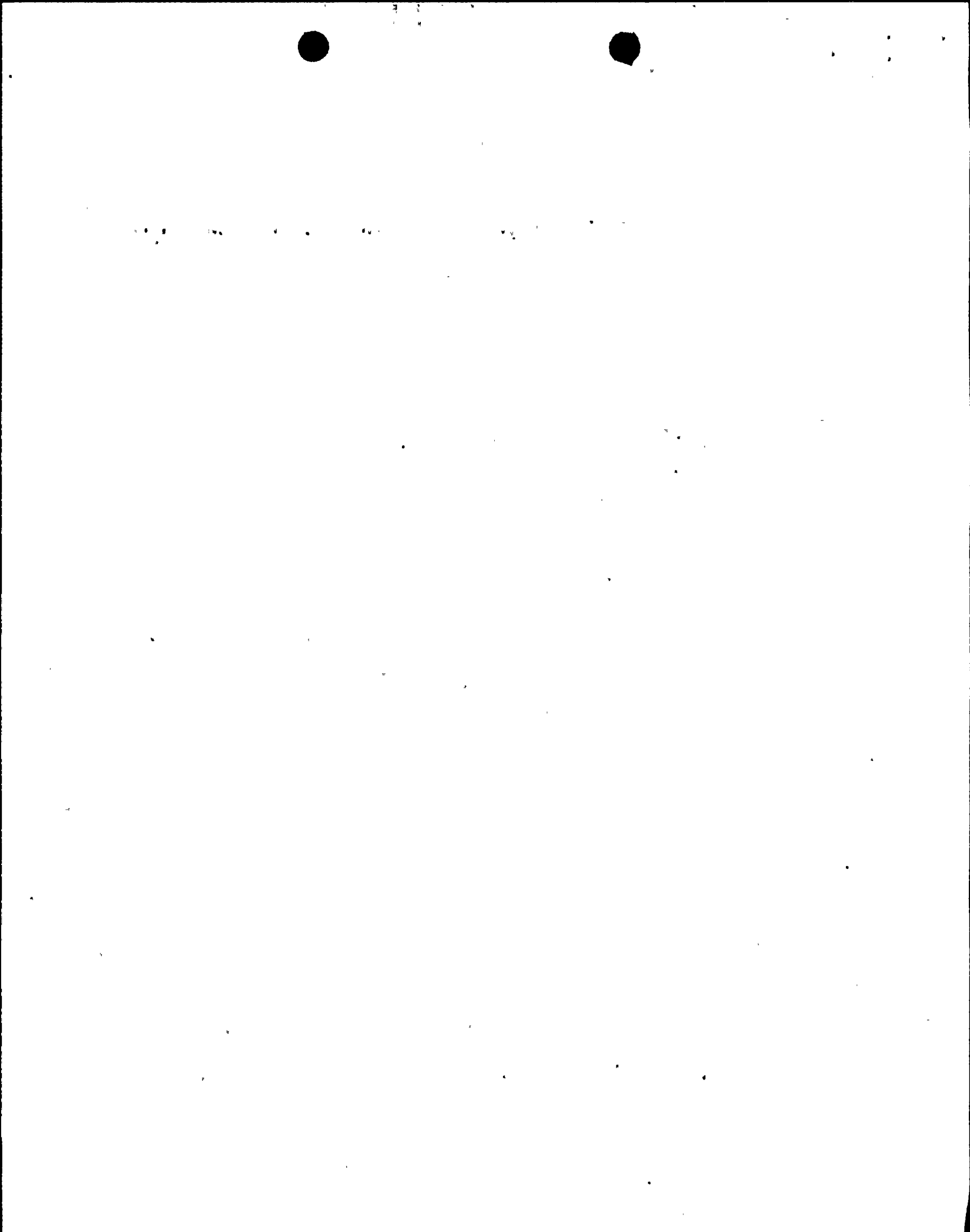
Excessive thermal overheating of the fuel rod cladding can result in cladding damage and the release of fission products. In order to protect the cladding against thermal overheating due to boiling transition, the THERMAL POWER, High Pressure and High Flow SAFETY LIMITs (Sections 2.1.1.2 of the Susquehanna SES Unit 1 Technical Specifications) were established.

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. 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, and uncertainty in the Critical Power Correlation.

The proposed SAFETY LIMIT MCPR values (two-loop and single-loop) were calculated using SPC's NRC approved licensing methods with the exception that the NRC approved ANFB-10 correlation is used in place of the ANFB correlation for ATRIUM™-10 fuel. The ANFB correlation will continue to be used for the 9x9-2 fuel.

Since the ANFB-10 correlation does not contain a flow dependence in the prediction of critical power, the new MCPR Safety Limits are not functions of core flow.

The proposed SAFETY LIMIT MCPRs (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.



Addition and Deletion of Methodology References (Section 5.6.5)

For Unit 1 Cycle 11, references 5 and 6 documented a conservative methodology for applying the ANFB critical power correlation to ATRIUM™-10 fuel at Susquehanna. This methodology conservatively accounted for a flow dependence in ANFB's ability to predict critical power for ATRIUM™-10 fuel. The impact on both MCPR Safety Limits and AOOs of ANFB's flow dependence in critical power prediction were also addressed. References 5 and 6 are being removed from the list of methodology references for Unit 1, and the ANFB-10 correlation is being added. ANFB-10 will be used to determine the critical power performance of the ATRIUM™-10 assemblies.

Reference 1 documents the ANFB-10 critical power correlation intended for use on the ATRIUM™-10 assemblies. The NRC-approved ANFB-10 topical Report (Reference 1) plus the other NRC-approved topical reports in Section 5.6.5 of the Technical Specifications contain methodology which will be used to generate Core Operating Limits for Unit 1 with ATRIUM™-10 fuel.

Other references are being removed in order to include only those references that directly support the generation of Core Operating Limits. Because the ABB LUAs will be discharged at end of cycle 11, Reference 7, that supported the generation of limits for the ABB LUAs, has also been removed. No new analysis approaches are used due to these changes.

BASES Changes

BASES Sections 2.1.1.1, 2.1.1.2, and 3.2.2 are changed to reflect the use of the ANFB-10 correlation. The range of the applicability of the ANFB-10 correlation is valid for pressures > 571 psia and bundle mass fluxes $> 0.115 \times 10^6$ lb/hr-ft². These values assure that a valid CPR calculation will result at or above 25% of rated core thermal power, that is, reactor steam dome pressure ≥ 785 psig and core flow ≥ 10 Mlbm/hr.

BASES Sections 3.2.1, 3.2.2, 3.2.3, and 3.2.4 are changed to remove the reference (i.e., Reference 7) for the ABB LUAs, since the four LUAs will be discharged from Unit 1 during the Unit 1 11th Refueling and Inspection Outage.

CONCLUSIONS

The proposed changes to the Susquehanna SES Unit 1 Technical Specifications support the use of the ANFB-10 correlation for analyzing ATRIUM™-10 fuel.



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NRC approval of the proposed Technical Specification changes will ensure that the methodologies contained in Technical Specification Section 5.6.5 can be used in support of ATRIUM™-10 fuel at Susquehanna Unit 1.

REFERENCES

1. 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.
2. ANF-1125 (P)(A) and ANF-1125 (P)(A), Supplement 1, "ANFB Critical Power Correlation," April 1990.
3. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," July 1992, plus Supplements 1-A (August 1995) and 2-A (July 1996).
4. EMF-97-010(P), Rev. 1, "Application of ANFB to ATRIUM™-10 for Susquehanna Reloads", March 1997.
5. PLA-4595, "Response to NRC Request for Additional Information on Siemens' Report EMF-97-010, Revision 1," March 27, 1997.
6. ANF-524(P)(A), Revision 2 and Supplement 1, Revision 2, "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors," November 1990.
7. CENPD-300-P, "Reference Safety Report for Boiling Water Reactor Reload Fuel," ABB Combustion Engineering Nuclear Operations, November 1994.

ENCLOSURE B TO PLA-5040

**NO SIGNIFICANT HAZARDS CONSIDERATIONS AND
ENVIRONMENTAL ASSESSMENT**



**NO SIGNIFICANT HAZARDS CONSIDERATIONS AND ENVIRONMENTAL
ASSESSMENT**

**ANFB-10 CRITICAL POWER CORRELATION AND MCPR SAFETY
LIMITS**

PP&L has evaluated the proposed Technical Specification change in accordance with the criteria specified by 10 CFR 50.92 and has determined that the proposed change does not involve a significant hazards consideration. The criteria and conclusions of our evaluation are presented below.

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The applicable sections of the FSAR are Chapters 4.4 and 15. FSAR Chapter 4.4 describes the MCPR Safety Limit, and Chapter 15 describes the transient and accident analyses. The reference to be added to Section 5.6.5 of the Unit 1 Technical Specifications describes a NRC approved critical power correlation for ATRIUM™-10 fuel. This correlation is appropriate for use in conservative methodologies for generating MCPR Safety Limits and MCPR Operating Limits to assure safe operation of Unit 1 with ATRIUM™-10 fuel. A discussion of the impact of the proposed Technical Specification change is provided below.

The proposed change in critical power correlation does not physically affect the plant or its systems. Thus, it does not increase the probability of an accident previously evaluated.

A Unit 1 Cycle 12 MCPR Safety Limit analysis was performed for PP&L by SPC. This analysis used NRC approved methods described in ANF-524(P)(A), Revision 2 and Supplement 1 Revision 2. These methods will be used each cycle to calculate the Unit 1 Safety Limits. For Unit 1 Cycle 12, the critical power performance of the 9x9-2 and ATRIUM™-10 fuel was determined using the NRC approved ANFB and ANFB-10 correlations, respectively. The SAFETY LIMIT MCPR calculations statistically combine uncertainties on feedwater flow, feedwater temperature, core flow, core pressure, core power distribution, and uncertainties in the Critical Power Correlation. The SPC analysis used cycle specific power distributions and calculated MCPR values such that at least 99.9% of the fuel rods are expected to avoid boiling transition during normal operation or anticipated operational occurrences. The resulting two-loop and single-loop MCPR Safety Limits are included in the proposed Technical Specification change. Thus, the cladding integrity and its ability to contain fission products are not adversely affected.

Analyses of the Single Loop Pump Seizure accident with the NRC approved ANFB-10 correlation for ATRIUM™-10 fuel (Reference 1) will be performed to demonstrate that the NRC acceptance criterion (i.e., small fraction of 10CFR100 dose limits) is met. Analyses will also be performed to validate the conclusion that two-loop transients are more severe than those events analyzed in single-loop operation.

Changes to Section 2.1.1.2 reflect the change from a flow dependent MCPR Safety Limit to a single value MCPR Safety Limit for two-loop operation and single-loop operation.

Changes to Reference 5.6.5 delete the methodology used for critical power analyses for ATRIUM™-10 fuel and add the NRC approved ANFB-10 methodology to the list of approved methodologies. Other changes in Reference 5.6.5 are administrative in nature because they delete references not directly related to the generation of Core Operating Limits. No new analysis approaches are used due to these changes.

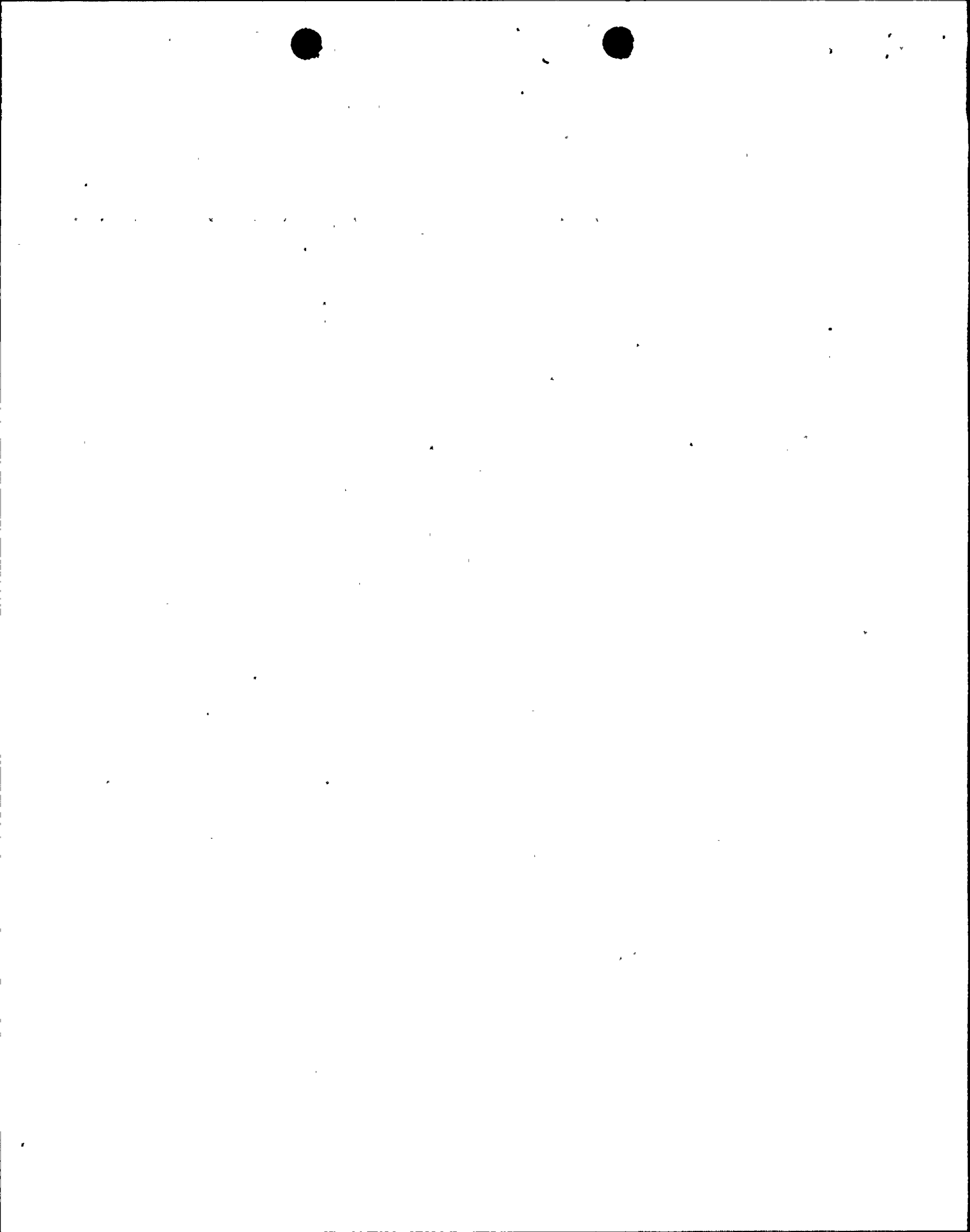
Changes to BASES Sections 2.1.1 and 3.2.2 reflect the inclusion of the ANFB-10 critical power correlation. The range of the applicability of the ANFB-10 is valid for pressures > 571 psia and bundle mass fluxes $> 0.115 \times 10^6$ lb/hr-ft². These values assure that a valid CPR calculation will result at or above 25% of rated core thermal power, that is, reactor steam dome pressure ≥ 785 psig and core flow ≥ 10 Mlbm/hr.

Changes to BASES Sections 3.2.1, 3.2.2, 3.2.3, and 3.2.4 reflect the removal of Reference 7 for the ABB LUAs, since the four LUAs will be discharged from Unit 1 during the Unit 1 11th Refueling and Inspection Outage.

The consequences of transients and accidents will remain within the criteria approved by the NRC. The methodology used to perform the analyses has been previously approved by the NRC. Thus, analysis results using the new methodology will continue to provide assurance that the reactor will perform its design safety function during normal operation and design basis events. Therefore, the proposed action does not involve an increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to the Unit 1 Technical Specifications (MCPR Safety Limits, removal of methodology references not directly supporting the generation of Core Operating Limits, removal of the two references describing previously approved methodology for applying ANFB to ATRIUM™-10 fuel, removal of the ABB LUA reference, and inclusion of the ANFB-10 correlation reference) do not require any physical plant modifications, physically affect any plant components, or entail changes in plant operation. Removal of the Unit 1 Cycle 11 footnote allows Unit 1 Cycle 12 and future cycle operation with NRC approved methodology. Thus, the proposed change



does not create the possibility of a previously unevaluated operator error or a new single failure. The consequences of transients and accidents will remain within the criteria approved by the NRC. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

The applicable Technical Specification Sections include 2.1.1.2 and 5.6.5.

The changes to the Unit 1 Technical Specifications discussed in Item 1 above do not require any physical plant modifications, physically affect any plant components, or entail changes in plant operation. Therefore, the proposed change will not jeopardize or degrade the function or operation of any plant system or component governed by Technical Specifications. The consequences of transients and accidents will remain within the criteria approved by the NRC. The proposed MCPR Safety Limits and use of the ANFB-10 critical power correlation described in the reference added to Section 5.6.5 do not involve a significant reduction in the margin of safety as currently defined in the Bases of the applicable Technical Specification sections.

Therefore, the proposed change does not involve a significant reduction in the margin of safety.

ENVIRONMENTAL CONSEQUENCES

An environmental assessment is not required for the proposed change because the requested change conforms to the criteria for actions eligible for categorical exclusion as specified in 10 CFR 51.22(c)(9). The requested change will have no impact on the environment. The proposed change does not involve a significant hazards consideration as discussed above. The proposed change does not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite. In addition, the proposed change does not involve a significant increase in the individual or cumulative occupational radiation exposure.

