Kevin Cimorelli
Site Vice President

Susquehanna Nuclear, LLC
769 Salem Boulevard
Berwick, PA 18603
Tel. 570.542.3795 Fax 570.542.1504
Kevin.Cimorelli@TalenEnergy.com

January 5, 2022

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

10 CFR 50.90

**SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED AMENDMENT TO LICENSES NPF-14
AND NPF-22: APPLICATION TO REVISE
TECHNICAL SPECIFICATIONS TO ADOPT
TSTF-564, "SAFETY LIMIT MCPR"
PLA-7962**

**Docket No. 50-387
and 50-388**

Pursuant to 10 CFR 50.90, Susquehanna Nuclear, LLC (Susquehanna), is submitting a request for an amendment to the Technical Specifications (TS) for the Susquehanna Steam Electric Station (SSES), Units 1 and 2, Facility Operating License Numbers NPF-14 and NPF-22.

Susquehanna requests the adoption of TSTF-564, "Safety Limit MCPR," Revision 2, which is an approved change to the Improved Standard Technical Specifications (ISTS), into the SSES, Units 1 and 2 TS. The proposed amendment revises the TS safety limit (SL) on minimum critical power ratio (MCPR) to reduce the need for cycle-specific changes to the value while still meeting the regulatory requirement for an SL.

Attachment 1 provides a description and assessment of the proposed changes. Attachment 2 provides the existing TS pages marked to show the proposed changes. Attachment 3 provides revised (clean) TS pages. Attachment 4 provides existing TS Bases pages marked to show the proposed changes for information only.

Attachment 5 contains a Framatome report which provides the calculation determining the appropriate safety limit MCPR (i.e., $SLMCPR_{95/95}$) for the ATRIUM 10 and ATRIUM 11 fuel types. This attachment contains information considered proprietary to Framatome, denoted by brackets. As owner of the proprietary information, Framatome has executed the affidavit contained in Attachment 7 which identifies the information as proprietary, is customarily held in confidence, and should be withheld from public disclosure pursuant to 10 CFR 2.390. Attachment 6 contains the non-proprietary version of this information.

Although not part of the consolidated line items improvement process (CLIIP), Susquehanna requests review and approval by July 31, 2022 based on the similarities between this application and other previously approved by the NRC. Once approved, the amendment shall be implemented no later than startup from the Unit 2 refueling outage in spring 2023 for Unit 2 and startup from the Unit 1 refueling outage in spring 2024 for Unit 1.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated Commonwealth of Pennsylvania state official.

Both the Plant Operations Review Committee and the Nuclear Safety Review Board have reviewed the proposed changes.

There are no new or revised regulatory commitments contained in this submittal.

Should you have any questions regarding this submittal, please contact Ms. Melisa Krick, Manager – Nuclear Regulatory Affairs, at (570) 542-1818.

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

Executed on: January 5, 2022

K. Cimorelli

Attachments:

1. Description and Assessment
2. Marked-Up Technical Specification Changes
3. Revised (Clean) Technical Specification Pages
4. Marked-Up Technical Specification Bases Changes (Mark-Up) for Information Only
5. Framatome Report ANP-3857P, Revision 2, "Design Limits for Framatome Critical Power Correlations" **[Proprietary Information – Withhold from Public Disclosure Pursuant to 10 CFR 2.390]**
6. Framatome Report ANP-3857NP, Revision 2, "Design Limits for Framatome Critical Power Correlations" [Non-Proprietary Version]
7. Framatome Affidavit for ANP-3857P, Revision 2, "Design Limits for Framatome Critical Power Correlations"

Copy: NRC Region I
Mr. C. Highley, NRC Senior Resident Inspector
Ms. A. Klett, NRC Project Manager
Mr. M. Shields, PA DEP/BRP

Attachment 1 to PLA-7962

Description and Assessment

SUSQUEHANNA ASSESSMENT

1. Description

Susquehanna Nuclear, LLC (Susquehanna), requests adoption of TSTF-564, "Safety Limit MCPR," Revision 2, which is an approved change to the Improved Standard Technical Specifications (ISTS), into the Susquehanna Steam Electric Station (SSES), Units 1 and 2 Technical Specifications (TS). The proposed amendment revises the TS safety limit (SL) on minimum critical power ratio (MCPR) to reduce the need for cycle-specific changes to the value while still meeting the regulatory requirement for an SL.

2. Assessment

2.1 Applicability of Safety Evaluation

Susquehanna has reviewed the safety evaluation for TSTF-564 provided to the Technical Specifications Task Force in a letter dated May 13, 2020. This review included a review of the NRC staff's evaluation, as well as the information provided in TSTF-564. Susquehanna has concluded that the justifications presented in TSTF-564 and the safety evaluation prepared by the NRC staff are applicable to SSES, Units 1 and 2 and justify this amendment for the incorporation of the changes to the SSES TS.

The SSES Unit 1 reactor is currently fueled with Framatome ATRIUM 10 fuel bundles and ATRIUM 11 fuel bundles will be introduced during the spring 2022 refueling outage. The SSES Unit 2 reactor is currently fueled with Framatome ATRIUM 10 and ATRIUM 11 fuel bundles. The proposed Safety Limit in SL 2.1.1.2 is 1.05 which is the most limiting value for ATRIUM 10 and ATRIUM 11 fuel types. ATRIUM 11 is identified in the TS Bases as the fuel type the SL is based upon since it is most limiting and will be the dominant fuel type at SSES going forward.

Attachment 5 contains Framatome report ANP-3857P, Revision 2, "Design Limits for Framatome Critical Power Correlations." This report provides details of the calculation of the $MCPR_{95/95}$ for ATRIUM 10 using the statistics from the SPCB/ATRIUM 10 CPR correlation database contained in Reference 1. This report also provides the details of the calculation of the $MCPR_{95/95}$ for ATRIUM 11 using the statistics from the ACE/ATRIUM 11 CPR correlation database contained in Reference 2.

Susquehanna is currently informed of any error that impacts the SPCB/ACE CPR correlations in accordance with Framatome's 10 CFR 50 Appendix B program. This existing process ensures that any potential errors in the SPCB/ACE CPR correlations will be addressed as they relate to the $MCPR_{95/95}$ for SL 2.1.1.2 following implementation of TSTF-564.

The MCPR value calculated as the point at which 99.9% of the fuel rods would not be susceptible to boiling transition (i.e., reduced heat transfer) during normal operation and anticipated operational occurrences is referred to as $MCPR_{99.9\%}$. Technical Specification 5.6.5, "Core Operating Limits Report (COLR)," is revised to require the $MCPR_{99.9\%}$ value to be included in the cycle-specific COLR. $MCPR_{99.9\%}$ will continue to be calculated using Reference 3.

2.2 Variations

Susquehanna is proposing the following variations from the TS changes described in TSTF-564 or the applicable parts of the NRC staff's safety evaluation.

The SSES TS specify a different reactor steam dome pressure value (575 psig) in Specifications 2.1.1.1 and 2.1.1.2, rather than the value of 785 psig specified in the ISTS. This is a plant-specific value and does not affect the applicability of TSTF-564 to SSES.

SSES uses Framatome ATRIUM 10 and ATRIUM 11 fuel types which are not identified in TSTF-564 Table 1. However, as discussed in TSTF-564, other fuel vendors may determine the $MCPR_{95/95}$ for other fuel designs using methodology described in TSTF-564. Framatome licensing report ANP-3857P (Attachment 5) includes the required derivation of the $MCPR_{95/95}$ for the ATRIUM 10 and ATRIUM 11 fuel types based on the information contained in the respective NRC approved CPR correlations (Reference 1 and Reference 2). The conclusions in Attachment 5 that the TSTF-564 $MCPR_{95/95}$ formulation remains applicable with Framatome's definition of Experimental CPR (ECPR) have been accepted by Susquehanna. This difference is within the scope of the TSTF-564 approval and does not affect the applicability of TSTF-564 to the SSES TS.

The SSES TS utilize different numbering than the ISTS on which TSTF-564 was based. Specifically, SSES TS 5.6.5, "Core Operating Limits Report (COLR)" corresponds to ISTS 5.6.3, "Core Operating Limits Report." This difference is administrative and does not affect the applicability of TSTF-564 to the SSES TS.

3. <u>Regulatory Analysis</u>

3.1 Precedent

The NRC has approved a similar amendment request for implementation of TSTF-564 for another station utilizing Framatome ATRIUM Fuel. Duke Energy Progress, LLC, submitted on March 9, 2020, an application to revise the Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Technical Specifications requesting the adoption of TSTF-564, “Safety Limit MCPR,” (Reference 4). The Brunswick nuclear power plants are fueled with Framatome ATRIUM 10XM fuel type and the ATRIUM 11 fuel type was indicated as expected to be introduced in the spring of 2020. This was approved by the NRC on September 29, 2020 (Reference 5).

3.2 No Significant Hazards Considerations Analysis

Susquehanna Nuclear, LLC (Susquehanna), requests adoption of TSTF-564, "Safety Limit MCPR," which is an approved change to the Improved Standard Technical Specifications (ISTS), into the SSES Unit 1 and 2 Technical Specifications (TS). The proposed change revises the TS safety limit on minimum critical power ratio (SLMCPR). The revised limit calculation method is based on using the Critical Power Ratio (CPR) data statistics and is revised from ensuring that 99.9% of the rods would not be susceptible to boiling transition to ensuring that there is a 95% probability at a 95% confidence level that no rods will be susceptible to transition boiling. A single SLMCPR value will be used instead of two values applicable when one or two recirculation loops are in operation. TS 5.6.5, "Core Operating Limits Report (COLR)," is revised to require the current cycle specific SLMCPR value (i.e., $MCPR_{99.9\%}$) to be included in the COLR.

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

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

   Response: No

   The proposed amendment revises the TS SLMCPR and the list of core operating limits to be included in the COLR. The SLMCPR is not an initiator of any accident previously evaluated. The revised safety limit values continue to ensure for all accidents previously evaluated that the fuel cladding will be protected from failure due to transition boiling. The proposed change does not affect plant operation or any procedural or administrative controls on plant operation that affect the functions of preventing or mitigating any accidents previously evaluated.

   Therefore, the proposed change does not involve a significant increase in the probability 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 proposed amendment revises the TS SLMCPR and the list of core operating limits to be included in the COLR. The proposed change will not affect the design function or operation of any structures, systems or components (SSCs). No new equipment will be installed. As a result, the proposed change will not create any credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing bases.

   Therefore, the proposed change 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

   The proposed amendment revises the TS SLMCPR and the list of core operating limits to be included in the COLR. This will result in a change to a safety limit, but will not result in a significant reduction in the margin of safety provided by the safety limit. As discussed in the application, changing the SLMCPR methodology to one based on a 95% probability with 95% confidence that no fuel rods experience transition boiling during an anticipated transient instead of the current limit based on ensuring that 99.9% of the fuel rods are not susceptible to boiling transition does not have a significant effect on plant response to any analyzed accident. The SLMCPR and the TS Limiting Condition for Operation (LCO) on MCPR continue to provide the same level of assurance as the current limits and do not reduce a margin of safety.

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

Based on the above, Susquehanna concludes that the proposed change 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.

3.2 Conclusion

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.

4. Environmental Consideration

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

5. References

1. AREVA Topical Report EMF-2209(P)(A), “SPCB Critical Power Correlation.”
2. Framatome Topical Report ANP-10335P-A, “ACE/ATRIUM-11 Critical Power Correlation.”
3. AREVA Topical Report ANP-10307PA, “AREVA MCPR Safety Limit Methodology for Boiling Water Reactors.”
4. Duke Energy Progress, LLC, “Application to Revise Technical Specification to Adopt TSTF 564, “Safety Limiy MCPR,” Dated March 9, 2020 (ADAMS Accession No. ML20070H939)
5. NRC letter to Duke Energy Progress LLC, “Brunswick Steam Electric Plant, Units 1 and 2 – Issuance of Amendment Nos. 301 and 329 to Revise Technical Specification to Adopt TSTF-564 (EPID L-1010-LLA-0043).” Dated September 29, 2020 (ADAMS Accession No. ML20269A305).

Attachment 2 of PLA-7962

Marked-Up Technical Specification Changes

Revised Technical Specifications Pages

Unit 1 TS Pages
2.0-1, 5.0-21

Unit 2 TS Pages
2.0-1, 5.0-21

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SL

2.1.1.1 With the reactor steam dome pressure < 575 psig or core flow < 10 million lbm/hr:

THERMAL POWER shall be $\leq$ 23% RTP.

2.1.1.2 With the reactor steam dome pressure $\geq$ 575 psig and core flow $\geq$ 10 million lbm/hr:

MCPR shall be $\geq$ 1.05~~1.09 for two recirculation loop operation or ≥ 1.12 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 $\leq$ 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

5.6.4 Not Used

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

1. The Average Planar Linear Heat Generation Rate for Specification 3.2.1;

2. The Minimum Critical Power Ratio (MCPR) and $MCPR_{99.9\%}$ for Specification3.2.2;

3. The Linear Heat Generation Rate for Specification 3.2.3;

4. The Shutdown Margin for Specification 3.1.1;

5. Oscillation Power Range Monitor (OPRM) Trip Setpoints, for Specification 3.3.1.1; and

6. The Allowable Values and power range setpoints for Rod Block Monitor Upscale Functions for Specification 3.3.2.1, Table 3.3.2.1-1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 575 psig or core flow < 10 million lbm/hr:

THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 575 psig and core flow ≥ 10 million lbm/hr:

MCPR shall be ≥ 1.05~~1.08 for two recirculation loop operation or ≥ 1.11 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

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   1. The Average Planar Linear Heat Generation Rate for Specification 3.2.1;

   2. The Minimum Critical Power Ratio (MCPR) and $MCPR_{99.9\%}$ for Specification 3.2.2;

   3. The Linear Heat Generation Rate for Specification 3.2.3;

   4. The Shutdown Margin for Specification 3.1.1;

   5. Oscillation Power Range Monitor (OPRM) Trip setpoints, for Specification 3.3.1.1; and

   6. The Allowable Values and power range setpoints for Rod Block Monitor Upscale Functions for Specification 3.3.2.1, Table 3.3.2.1-1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC.

Attachment 3 of PLA-7962

Revised (Clean) Technical Specification Pages

Revised Technical Specifications Pages

Unit 1 TS Pages
2.0-1, 5.0-21

Unit 2 TS Pages
2.0-1, 5.0-21

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SL

2.1.1.1 With the reactor steam dome pressure < 575 psig or core flow < 10 million lbm/hr:

THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 575 psig and core flow ≥ 10 million lbm/hr:

MCPR shall be ≥ 1.05.

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

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1. The Average Planar Linear Heat Generation Rate for Specification 3.2.1;

2. The Minimum Critical Power Ratio (MCPR) and $MCPR_{99.9\%}$ for Specification3.2.2;

3. The Linear Heat Generation Rate for Specification 3.2.3;

4. The Shutdown Margin for Specification 3.1.1;

5. Oscillation Power Range Monitor (OPRM) Trip Setpoints, for Specification 3.3.1.1; and

6. The Allowable Values and power range setpoints for Rod Block Monitor Upscale Functions for Specification 3.3.2.1, Table 3.3.2.1-1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 575 psig or core flow < 10 million lbm/hr:

THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 575 psig and core flow ≥ 10 million lbm/hr:

MCPR shall be ≥ 1.05.

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

5.6.4 Not Used

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

1. The Average Planar Linear Heat Generation Rate for Specification 3.2.1;

2. The Minimum Critical Power Ratio (MCPR) and $MCPR_{99.9\%}$ for Specification 3.2.2;

3. The Linear Heat Generation Rate for Specification 3.2.3;

4. The Shutdown Margin for Specification 3.1.1;

5. Oscillation Power Range Monitor (OPRM) Trip setpoints, for Specification 3.3.1.1; and

6. The Allowable Values and power range setpoints for Rod Block Monitor Upscale Functions for Specification 3.3.2.1, Table 3.3.2.1-1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC.

Attachment 4 of PLA-7962

Marked-Up Technical Specification Bases Changes

Revised Technical Specifications Bases Pages

Unit 1 TS Bases Pages
2.0-1, 2.0-2, 2.0-3, 2.0-5, 3.2-5, 3.2-6, 3.2-8, 3.2-9

Unit 2 TS Bases Pages
2.0-1, 2.0-2, 2.0-3, 2.0-5, 3.2-5, 3.2-6, 3.2-9

(Provided for Information Only)

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires, and SLs ensure, that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs).

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2 for ATRIUM 10 and ATRIUM 11 fuel. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical barriers that separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses, which occur from reactor operation significantly above design conditions.

While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross, rather than incremental, cladding deterioration. Therefore, the fuel cladding SL is defined with a margin to the conditions that would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. This is accomplished by having a Safety Limit Minimum Critical Power Ratio (SLMCPR) design basis, referred to as the $SLMCPR_{95/95}$, which corresponds to a 95% probability at a 95% confidence level (the 95/95 MCPR criterion) that transition boiling will not occur.~~The MCPR fuel cladding integrity SL ensures that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.~~

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

BASES

BACKGROUND (continued)

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 Technical Specification SL is set generically on a fuel product MCPR correlation basis as the MCPR which corresponds to a 95% probability at a 95% confidence level that transition boiling will not occur, referred to as $SLMCPR_{95/95}$.~~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 SPCB (Reference 4) correlation is valid for critical power calculations with ATRIUM 10 fuel at pressures ≥ 571.4 psia (conservatively bounded by 575 psig) and bundle mass fluxes > 0.087×10^6 lb/hr-ft^2.

The use of the ACE/ATRIUM 11 (Reference 6) correlation is valid for critical power calculations with ATRIUM 11 fuel at pressures ≥ 588.8 psia (conservatively bounded by 575 psig) with no minimum bundle mass flux.

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. For ATRIUM 10 and ATRIUM 11 fuel, the minimum bundle flow is > 28×10^3 lb/hr, and the coolant minimum bundle flow and maximum area are such that the mass flux is always > 0.24×10^6 lb/hr-ft^2. Full scale critical power test data taken from various fuel designs at pressures from 14.7 psia to 1400 psia indicate that the fuel assembly critical power at 0.24×10^6 lb/hr-ft^2 is approximately 3.35 MWt. At 23% RTP, a bundle power of

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APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity (continued)

pressures < 575 psig is conservative and for conditions of lesser power would remain 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, 5, and 6 describe the methodology used in determining the MCPR SL.~~The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. The fuel cladding integrity SL value is dependent on the fuel product line and the corresponding MCPR correlation, which is cycle independent. The value is based on the Critical Power Ratio (CPR) data statistics and a 95% probability with 95% confidence that rods are not susceptible to boiling transition, referred to as $MCPR_{95/95}$.

The SL is based on ATRIUM 11 fuel. For cores with a single fuel product line, the $SLMCPR_{95/95}$ is the $MCPR_{95/95}$ for the fuel type. For cores loaded with a mix of applicable fuel types, the $SLMCPR_{95/95}$ is based on the largest (i.e., most limiting) of the $MCPR_{95/95}$ values for the fuel product lines that are fresh or once-burnt at the start of the cycle. References 4, 6, and 7 described the methodology used in determining the $SLMCPR_{95/95}$.

The SPCB and ACE/ATRIUM 11 critical power correlations 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

BASES

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of regulatory limits. 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. ~~ANP-10307PA, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," (as identified in the COLR).~~Not used.

3. Not used.

4. EMF-2209(P)(A), "SPCB Critical Power Correlation," ~~AREVA NP,~~ (as identified in the COLR).

5. ~~EMF-2158(P)(A), "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/Microburn-B2," (as identified in the COLR).~~Not used.

6. ANP-10335P-A, "ACE/ATRIUM 11 Critical Power Correlation," (as identified in the COLR).

~~6.~~7. ANP-3857P, "Design Limits for Framatome Critical Power Correlations," Revision 2.

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), and that 99.9% of the fuel rods are not susceptible to boiling transition if the limit is not violated. 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, 3, 5, 7, and 10 for ATRIUM 10 fuel design analysis and references 2, 3, 5, ~~7,~~ 10, and 12 through 15 for ATRIUM 11 fuel designs. To ensure that the MCPR Safety Limit (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 combined with~~added to~~ the $MCPR_{99.9\%}$ ~~SL~~, the required operating limit MCPR is obtained.

$MCPR_{99.9\%}$ is determined to ensure more than 99.9% of the fuel rods in the core are not susceptible to boiling transition using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved Critical Power correlations.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Details of the $MCPR_{99.9\%}$ calculation are given in References 7, 15, and 16. References 7 and 15 also include a tabulation of the uncertainties and the nominal values of the parameters used in the $MCPR_{99.9\%}$ statistical analysis.

The MCPR operating limits are derived from the $MCPR_{99.9\%}$ value and the transient analysis, and are dependent on the operating core flow and power 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.

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

LCO

The MCPR operating limits specified in the COLR ($MCPR_{99.9\%}$ value, $MCPR_f$ values and $MCPR_p$ values) 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, which are based on the $MCPR_{99.9\%}$ limit specified in the COLR.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 23% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 23% 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 23% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 23% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.2.2 (continued)

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.

REFERENCES

1. NUREG-0562, June 1979.
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.
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. Not used.
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. Not used.
7. EMF-2209(P)(A), “SPCB Critical Power Correlation,” ~~AREVA NP,~~ (as identified in the COLR).
8. Not used.
9. Not used
10. ANF-1358(P)(A), “The Loss of Feedwater Heating Transient in Boiling Water Reactors,” ~~Framatome ANP,~~ (as identified in the

BASES

REFERENCES (continued)

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

12. ANP-10300P-A, "AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios," (as identified in the COLR).

13. BAW-10247PA, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," (as identified in the COLR)

14. BAW-10247P-A, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods," (as identified in the COLR).

15. ANP-10335P-A, "ACE/ATRIUM-11 Critical Power Correlation," (as identified in the COLR).

~~15.~~16. ANP-10307PA, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," (as identified in the COLR).

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires, and SLs ensure, that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs).

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2 for ATRIUM 10 and ATRIUM 11 fuel. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical barriers that separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses, which occur from reactor operation significantly above design conditions.

While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross, rather than incremental, cladding deterioration. Therefore, the fuel cladding SL is defined with a margin to the conditions that would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. This is accomplished by having a Safety Limit Minimum Critical Power Ratio (SLMCPR) design basis, referred to as the $SLMCPR_{95/95}$, which corresponds to a 95% probability at a 95% confidence level (the 95/95 MCPR criterion) that transition boiling will not occur.~~The MCPR fuel cladding integrity SL ensures that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.~~

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.

BASES

BACKGROUND (continued)

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 Technical Specification SL is set generically on a fuel product MCPR correlation basis as the MCPR which corresponds to a 95% probability at a 95% confidence level that transition boiling will not occur, referred to as $SLMCPR_{95/95}$. ~~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 SPCB (Reference 4) correlation is valid for critical power calculations with ATRIUM 10 fuel at pressures ≥ 571.4 psia (conservatively bounded by 575 psig) and bundle mass fluxes > 0.087×10^6 lb/hr-ft^2.

The use of the ACE/ATRIUM 11 (Reference 6) correlation is valid for critical power calculation with ATRIUM 11 fuel at pressures ≥ 588.8 psia (conservatively bounded by 575 psig) with no minimum bundle mass flux.

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. For ATRIUM 10 and ATRIUM 11 fuel, the minimum bundle flow is > 28×10^3 lb/hr and the coolant minimum bundle flow and maximum area are such that the mass flux is always > 0.24×10^6 lb/hr-ft^2. Full scale critical power test data taken from various fuel designs at pressures from 14.7 psia to 1400 psia indicate that the fuel assembly critical power at

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity (continued)

expected peaking factor. Thus, a THERMAL POWER limit of 23% RTP for reactor pressures < 575 psig is conservative and for conditions of lesser power would remain 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, 5, and 6 describe the methodology used in determining the MCPR SL.~~The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. The Technical Specification SL value is dependent on the fuel product line and the corresponding MCPR correlation, which is cycle independent. The value is based on the Critical Power Ratio (CPR) data statistics and a 95% probability with 95% confidence that rods are not susceptible to boiling transition, referred to as $MCPR_{95/95}$.

The SL is based on ATRIUM 11 fuel. For cores with a single fuel product line, the $SLMCPR_{95/95}$ is the $MCPR_{95/95}$ for the fuel type. For cores loaded with a mix of applicable fuel types, the $SLMCPR_{95/95}$ is based on the largest (i.e., most limiting) of the $MCPR_{95/95}$ values for the fuel product lines that are fresh or once-burnt at the start of the cycle. References 4, 6, and 7 described the methodology used in determining the $SLMCPR_{95/95}$.

The SPCB and ACE/ATRIUM 11 critical power correlations 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 correlation (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

BASES

SAFETY LIMIT VIOLATIONS	Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of regulatory limits. 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. ~~ANP-10307PA, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," (as identified in the COLR)~~Not used.

3. Not used.

4. EMF-2209(P)(A), "SPCB Critical Power Correlation," ~~AREVA NP,~~ (as identified in the COLR).

5. ~~EMF-2158(P)(A), "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4 / MICROBURN-B2," (as identified in the COLR)~~Not used.

6. ANP-10335P-A, "ACE/ATRIUM 11 Critical Power Correlation," (as identified in the COLR).

~~6.~~7. ANP-3857P, "Design Limits for Framatome Critical Power Correlations," Revision 2.

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), and that 99.9% of the fuel rods are not susceptible to boiling transition if the limit is not violated. 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, 3, 5, 7, and 10 for ATRIUM 10 fuel design analysis and References 2, 3, 5, ~~7,~~ 10, and 12 through 15 for ATRIUM 11 fuel designs. To ensure that the MCPR Safety Limit (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 combined with ~~added to~~ the $MCPR_{99.9\%}$ ~~SL~~, the required operating limit MCPR is obtained.

$MCPR_{99.9\%}$ is determined to ensure more than 99.9% of the fuel rods in the core are not susceptible to boiling transition using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the

BASES

APPLICABLE SAFETY ANALYSES (continued)

occurrence of boiling transition is determined using the approved Critical Power correlations. Details of the $MCPR_{99.9\%}$ calculation are given in References 7, 15, and 16. References 7 and 15 also include a tabulation of the uncertainties and the nominal values of the parameters used in the $MCPR_{99.9\%}$ statistical analysis

The MCPR operating limits are derived from the $MCPR_{99.9\%}$ value and the transient analysis, and are dependent on the operating core flow and power 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.

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

LCO

The MCPR operating limits specified in the COLR ($MCPR_{99.9\%}$ value, $MCPR_f$ values and $MCPR_p$ values) 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, which are based on the $MCPR_{99.9\%}$ limit specified in the COLR.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 23% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 23% 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 23% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 23% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

BASES

REFERENCES (continued)

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

12. ANP-10300P-A, “AURORA-B: An Evaluation Model for Boiling Water Reactors; Application to Transient and Accident Scenarios,” (as identified in the COLR).

13. BAW-10247PA, “Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors,” (as identified in the COLR).

14. BAW-10247P-A Supplement 2P-A, “Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods,” (as identified in the COLR).

15. ANP-10335P-A, “ACE/ATRIUM-11 Critical Power Correlation,” (as identified in the COLR).

~~15.~~16. ANP-10307PA, “AREVA MCPR Safety Limit Methodology for Boiling Water Reactors,” (as identified in the COLR).

Attachment 6 of PLA-7962

Framatome Report ANP-3857NP, Revision 2

Design Limits for Framatome Critical Power Correlations

(Non-proprietary Version)

For Information Only

Design Limits for Framatome Critical Power Correlations

ANP-3857NP
Revision 2

July 2020

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Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Corrected the reference numbers in Table 1.

For Information Only

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1.0 INTRODUCTION

Design limits are provided for Boiling Water Reactor (BWR) critical power correlations. These limits are based solely on the critical power correlation uncertainty determined from benchmarking the correlation to experimental data. Pressurized Water Reactor (PWR) Departure from Nucleate Boiling (DNB) correlation design limits are typically determined from the correlation uncertainty. The statistical expectation for the design limit is provided in Reference 1 where it states,

> *For departure from nucleate boiling ratio (DNBR), CHFR or CPR correlations, there should be a 95-percent probability at the 95-percent confidence level that the hot rod in the core does not experience a DNB or boiling transition condition during normal operation or AOOs*

Design limits for PWR are typically reported and reviewed in the correlation topical reports (for example Reference 2) and are associated with the fuel, independent of the reactor. In this report, comparable limits are determined for BWR critical power correlations.

For Information Only

2.0 DEFINITIONS

With respect to the limit on Critical Heat Flux (CHF), the Safety Limit (SL) in the BWR is defined in the technical specifications as the lowest allowable Critical Power Ratio (CPR) in the reactor core. The CPR is defined in Reference 3.

> *The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.*

The SL on MCPR (sometimes referred to as SLMCPR) includes the influence of plant dependent and fuel dependent uncertainties.

In this report, a design limit is established from the critical power correlation and its uncertainty. This limit determined to be the value of CPR such that, at the 95% confidence level, there is 95% probability that dryout is avoided.

For Information Only

3.0 CRITICAL POWER CORRELATION STATISTICS

In the context of the CPR defined above, an Experimental Critical Power Ratio (ECPR) is defined

$$\text{ECPR} = \frac{\text{Measured Critical Power}}{\text{Calculated Critical Power}} \tag{1}$$

According to this definition, an ECPR that is less than 1.0 is non-conservative and a value greater than 1.0 is conservative. For best estimate critical power correlations, the mean value of ECPR is 1.0 or very close to 1.0. The representation of the ECPR as a normal distribution is addressed directly in the critical power correlation topical report. With the conclusion that the distribution is represented by a normal distribution, the design limit is calculated

$$\text{MCPR}_{95/95} = \frac{1 + k_{95/95}s}{\text{ECPR}} \tag{2}$$

where k is the one sided tolerance limit factor generally attributed to D. B. Owen and given by Reference 4 or 5 and s is the sample standard deviation.

The ECPR definition applied by Framatome for statistical analysis of the critical power correlations is the inverse of the definition shown in Equation (1).

$$\text{ECPR} = \frac{\text{Calculated Critical Power}}{\text{Measured Critical Power}} \tag{3}$$

According to this definition of the ECPR, the design limit is calculated

$$\text{MCPR}_{95/95} = \text{ECPR} \times \left(1 + k_{95/95}s\right) \tag{4}$$

It is observed that when the ECPR is equal to 1.0, Equations (2) and (4) become equal.

For Information Only

4.0 DESIGN LIMITS

The design limits for Framatome fuel are calculated in Table 1 using Equation (4).

Table 1 $MCPR_{95/95}$ Limits for Framatome Fuel

Correlation:	SPCB (ATRIUM 10)	ACE ATRIUM 10	ACE ATRIUM 10XM	ACE ATRIUM 11
Reference	6	7	8	9
Number of Data	[ ]	[ ]	[ ]	[ ]
Mean ECPR	[ ]	[ ]	[ ]	[ ]
Std. Dev.	[ ]	[ ]	[ ]	[ ]
D.B. Owen $k_{95/95}$*	[ ]	[ ]	[ ]	[ ]
Design Limit†	1.04	1.04	1.05	1.05

* The D.B. Owen k factors were interpolated from the table and then conservatively rounded upward to three digits.

† The design limit was rounded upward to two digits.

For Information Only

5.0 REFERENCES

1. “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition – Reactor,” NUREG-0800, Chapter 4.4, Revision 2, page 4.4-5.

2. EMF-92-153(P)(A), Revision 1, “HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel,” Siemens Power Corporation, Nuclear Division, January 2005.

3. “Standard Technical Specifications General Electric BWR/4 Plants Volume 1,” NUREG-1433 Volume 1.0, Revision 4.0, ADAMS Ascension No. ML12104A192, page 1.1-4.

4. D. B. Owen, “Factors for One-sided Tolerance Limits and for Variables Sampling Plans,” Sandia Corporation Report SCR-607, March 1963.

5. M. G. Natrella, “Experimental Statistics,” National Bureau of Standards Handbook 91, August 1963.

6. EMF-2209(P)(A), Revision 3, “SPCB Critical Power Correlation,” AREVA NP Inc., September 2009.

7. ANP-10249P-A, Revision 2, “ACE/ATRIUM-10 Critical Power Correlation,” AREVA Inc., March 2014.

8. ANP-10298P-A, Revision 1, “ACE/ATRIUM 10XM Critical Power Correlation,” AREVA Inc., March 2014.

9. ANP-10335P-A, Revision 0, “ACE/ATRIUM 11 Critical Power Correlation,” Framatome Inc., May 2018.

Attachment 7 of PLA-7962

Framatome Affidavit

Affidavit for ANP-3857P, Revision 2, “Design Limits for Framatome Critical Power Correlations”

A F F I D A V I T

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for Framatome Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in the report ANP-3857P, Revision 2 "Design Limits for Framatome Critical Power Correlations," dated July 2020 and referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

(a) The information reveals details of Framatome's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

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

Executed on: September 23, 2021

MEGINNIS Alan Digitally signed by MEGINNIS Alan
Date: 2021.09.23 14:31:30 -07'00'

Alan B. Meginnis