Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

CNL-14-089

December 11, 2014

10 CFR 50.90

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Browns Ferry Nuclear Plant, Units 1, 2, and 3
Renewed Facility Operating License Nos. DPR-33, DPR-52, and DPR-68
NRC Docket Nos. 50-259, 50-260, and 50-296

Subject: **Browns Ferry Nuclear Plant (BFN), Units 1, 2, and 3 - Application to Modify Technical Specification 2.1.1, Reactor Core Safety Limits (BFN-TS-492)**

Reference: GE Nuclear Energy, "10 CFR 21 Reportable Condition Notification: Potential to Exceed Low Pressure Technical Specification Safety Limit," MFN 05-021, dated March 29, 2005 (Accession No. ML050950428)

In accordance with the provisions of 10 CFR 50.90, ''Application for amendment of license, construction permit, or early site permit," Tennessee Valley Authority (TVA) is submitting a request for amendment to the Technical Specifications (TS) for Browns Ferry Nuclear Plant (BFN), Units 1, 2 and 3. The proposed amendment modifies TS 2.1.1 to revise the reactor dome pressure limit as noted in the reference document.

The enclosure to this letter provides a description of the proposed changes, technical evaluation of the proposed changes, regulatory evaluation, and a discussion of environmental considerations. Attachments 1 and 3 of the Enclosure provide the existing BFN, Units 1, 2, and 3, TS and TS Bases pages marked-up to show the proposed changes. Attachments 2 and 4 provide clean typed BFN, Units 1, 2, and 3 TS and TS Bases pages revised to show the proposed changes. For Attachments 3 and 4, the TS Bases include changes approved in Amendment Nos. 285, 311, and 270, TS-478, which are scheduled for implementation in Spring 2015 (Unit 2), Spring 2016 (Unit 3), and Fall 2016 (Unit 1).

Attachments 5 and 6 contain technical information supporting the acceptability of the revised TS 2.1.1 limit. Attachment 5 contains information that AREVA NP considers to be proprietary in nature and subsequently, pursuant to 10 CFR 2.390, "Public inspections, exemptions, requests for withholding," paragraph (a)(4), it is requested that such information be withheld from public disclosure. Attachment 6 contains the non-proprietary version of the Attachment 5 report with the proprietary material removed, and is suitable for public disclosure. Attachment 7 provides the affidavit supporting this request.

TVA has determined that there are no significant hazards considerations associated with the proposed changes and that the TS changes qualify for a categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and the enclosure to the Alabama State Department of Public Health.

The BFN Plant Operations Review Committee has reviewed this proposed change and determined that operation of BFN in accordance with the proposed change will not endanger the health and safety of the public.

TVA requests approval of these TS changes by December 11, 2015, with implementation within 60 days of issuance.

There are no new regulatory commitments associated with this submittal. If there are any questions or if additional information is needed, please contact Mr. Edward D. Schrull at (423) 751-3850.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 11th day of December 2014.

Respectfully,

J. W. Shea
Vice President, Nuclear Licensing

Enclosure: Technical Specification (TS) Change TS-492 – Changes to Technical Specification 2.1.1 for Browns Ferry Units 1, 2, and 3

cc (Enclosure):

NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Browns Ferry Nuclear Plant
State Health Officer, Alabama State Department of Public Health

Enclosure

Technical Specification (TS) Change TS-492 -
Changes to Technical Specification 2.1.1 for Browns Ferry Units 1, 2, and 3

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend the Operating Licenses for Browns Ferry Nuclear Plant (BFN) Unit 1 (DPR-33), Unit 2 (DPR-52), and Unit 3 (DPR-68). The proposed changes would revise Technical Specification (TS) 2.1.1 for all three units, to lower the value of the reactor steam dome pressure safety limit (SL) to 585 psig. The change resolves the compliance issue outlined in GE Nuclear Energy (GE) 10 CFR Part 21 Reportable Condition Notification MFN 05-021 (Reference 1) (also referred to as Safety Communication (SC) 05-03).

2.0 DETAILED DESCRIPTION

On March 29, 2005, GE Nuclear Energy (GE) issued a 10 CFR 21 Reportable Condition Notification (Reference 1) involving a potential to violate the TS 2.1.1 reactor steam dome pressure safety limit. GE identified that one particular Anticipated Operational Occurrence (AOO) could result in this TS safety limit being violated. The AOO of interest is the Pressure Regulator Failure Open (PRFO) event, which can potentially cause the reactor pressure to decrease below the TS 2.1.1 value of 785 psig while reactor power is at or above 25% of rated thermal power (RTP). GE identified that even plants with a main steam isolation valve (MSIV) low pressure isolation setpoint ≥ 785 psig may experience a PRFO event that could potentially violate the safety limit (SL). The value currently in the BFN TS 2.1.1 of 785 psig corresponds to the lower end of the pressure range over which the GE GEXL critical power correlation was originally tested.

In Reference 1, GE recommended to utilities that the compliance issue outlined in SC 05-03 is best resolved by lowering the SL value in the TS. This approach takes advantage of the fact that more recent critical power correlations have been tested over a wider range of pressure. The current NRC-approved Global Nuclear Fuels (GNF) and AREVA critical power correlations have been tested down to pressures below the current TS 2.1.1 value of 785 psig. The revised TS 2.1.1 SL value of 585 psig proposed in this license amendment request (LAR) is consistent with the lower range of the critical power correlations in use at BFN. The revised TS 2.1.1 SL value also adequately bounds a PRFO transient event. Attachments 1 and 2 of this enclosure provide the marked up and retyped TS pages, for the proposed TS 2.1.1 value.

This LAR also provides the proposed changes to the affected TS Bases pages. Attachments 3 and 4 of this enclosure provide the marked up and retyped Bases pages for information only.

In support of the TS change, a BFN-specific evaluation of the PRFO event was performed by AREVA to demonstrate that the minimum pressure during this AOO would remain above the proposed TS 2.1.1 value. A proprietary version of this AREVA report is included as Attachment 5 of this enclosure and a nonproprietary version is included as Attachment 6 of this enclosure. An affidavit for withholding the proprietary version from public disclosure is included as Attachment 7 of this enclosure.

3.0 TECHNICAL EVALUATION

SC 05-03 concerns the potential for a PRFO event to result in a violation of the reactor dome low pressure SL in TS 2.1.1. The PRFO event involves the failure of the pressure regulator in the open direction, causing the turbine control valves to fully open, including the turbine bypass valves. This failure would result in a rapid depressurization of the reactor. Reactor scram would occur either as a result of the reactor water level swelling to the high level turbine trip setpoint with a scram signal initiated via the main turbine trip, or by the MSIV low pressure isolation setpoint being reached, resulting in an isolation and a scram. The scram would terminate the event, and compliance with the TS 2.1.1 safety limit would be quickly restored, as power would be rapidly reduced to below 25% of RTP.

According to SC 05-03, prior to the scram occurring, the reactor pressure could drop below the SL value while reactor power is still at or above 25% of RTP. However, there would be no actual threat to fuel cladding integrity, because in pressure decrease events in a Boiling Water Reactor (BWR), the reduction in power more than offsets any critical power effect of a reduced pressure. Consequently, the margin to transition boiling would actually increase during this time. Therefore, the issue is one of TS compliance, as the reactor could briefly be in a condition that is not allowed with the current TS low pressure SL value.

The current SL value was established at a time when the critical power correlation of the original equipment fuel vendor had only been tested down to a pressure of 785 psig. Since that time, both GNF and AREVA have tested their critical power correlations over a wider range of pressures (References 2, 3, and 4), such that the lower end of the various tested pressure ranges are all significantly below the 785 psig value. A greater range of pressure is available to increase the margin for transient events that decrease pressure, such as PRFO. Therefore, the TS noncompliance issue can be resolved by taking advantage of the expansion of the tested range of pressures and using it as the basis for lowering the TS 2.1.1 SL value.

TVA proposes that the TS 2.1.1 SL value be reduced from the current 785 psig value to a value of 585 psig. This reduced value remains above the lower bound of both AREVA Critical Power Ratio (CPR) correlations in use at BFN (References 3 and 4).

To demonstrate that the reduced SL value would provide sufficient margin and would not be exceeded during a PRFO event, a plant-specific evaluation of the PRFO for BFN was performed. The analysis (Attachments 5 and 6) included sensitivity studies of the effect of key parameters that affect the minimum reactor pressure obtained during the PRFO event. Included in these sensitivity cases were initial core power, initial core flow, feedwater temperature, MSIV closure time, cycle exposure, scram speed, core average gap conductance, and main steam line pressure drop. The effect of minimum initial dome pressure was accounted for in the feedwater temperature sensitivity cases. The final PRFO analyses assumed that each of these parameters or initial conditions were concurrently taken at the value most adverse in terms of producing the minimum reactor pressure while still above 25% of RTP. Therefore, the analysis bounds the worst case combination of all of the key parameters and is considered to be cycle and unit independent. As noted in the report, the results are insensitive to fuel type, because any new fuel type introduced would be hydraulically matched to existing fuel types, including the fuel type used in the report.

The Attachment 5 report shows that the lowest reactor pressure obtained while power is still above 25% of RTP was 636 psig. This value is above the low end of the tested pressure range of the Reference 3 and 4 AREVA critical power correlations used to monitor the fuel at BFN. It is also above the proposed TS 2.1.1 value of 585 psig. Reducing the TS 2.1.1 value to 585 psig is an acceptable resolution to the TS compliance issue, because the proposed SL value is within the tested pressure range of the AREVA correlations and would not be violated should a PRFO event occur at BFN.

It should be noted that BFN Unit 1 contains legacy GNF GE14 fuel. The GE14 fuel in BFN Unit 1 is monitored using a modified version of the Siemens Power Correlation for BWRs (SPCB) in Reference 3, using the indirect method described in Reference 5. The indirect method uses critical power data generated using the legacy vendor critical power correlation (Reference 2) to determine additive constants for application of the SPCB correlation to the legacy GE14 fuel. This modified correlation is termed SPCB/GE14. While the SPCB correlation itself has a tested pressure range below the proposed 585 psig SL, the Reference 2 GEXL correlation was only tested down to a pressure of 685 psig. A technical justification for applying the SPCB correlation to GE14 fuel for pressures below 685 psig was developed and is provided in the Attachment 5 report.

The justification for applying the SPCB correlation to GE14 fuel at pressures below the tested range of the GEXL correlation relies on the behavior of critical power at pressures in the range of interest. Open literature data shows that critical power increases as pressure decreases in the range of pressure between 585 psig and 685 psig. Testing of the SPCB correlation on ATRIUM-10 fuel shows the behavior of the SPCB correlation is consistent with the behavior described in the literature. Therefore, extending the application of SPCB/GE14 down to pressures as low as 585 psig is justified. To address uncertainties that could result from applying the correlation in this pressure range, AREVA added conservatism to the evaluation of GE14 in Attachment 5, by clamping the pressure used in SPCB/GE14 at 685 psig if the calculated pressure falls below that value. This results in lower calculated critical powers than if the actual pressure were provided to the SPCB/GE14 correlation, thus ensuring that the critical power of the GE14 is calculated conservatively in this pressure range. In addition, all the remaining GE14 fuel in the BFN Unit 1 core is third cycle fuel, with large MCPR margins due to the depleted state of the fuel and the lower power locations of those bundles. Therefore, the GE14 fuel will be adequately protected down to pressures as low as the proposed TS value of 585 psig.

The proposed activity of reducing the low pressure SL will not adversely affect any UFSAR accident analyses. Having reactor pressure as low as 585 psig with reactor power at or above 25% of RTP is by definition a transient condition, because an MSIV closure would occur at the analytical limit of 825 psig. Therefore, these conditions would not be considered as viable initial conditions for any UFSAR accident, because the licensing basis does not require consideration of an accident concurrent with a transient AOO event.

4.0 REGULATORY EVALUATION

4.1 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

10 CFR 50, Appendix A, General Design Criterion (GDC) 10, “Reactor design,” states 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 during any condition of normal operation, including the effects of anticipated operational occurrences. The proposed decrease in the reactor dome pressure safety limit in TS 2.1.1 complies with the requirements of GDC 10 and will continue to ensure that fuel clad integrity is maintained.

10 CFR 50.36(c)(1) requires that SLs be included in the TS. SLs for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. The proposed change modifies existing SLs.

4.2 PRECEDENT

The NRC has previously reviewed and approved the approach of resolving the SC 05-03 noncompliance concern via modifying the TS 2.1.1 low pressure safety limit value, by crediting the broader tested pressure range of the NRC approved critical power correlations now in use. The relevant portion of the license amendment listed below provides a precedent.

> Grand Gulf Nuclear Station Unit 1, Issuance of Amendment No. 191, RE: Extended Power Uprate (pages 324-325), dated July 18, 2012 (TAC NO. ME 4679)

4.3 NO SIGNIFICANT HAZARDS CONSIDERATION

This analysis addresses the proposed change to amend Operating Licenses DPR-33, DPR-52, and DPR-68 for BFN to reduce the TS 2.1.1 low pressure safety limit value.

TVA 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

Decreasing the reactor dome pressure limit in TS 2.1.1 effectively expands the validity range for the AREVA SPCB and ACE/ATRIUM-10 XM critical power correlations and the calculation of Minimum Critical Power Ratio (MCPR). MCPR rises during the pressure reduction that occurs during the PRFO event, and the event is terminated by a scram. Fuel clad integrity is not challenged during any portion of this event. Because the change does not involve a modification to plant hardware, the probability and consequences of the PRFO transient are not affected. The reduction in the reactor dome pressure safety limit from 785 psig to 585 psig provides greater margin to accommodate the pressure reduction during the transient.

The proposed change will continue to support the validity of the critical power correlations applied at BFN. The proposed TS revision involves no significant changes to the operation of any system or component during normal, accident, or transient operating conditions. 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 reduction in the reactor dome pressure safety limit from 785 psig to 585 psig is an administrative change and does not involve changes to the plant hardware or its operating characteristics. As a result, no new failure modes are being introduced. Therefore, the change does not introduce a new or different kind of accident from those previously evaluated.

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

Response: No

The margin of safety is established through the design of plant structures, systems, and components, and through the parameters for safe operation and setpoints of equipment relied upon to respond to transients and design basis accidents. The proposed change in reactor dome pressure does not change the requirements governing operation or availability of safety equipment assumed to operate to preserve the margin of safety. The change does not alter the behavior of the plant equipment, which remains unchanged. The available pressure margin is expanded by the change, thus offering greater margin for pressure reduction during the transient. The critical power capability of the fuel increases as the pressure is reduced from the current TS value to the proposed TS value, so the fuel cladding integrity margin during a PRFO event is not adversely impacted. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, TVA concludes the proposed amendment does not involve a 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.

4.4 CONCLUSIONS

The proposed reduction of the TS 2.1.1 low pressure safety limit value is acceptable based on the following:

- The revised low pressure safety limit is within the range of pressures tested for the AREVA SPCB and ACE/ATRIUM-10 XM critical power correlations.

- The legacy GE14 fuel in BFN Unit 1 has been evaluated using a conservative application of the SPCB correlation for pressures down to the new proposed low pressure SL. The GE14 fuel will be adequately protected against a PRFO event.

- A BFN-specific analysis of the PRFO event has been completed to demonstrate the adequacy of the revised low pressure SL value. This analysis utilized the NRC-approved AREVA transient methods listed in TS 5.6.5.b of the BFN TS.

- The resolution of the TS noncompliance via the proposed change does not require any plant modification that could affect the behavior of the plant during normal, transient, or accident operation.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance 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.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined the proposed amendment 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 amendment 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 amendment 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 amendment.

6.0 REFERENCES

1. GE Energy-Nuclear, "10 CFR 21 Reportable Condition Notification: Potential to Exceed Low Pressure Technical Specification Safety Limit," MFN 05-021, March 29, 2005. (ML050950428)

2. Global Nuclear Fuel, "GEXL14 Correlation for GE14 Fuel," NEDC-32851P-A, Revision 4, September 2007

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

4. AREVA NP Inc., "ACE/ATRIUM 10XM Critical Power Correlation," ANP-10298PA, Revision 0, March 2010

5. Siemens Power Corporation, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," EMF-2245(P)(A), Revision 0, August 2000

ATTACHMENT 1

Proposed Technical Specification Pages (Mark-up)

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~~ 585 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ ~~785~~ 585 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.11 for single 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.

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~~ 585 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ ~~785~~ 585 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.10 for single 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.

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~~ 585 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ ~~785~~ 585 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.11 for single 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.

ATTACHMENT 2

Proposed Technical Specification Pages (Retyped)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 585 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 585 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.11 for single 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.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 585 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 585 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.10 for single 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.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 585 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 585 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.11 for single 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.

ATTACHMENT 3

Proposed Technical Specification Bases Pages (Mark-up)

For Information Only

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity

Critical power correlations are valid over a wide range of conditions per References 2 and 5, extending to expected conditions below 25% THERMAL POWER. For core thermal power levels at, or above 25% rated, the hot channel flow rate is expected to be >28,000 lbm/hr, (core flow not less than natural circulation i.e., ~25%-30 % core flow for 25% power); therefore, the fuel cladding integrity SL is conservative relative to the applicable range of the critical power correlations. For operation at low pressure/flow conditions, consistent with the low power region of the Power/Flow operating map, another basis is used as follows:

The static head across the fuel bundles is due to elevation effects from water solid channel, core bypass, and annulus regions, is approximately 4.5 psid. The pressure differential is maintained by the water solid bypass region of the core, along with the annulus region of the vessel. Elevation head provided by the bypass and annulus regions produces natural circulation flow conditions balancing pressure head with loss terms inside the core shroud.

Natural circulation principles maintain a core plenum to plenum pressure drop of approximately 4.5 to 5 psid along the natural circulation flow line of the Power/Flow operating map. When power levels approach 25% rated, pressure drop and density head terms are closely balanced as power changes, such that natural circulation flow is nearly independent of reactor power.

The flow characteristic is represented by the nearly vertical portion of the natural circulation line on the Power/Flow operating map. For a core pressure drop of approximately 4.5 to 5 psid, the hot channel flow rate is expected to be >28,000 lbm/hr in the region of operation when core power is $\leq$ 25% with a corresponding core pressure drop of about 4.5 to 5 psid.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity (continued)

For example, Reference 5 test data, taken at low pressures and flow rates, indicate assembly critical power in excess of 4 MWt, for flow rates indicative of natural circulation conditions. At 25% rated power, assembly average power is ≤ 1.2 MWt. When considering design peaking factors, hot channel power could be expected to be on the order of 2 MWt. Consequently, operation up to 25% rated core power, with normal natural circulation available, is conservative even if reactor pressure is less than the lower pressure limit of the critical power correlation.

When reactor power is significantly less than 25% of rated (e.g., below 10% of rated), hot channel flow supported by the available driving head may fall below 28,000 lbm/hr (along the lower portion of the natural circulation flow characteristic on the Power/Flow map). However, the critical power supported by the flow, remains above actual hot channel power conditions. The inherent characteristics of BWR natural circulation make core power/flow follow the natural circulation line as long as normal annulus water level is maintained.

Operation below 25% rated core thermal power is conservatively acceptable, even for reactor operations at natural circulation. Adequate fuel thermal margins are maintained for low power conditions present during core natural circulation, even though the flow may be less than the critical power correlation applicability range.

Add new paragraph:
The low pressure safety limit value of 585 psig has been determined to adequately bound the minimum pressure that might occur while reactor power is at or above 25% of rated. This condition would most likely be created by a pressure regulator failure open transient (PRFO) that results in a rapid depressurization of the vessel and a subsequent scram. Reference 8 provides a detailed evaluation of this transient event, and provides the basis for the low pressure safety limit of 585 psig.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no 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. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model combining 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 AREVA critical power correlations. One specific uncertainty included in the SL is the uncertainty inherent in the critical power correlation. References 2, 3, 4, 5, and 6 describe the uncertainties and methodologies used in determining the MCPR SL.

(continued)

BASES (continued)

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 50.67, "Accident Source Term," limits (Ref. 7). 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. EMF-2209(P)(A), SPCB Critical Power Correlation, (as identified in the COLR).
3. EMF-2245(P)(A), Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel, (as identified in the COLR).
4. ANP-10307PA Revision 0, AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, AREVA NP, June 2011.
5. ANP-10298PA Revision 0, ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, March 2010.
6. ANP-3140(P) Revision 0, Browns Ferry Units 1, 2, and 3 Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, August 2012.
7. 10 CFR 50.67.
8. ANP-3245P, Revision 1, Browns Ferry Evaluation of PRFO Low Pressure Technical Specification Value, AREVA Inc., February 2014.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b. Main Steam Line Pressure - Low (PIS-1-72, 76, 82, 86)

Low MSL pressure with the reactor at power indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure - Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below ~~785~~ 585 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure - Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 2).

This Function isolates the Group 1 valves excluding the Recirculation Loop Sample valves.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity

Critical power correlations are valid over a wide range of conditions per References 2 and 5, extending to expected conditions below 25% THERMAL POWER. For core thermal power levels at, or above 25% rated, the hot channel flow rate is expected to be >28,000 lbm/hr, (core flow not less than natural circulation i.e., ~25%-30 % core flow for 25% power); therefore, the fuel cladding integrity SL is conservative relative to the applicable range of the critical power correlations. For operation at low pressure/flow conditions, consistent with the low power region of the Power/Flow operating map, another basis is used as follows:

The static head across the fuel bundles is due to elevation effects from water solid channel, core bypass, and annulus regions, is approximately 4.5 psid. The pressure differential is maintained by the water solid bypass region of the core, along with the annulus region of the vessel. Elevation head provided by the bypass and annulus regions produces natural circulation flow conditions balancing pressure head with loss terms inside the core shroud.

Natural circulation principles maintain a core plenum to plenum pressure drop of approximately 4.5 to 5 psid along the natural circulation flow line of the Power/Flow operating map. When power levels approach 25% rated, pressure drop and density head terms are closely balanced as power changes, such that natural circulation flow is nearly independent of reactor power.

The flow characteristic is represented by the nearly vertical portion of the natural circulation line on the Power/Flow operating map. For a core pressure drop of approximately 4.5 to 5 psid, the hot channel flow rate is expected to be >28,000 lbm/hr in the region of operation when core power is ≤ 25% with a corresponding core pressure drop of about 4.5 to 5 psid.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity (continued)

For example, Reference 5 test data, taken at low pressures and flow rates, indicate assembly critical power in excess of 4 MWt, for flow rates indicative of natural circulation conditions. At 25% rated power, assembly average power is ≤ 1.2 MWt. When considering design peaking factors, hot channel power could be expected to be on the order of 2 MWt. Consequently, operation up to 25% rated core power, with normal natural circulation available, is conservative even if reactor pressure is less than the lower pressure limit of the critical power correlation.

When reactor power is significantly less than 25% of rated (e.g., below 10% of rated), hot channel flow supported by the available driving head may fall below 28,000 lbm/hr (along the lower portion of the natural circulation flow characteristic on the Power/Flow map). However, the critical power supported by the flow, remains above actual hot channel power conditions. The inherent characteristics of BWR natural circulation make core power/flow follow the natural circulation line as long as normal annulus water level is maintained.

Operation below 25% rated core thermal power is conservatively acceptable, even for reactor operations at natural circulation. Adequate fuel thermal margins are maintained for low power conditions present during core natural circulation, even though the flow may be less than the critical power correlation applicability range.

Add new paragraph:
The low pressure safety limit value of 585 psig has been determined to adequately bound the minimum pressure that might occur while reactor power is at or above 25% of rated. This condition would most likely be created by a pressure regulator failure open transient (PRFO) that results in a rapid depressurization of the vessel and a subsequent scram. Reference 8 provides a detailed evaluation of this transient event, and provides the basis for the low pressure safety limit of 585 psig.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no 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. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model combining 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 AREVA critical power correlations. One specific uncertainty included in the SL is the uncertainty inherent in the critical power correlation. References 2, 3, 4, 5, and 6 describe the uncertainties and methodologies used in determining the MCPR SL.

(continued)

BASES (continued)

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 50.67, "Accident Source Term," limits (Ref. 7). 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. EMF-2209(P)(A), SPCB Critical Power Correlation, (as identified in the COLR).
3. EMF-2245(P)(A), Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel, (as identified in the COLR).
4. ANP-10307PA Revision 0, AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, AREVA NP, June 2011.
5. ANP-10298PA Revision 0, ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, March 2010.
6. ANP-3140(P) Revision 0, Browns Ferry Units 1, 2, and 3 Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, August 2012.
7. 10 CFR 50.67.

8. ANP-3245P, Revision 1, Browns Ferry Evaluation of PRFO Low Pressure Technical Specification Value, AREVA Inc., February 2014.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b. Main Steam Line Pressure - Low (PIS-1-72, 76, 82, 86)

Low MSL pressure with the reactor at power indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure - Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below ~~785~~ 585 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure - Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 2).

This Function isolates the Group 1 valves excluding the Recirculation Loop Sample valves.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity

Critical power correlations are valid over a wide range of conditions per References 2 and 5, extending to expected conditions below 25% THERMAL POWER. For core thermal power levels at, or above 25% rated, the hot channel flow rate is expected to be >28,000 lbm/hr, (core flow not less than natural circulation i.e., ~25%-30 % core flow for 25% power); therefore, the fuel cladding integrity SL is conservative relative to the applicable range of the critical power correlations. For operation at low pressure/flow conditions, consistent with the low power region of the Power/Flow operating map, another basis is used as follows:

The static head across the fuel bundles is due to elevation effects from water solid channel, core bypass, and annulus regions, is approximately 4.5 psid. The pressure differential is maintained by the water solid bypass region of the core, along with the annulus region of the vessel. Elevation head provided by the bypass and annulus regions produces natural circulation flow conditions balancing pressure head with loss terms inside the core shroud.

Natural circulation principles maintain a core plenum to plenum pressure drop of approximately 4.5 to 5 psid along the natural circulation flow line of the Power/Flow operating map. When power levels approach 25% rated, pressure drop and density head terms are closely balanced as power changes, such that natural circulation flow is nearly independent of reactor power.

The flow characteristic is represented by the nearly vertical portion of the natural circulation line on the Power/Flow operating map. For a core pressure drop of approximately 4.5 to 5 psid, the hot channel flow rate is expected to be >28,000 lbm/hr in the region of operation when core power is ≤25% with a corresponding core pressure drop of about 4.5 to 5 psid.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity (continued)

For example, Reference 5 test data, taken at low pressures and flow rates, indicate assembly critical power in excess of 4 MWt, for flow rates indicative of natural circulation conditions. At 25% rated power, assembly average power is ≤1.2 MWt. When considering design peaking factors, hot channel power could be expected to be on the order of 2 MWt. Consequently, operation up to 25% rated core power, with normal natural circulation available, is conservative even if reactor pressure is less than the lower pressure limit of the critical power correlation.

When reactor power is significantly less than 25% of rated (e.g., below 10% of rated), hot channel flow supported by the available driving head may fall below 28,000 lbm/hr (along the lower portion of the natural circulation flow characteristic on the Power/Flow map). However, the critical power supported by the flow, remains above actual hot channel power conditions. The inherent characteristics of BWR natural circulation make core power/flow follow the natural circulation line as long as normal annulus water level is maintained.

Operation below 25% rated core thermal power is conservatively acceptable, even for reactor operations at natural circulation. Adequate fuel thermal margins are maintained for low power conditions present during core natural circulation, even though the flow may be less than the critical power correlation applicability range.

Add new paragraph:
The low pressure safety limit value of 585 psig has been determined to adequately bound the minimum pressure that might occur while reactor power is at or above 25% of rated. This condition would most likely be created by a pressure regulator failure open transient (PRFO) that results in a rapid depressurization of the vessel and a subsequent scram. Reference 8 provides a detailed evaluation of this transient event, and provides the basis for the low pressure safety limit of 585 psig.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no 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. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model combining 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 AREVA critical power correlations. One specific uncertainty included in the SL is the uncertainty inherent in the critical power correlation. References 2, 3, 4, 5, and 6 describe the uncertainties and methodologies used in determining the MCPR SL.

(continued)

BASES (continued)

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 50.67, "Accident Source Term," limits (Ref. 7). 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. EMF-2209(P)(A), SPCB Critical Power Correlation, (as identified in the COLR).

3. EMF-2245(P)(A), Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel, (as identified in the COLR).

4. ANP-10307PA Revision 0, AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, AREVA NP, June 2011.

5. ANP-10298PA Revision 0, ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, March 2010.

6. ANP-3140(P) Revision 0, Browns Ferry Units 1, 2, and 3 Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, August 2012.

7. 10 CFR 50.67.

8. ANP-3245P, Revision 1, Browns Ferry Evaluation of PRFO Low Pressure Technical Specification Value, AREVA Inc., February 2014.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

<u>1.b. Main Steam Line Pressure - Low</u> (PIS-1-72, 76, 82, 86)

Low MSL pressure with the reactor at power indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure - Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below ~~785~~ 585 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure - Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 2).

This Function isolates the Group 1 valves excluding the Recirculation Loop Sample valves.

(continued)

ATTACHMENT 4

Proposed Technical Specification Bases Pages (Retyped)

For Information Only

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity (continued)

For example, Reference 5 test data, taken at low pressures and flow rates, indicate assembly critical power in excess of 4 MWt, for flow rates indicative of natural circulation conditions. At 25% rated power, assembly average power is $\leq$ 1.2 MWt. When considering design peaking factors, hot channel power could be expected to be on the order of 2 MWt. Consequently, operation up to 25% rated core power, with normal natural circulation available, is conservative even if reactor pressure is less than the lower pressure limit of the critical power correlation.

When reactor power is significantly less than 25% of rated (e.g., below 10% of rated), hot channel flow supported by the available driving head may fall below 28,000 lbm/hr (along the lower portion of the natural circulation flow characteristic on the Power/Flow map). However, the critical power supported by the flow, remains above actual hot channel power conditions. The inherent characteristics of BWR natural circulation make core power/flow follow the natural circulation line as long as normal annulus water level is maintained.

Operation below 25% rated core thermal power is conservatively acceptable, even for reactor operations at natural circulation. Adequate fuel thermal margins are maintained for low power conditions present during core natural circulation, even though the flow may be less than the critical power correlation applicability range.

The low pressure safety limit value of 585 psig has been determined to adequately bound the minimum pressure that might occur while reactor power is at or above 25% of rated. This condition would most likely be created by a pressure regulator failure open transient (PRFO) that results in a rapid depressurization of the vessel and a subsequent scram. Reference 8 provides a detailed evaluation of this transient event, and provides a basis for the low pressure safety limit of 585 psig.

(continued)

BASES (continued)

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 50.67, “Accident Source Term,” limits (Ref. 7). 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. EMF-2209(P)(A), SPCB Critical Power Correlation, (as identified in the COLR).
3. EMF-2245(P)(A), Application of Siemens Power Corporation’s Critical Power Correlations to Co-Resident Fuel, (as identified in the COLR).
4. ANP-10307PA Revision 0, AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, AREVA NP, June 2011.
5. ANP-10298PA Revision 0, ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, March 2010.
6. ANP-3140(P) Revision 0, Browns Ferry Units 1, 2, and 3 Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, August 2012.
7. 10 CFR 50.67.
8. ANP-3245P, Revision 1, Browns Ferry Evaluation of PRFO Low Pressure Technical Specification Value, AREVA Inc., February 2014.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b. Main Steam Line Pressure - Low (PIS-1-72, 76, 82, 86)

Low MSL pressure with the reactor at power indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure - Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 585 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure - Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 2).

This Function isolates the Group 1 valves excluding the Recirculation Loop Sample valves.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity (continued)

For example, Reference 5 test data, taken at low pressures and flow rates, indicate assembly critical power in excess of 4 MWt, for flow rates indicative of natural circulation conditions. At 25% rated power, assembly average power is ≤ 1.2 MWt. When considering design peaking factors, hot channel power could be expected to be on the order of 2 MWt. Consequently, operation up to 25% rated core power, with normal natural circulation available, is conservative even if reactor pressure is less than the lower pressure limit of the critical power correlation.

When reactor power is significantly less than 25% of rated (e.g., below 10% of rated), hot channel flow supported by the available driving head may fall below 28,000 lbm/hr (along the lower portion of the natural circulation flow characteristic on the Power/Flow map). However, the critical power supported by the flow, remains above actual hot channel power conditions. The inherent characteristics of BWR natural circulation make core power/flow follow the natural circulation line as long as normal annulus water level is maintained.

Operation below 25% rated core thermal power is conservatively acceptable, even for reactor operations at natural circulation. Adequate fuel thermal margins are maintained for low power conditions present during core natural circulation, even though the flow may be less than the critical power correlation applicability range.

The low pressure safety limit value of 585 psig has been determined to adequately bound the minimum pressure that might occur while reactor power is at or above 25% of rated. This condition would most likely be created by a pressure regulator failure open transient (PRFO) that results in a rapid depressurization of the vessel and a subsequent scram. Reference 8 provides a detailed evaluation of this transient event, and provides the basis for the low pressure safety limit of 585 psig.

(continued)

BASES (continued)

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 50.67, "Accident Source Term," limits (Ref. 7). 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. EMF-2209(P)(A), SPCB Critical Power Correlation, (as identified in the COLR).
3. EMF-2245(P)(A), Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel, (as identified in the COLR).
4. ANP-10307PA Revision 0, AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, AREVA NP, June 2011.
5. ANP-10298PA Revision 0, ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, March 2010.
6. ANP-3140(P) Revision 0, Browns Ferry Units 1, 2, and 3 Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, August 2012.
7. 10 CFR 50.67.
8. ANP-3245P, Revision 1, Browns Ferry Evaluation of PRFO Low Pressure Technical Specification Value, AREVA Inc., February 2014.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

<u>1.b. Main Steam Line Pressure - Low</u> (PIS-1-72, 76, 82, 86)

Low MSL pressure with the reactor at power indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure - Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 585 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure - Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 2).

This Function isolates the Group 1 valves excluding the Recirculation Loop Sample valves.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity (continued)

For example, Reference 5 test data, taken at low pressures and flow rates, indicate assembly critical power in excess of 4 MWt, for flow rates indicative of natural circulation conditions. At 25% rated power, assembly average power is ≤1.2 MWt. When considering design peaking factors, hot channel power could be expected to be on the order of 2 MWt. Consequently, operation up to 25% rated core power, with normal natural circulation available, is conservative even if reactor pressure is less than the lower pressure limit of the critical power correlation.

When reactor power is significantly less than 25% of rated (e.g., below 10% of rated), hot channel flow supported by the available driving head may fall below 28,000 lbm/hr (along the lower portion of the natural circulation flow characteristic on the Power/Flow map). However, the critical power supported by the flow, remains above actual hot channel power conditions. The inherent characteristics of BWR natural circulation make core power/flow follow the natural circulation line as long as normal annulus water level is maintained.

Operation below 25% rated core thermal power is conservatively acceptable, even for reactor operations at natural circulation. Adequate fuel thermal margins are maintained for low power conditions present during core natural circulation, even though the flow may be less than the critical power correlation applicability range.

The low pressure safety limit value of 585 psig has been determined to adequately bound the minimum pressure that might occur while reactor power is at or above 25% of rated. This condition would most likely be created by a pressure regulator failure open transient (PRFO) that results in a rapid depressurization of the vessel and a subsequent scram. Reference 8 provides a detailed evaluation of this transient event, and provides the basis for the low pressure safety limit of 585 psig.

(continued)

BASES (continued)

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 50.67, "Accident Source Term," limits (Ref. 7). 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. EMF-2209(P)(A), SPCB Critical Power Correlation, (as identified in the COLR).
3. EMF-2245(P)(A), Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel, (as identified in the COLR).
4. ANP-10307PA Revision 0, AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, AREVA NP, June 2011.
5. ANP-10298PA Revision 0, ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, March 2010.
6. ANP-3140(P) Revision 0, Browns Ferry Units 1, 2, and 3 Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, August 2012.
7. 10 CFR 50.67.
8. ANP-3245P, Revision 1, Browns Ferry Evaluation of PRFO Low Pressure Technical Specification Value, AREVA Inc., February 2014.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

<u>1.b. Main Steam Line Pressure - Low</u> (PIS-1-72, 76, 82, 86)

Low MSL pressure with the reactor at power indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure - Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 585 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure - Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 2).

This Function isolates the Group 1 valves excluding the Recirculation Loop Sample valves.

(continued)

ATTACHMENT 6

ANP-3245NP Revision 1
Browns Ferry Evaluation of PRFO Low Pressure Technical Specification Value
(Non-Proprietary)

ANP-3245NP
Revision 1

Browns Ferry Evaluation of PRFO Low Pressure Technical Specification Value

February 2014

AREVA Inc.

AREVA Inc.

ANP-3245NP
Revision 1

Browns Ferry Evaluation of PRFO Low Pressure Technical Specification Value

AREVA Inc.

ANP-3245NP
Revision 1

Browns Ferry Evaluation of PRFO Low Pressure Technical Specification Value

skm

Nature of Changes

Item	Page	Description and Justification
1.	All	Changed classification from “Proprietary” to “Proprietary – Commercial”

Contents

Tables

Figures

1.0 Introduction

TVA requested AREVA to evaluate (Reference 1) if the low pressure isolation setpoint (LPIS) for the main steam isolation valve (MSIV) is adequate to support the critical power ratio (CPR) safety limit being maintained during the time that the reactor is above 25% rated thermal power (RTP) during the pressure regulator failure open (PRFO) event.

The purpose of this document is to present the analysis results for the PRFO event with respect to the lowest pressure predicted at the steam dome during the transient. AREVA has previously dispositioned this event as a non-limiting event with respect to CPR, References 2 and 3, for Browns Ferry. The current pressure limit for the safety limit minimum critical power ratio (SLMCPR) is provided in the Technical Specifications (TS) for each of the Browns Ferry Nuclear Station units is 785 psig, References 4, 5, and 6.

2.0 Summary of Results

During the PRFO event, the reactor will depressurize and the steam dome pressure will drop below the current value of 785 psig identified in Browns Ferry Technical Specifications (TS) Section 2.1.1 and associated bases, while reactor thermal power is greater than 25% of rated. Therefore, the current analytical value of the LPIS of 825 psig is not adequate to support the TS pressure limit.

Section 3.0 presents the AREVA analysis results for the PRFO event evaluated for Browns Ferry Units 1, 2, and 3. The evaluation is performed such that the results are cycle independent and unit independent at the Browns Ferry Nuclear Station. The lowest pressure calculated for Browns Ferry, while reactor thermal power is greater than 25% of rated, is 636 psig.

Section 4.0 provides a technical justification for extending the lower pressure boundary of the SPCB critical power correlation being applied to co-resident GE14 fuel in Browns Ferry Unit 1. The current core composition of Browns Ferry Units 2 and 3 is 100% ATRIUM™-10* fuel.

The lower bound of the pressure range for AREVA's critical power correlations are [ ], References 7 and 8 respectively.

The results provided in Section 3.0 support an update to the Browns Ferry Technical Specifications Section 2.1.1 SLMCPR pressure limit value of 585 psig.

The pressure results presented in this report were obtained from full core configurations of ATRIUM-10 fuel or mixed cores of GE14 and ATRIUM-10 fuel for Browns Ferry. However, the conclusions are applicable to future core loadings that include different fuel designs. The main basis of the event is not fast, (i.e. LRNB or FWCF) such that differences in neutronics feedback of different fuel designs are not significant. This event is driven primarily by a depressurization of the reactor system, which is a result of valve stroke times and set points. As long as the thermal-hydraulic characteristics of the new fuel design are similar to the ATRIUM-10 and it is determined to be hydraulically compatible, the overall response during a PRFO transient will not

* ATRIUM is a trademark of AREVA.

be significantly different for transition cores of coresident fuel or full cores of different fuel designs. In addition, since about 95% of the reactor system volume is outside the core region, slight changes in core volume and fluid energy due to fuel differences will produce an insignificant change in total system volume and energy. For these reasons, the overall system response and hence the lowest calculated pressure for cores including other characteristically similar and compatible fuel are not significantly different during the transition to a full core of that fuel design.

3.0 Event Evaluation

Section 14.5.5.1 of Reference 9 addresses the PRFO event. Should the pressure regulation function of the turbine control system fail in an open direction, the turbine admission valves can be fully opened with the turbine bypass valves partially or fully opened. This condition results in an initial decrease in the coolant inventory in the reactor vessel as the mass flow of steam leaving the vessel exceeds the mass flow of water entering the vessel. The total steam flow rate resulting from a pressure regulation malfunction is limited by the turbine controls to the total capacity of turbine control valves and turbine bypass valves.

The reactor water level swelling due to the decreasing reactor vessel pressure may reach the high level L8 setpoint initiating a turbine stop valve closure. Following this action, feedwater pumps trip, recirculation pumps trip, and reactor scram will take place. If L8 is not reached, the vessel depressurizes and the turbine header pressure may drop to the low pressure setpoint for reactor isolation; the MSIVs will then close, and a reactor scram will be initiated.

3.1 Sensitivity Evaluation

3.1.1 Core flow

Table 3.1 presents the minimum dome pressure sensitivity evaluation on reactor core flow. The evaluation was performed for the highest and lowest core flow allowed on the power/flow map for a given power level. Less core flow for a given power level results in less mass in the core during the depressurization phase of the event. Therefore, there is a slightly higher depressurization rate in the steam dome with the lower core flow conditions. The calculated pressures show that lower core flows for a given power level result in a lower dome pressure during the event.

**Table 3.1 Core Flow Sensitivity of
Minimum Steam Dome Pressure (psig)**

State Point	BFE1	BFE2	BFE3
100/105	821	832	834
100/81	809	822	822
65/110	805	812	812
65/40	753	764	761

3.1.2 Initial Conditions

Browns Ferry licensing calculations support plant operation within a range of dome pressures and feedwater temperatures, which is considered base case operation and not an EOOS condition. An example of the range of initial conditions for dome pressure and feedwater temperature is provided in Figures 2.2 and 2.3 of Reference 10.

Table 3.2 presents the sensitivity results for the assumed initial conditions. The event is not significantly affected by the initial dome pressure. However, there is an impact due to the initial feedwater temperature. Lower initial feedwater temperatures produce less steam during the transient. Therefore, the depressurization of the system occurs more quickly and a lower dome pressure is obtained before the MSIV has a chance to completely close.

It is clear that the feedwater heaters out-of-service (FHOOS) condition (the event with the lowest initial dome pressure and feedwater temperature), results in the most conservative minimum steam dome pressure during the PRFO event.

Table 3.2 Initial Conditions Sensitivity of Minimum Steam Dome Pressure (psig)

Initial Conditions	BFE1	BFE2	BFE3
Nominal Temperature Increased Pressure	809	822	822
Nominal Temperature Reduced Pressure	809	823	822
Reduced Temperature Increased Pressure	806	820	819
Reduced Temperature Reduced Pressure	807	820	819
FHOOS Temperature	791	804	802

3.1.3 MSIV closure time

The minimum steam dome pressure for the PRFO event is significantly affected by the closure time assumed for the MSIV. There is a minimum and maximum closure time defined for AREVA licensing calculations. The range is from 3.0 seconds to 5.0 seconds, as noted in Items 3.7.1 and 3.7.2 of Reference 10.

As the closure time increases, the time it takes to isolate the vessel is increased. This allows more time for the vessel to depressurize during the event. Table 3.3 provides the sensitivity results for the MSIV closure time. The results support the conclusion that a longer closure time is conservative for this event.

Table 3.3 MSIV Closure Time Sensitivity of Minimum Steam Dome Pressure (psig)

MSIV Closure	BFE1	BFE2	BFE3
3-second closure	789	801	799
4-second closure	746	757	757
5-second closure	709	716	717

3.1.4 Cycle Exposure

In order to determine the variation of the minimum dome pressure due to cycle operation, calculations were performed for the range of licensing exposure typically analyzed in support of plant operation. The vessel response during the depressurization phase of the event is dependent upon the axial power shape at the time of the event. In general, the axial power shape at the beginning of a cycle is significantly negative (meaning more power is generated in the bottom half of the core than the top), but shifts higher in the core as the cycle nears completion.

Table 3.4 presents the minimum steam dome pressures for the cycle exposure sensitivity. The calculations represent Browns Ferry Unit 1 Cycle 10, Unit 2 Cycle 18, and Unit 3 Cycle 16. It is difficult to isolate the cycle exposure impact since there are competing effects that are interconnected during plant operation (i.e., core average rod gap conductance, void reactivity, axial power shape and magnitude). However, the results of trends provided in Table 3.4 are consistent for three different reactor cycles. They also show that the minimum dome pressure of the PRFO event is relatively insensitive to the cycle exposure.

Table 3.4 Cycle Exposure Sensitivity of Minimum Steam Dome Pressure (psig)

Cycle Exposure	BFE1	BFE2	BFE3
BOC	709	716	717
MOC	708	716	716
Licensing EOFP	707	712	713
Coastdown	709	714	715

3.1.5 Scram insertion

The PRFO event is terminated from an MSIV closure. Once the MSIV begins to close, the reactor protection system initiates a reactor scram once the MSIV reaches 90% open. Insertion time of the control blades directly controls the rate of power decrease and therefore, the rate of depressurization before the MSIVs have a chance to fully close and stop the reduction of pressure.

Table 3.5 presents the pressure sensitivity results due to scram insertion speeds. AREVA typically analyzes 3 separate sets of scram speeds for Browns Ferry, provided in Item 4.3 of Reference 10. One extra scram speed curve was included in this sensitivity. The entire optimal scram speed (OSS) insertion time curve was reduced by 10% to allow a faster insertion of the blades. The results show that the minimum steam dome pressure is relatively insensitive to the scram speed. However, there is a definite trend of faster scram insertion times result in a lower, more conservative minimum steam dome pressure during the PRFO event.

Table 3.5 Scram Insertion Sensitivity of Minimum Steam Dome Pressure (psig)

Scram Time	BFE1	BFE2	BFE3
TSSS	792	804	803
NSS	791	803	801
OSS	790	802	800
OSS reduced by 10%	789	801	799

3.1.6 Core Average Gap Conductance

The amount of heat that is transferred from the fuel to the coolant is a function of the core average fuel rod gap conductance (HGAP). During the event HGAP will have an effect on the minimum steam dome pressure. A higher core average HGAP, assuming all other parameters are held constant, will result in more heat being transferred into the coolant. Therefore, during the event, there is less power and a faster rate of depressurization of the steam dome.

Table 3.6 presents the pressure sensitivity results due to core average HGAP. As shown, an increase of 20% to the core average HGAP value resulted in a lower minimum steam dome pressure.

Table 3.6 Core Average HGAP Sensitivity of Minimum Steam Dome Pressure (psig)

Condition	BFE1	BFE2	BFE3
Nominal HGAP	709	716	717
HGAP +20%	705	714	713
HGAP -20%	713	719	719

3.2 Conclusions

The sensitivity to various parameters affecting the minimum steam dome pressure during a PRFO transient is presented in Sections 3.1. The conclusions from these studies are:

- Low core flow bounds high core flow
- Initial conditions of dome pressure and feedwater temperature. FHOOS conditions and the corresponding dome pressure are conservative
- Slower MSIV closure time, 5 seconds, is conservative
- Minimum pressure of the PRFO event is relatively insensitive to cycle exposure
- Faster scram times provide a lower minimum steam dome pressure during the event
- Higher core average gap conductance providing a lower minimum steam dome pressure during the event

Table 3.7 presents the results for a range of power levels at each of the Browns Ferry units. These cases are performed using the conclusions outlined above from the sensitivity analyses documented in Section 3.1. This includes FHOOS temperatures and 5 second MSIV closure. The BOC cycle exposure was chosen for analysis. To ensure the variability due to cycle operation and bundle design is bound, a 20% increase to the unit/cycle specific BOC core average HGAPs are included as well as reducing the reactor scram curve by 10% for OSS.

The results in Table 3.7 show that Browns Ferry Unit 1 is the most limiting of the three units. The primary reason for this is Unit 1 has the lowest steam line pressure drop compared to Units 2 and 3. The conservative minimum steam dome pressure for this event is 636 psig, which is obtained from the 60/35 state point for Unit 1.

In each of the results shown previously in Tables 3.1 – 3.6, the minimum steam dome pressure occurred while reactor power was greater than 25% of rated. However, as the state point decreases in power, the thermal power during the event will decrease below 25% of rated. When this occurs, the minimum steam dome pressure in Table 3.7 is reported as the pressure at the time when heat flux equals 25% of rated.

Responses of various reactor and plant parameters during the limiting Unit 1 PRFO event initiated at 60% of rated power and 35% of rated core flow are shown in Figures 3.1-3.2.

Table 3.7 Minimum Steam Dome Pressure (psig) for the PRFO Event

State Point	BFE1	BFE2	BFE3
100/81	705	714	713
90/70	688	696	695
75/50	653	659	657
65/40	637	645	641
60/35	636*	652*	650*
50/35	690*	709*	707*
40/35	762*	770*	773*
30/35	861*	857*	867*

* These pressures reported for these cases are obtained at the time when the heat flux during the event decreases below 25% of rated. This occurs prior to full closure of the MSIV.

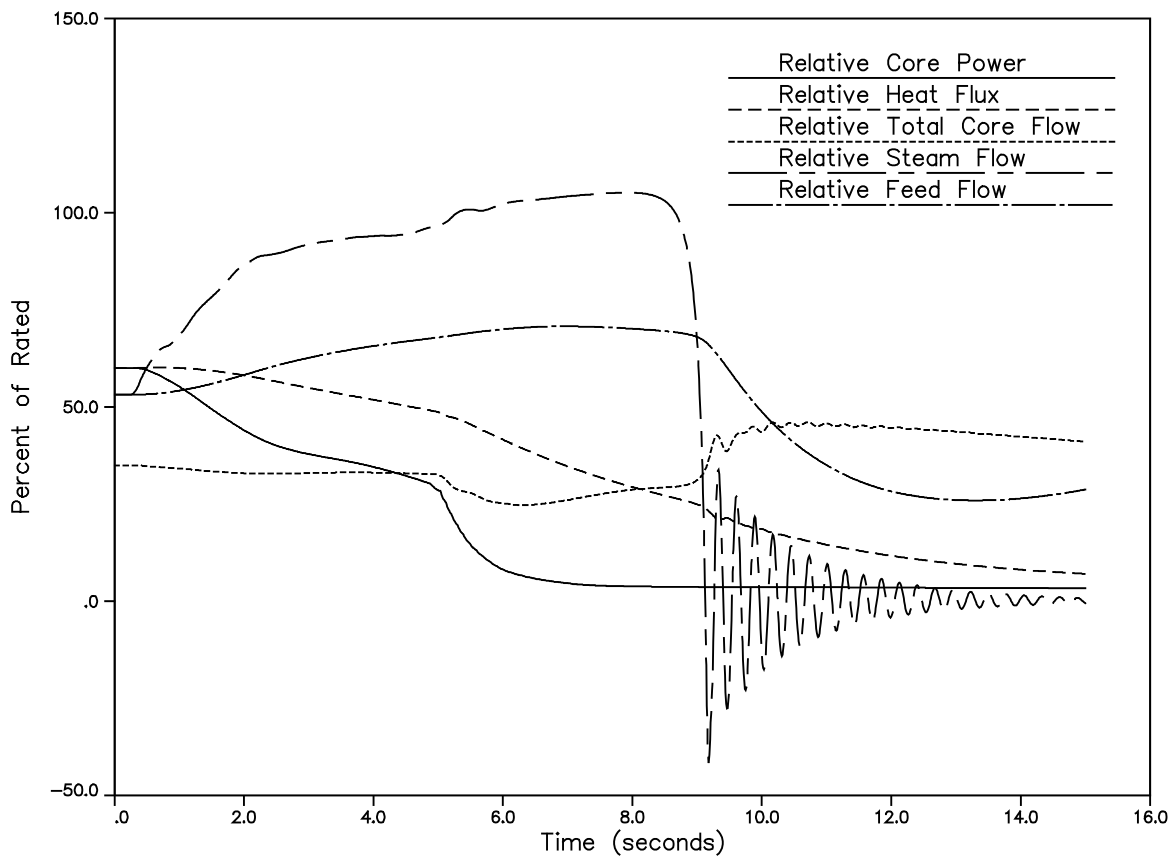


Figure 3.1 Browns Ferry Unit 1 PRFO Transient at 60P/35F – Key Parameters

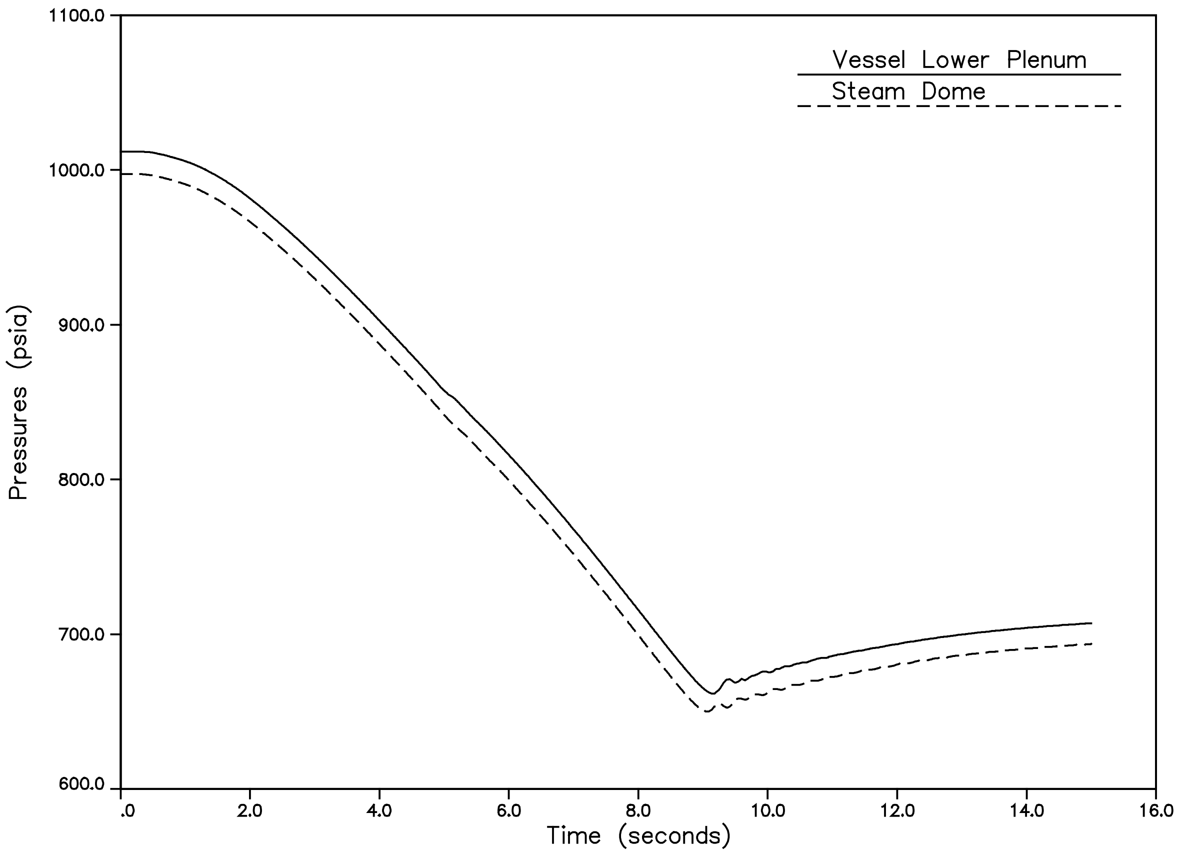


Figure 3.2 Browns Ferry Unit 1 PRFO Transient at 60P/35F – Vessel Pressures

4.0 Extending SPCB/GE14 Low Pressure Boundary

Since the PRFO event results in the depressurization of the reactor vessel, this event imposes a requirement that the critical power correlation support pressures lower than the normal operating pressure range.

Co-resident fuel is modeled with an approved AREVA critical power correlation according to the methodology described in Reference 11. Co-resident GE14 fuel is modeled with the SPCB correlation, Reference 7. The range of data used to construct additive constants for the Browns Ferry Unit 1 GE14 fuel did not extend below 700 psia for fuel loaded in Cycle 9. The range of data extended to 800 psia for fuel loaded prior to Cycle 9. This imposes a low pressure boundary on the SPCB/GE14 correlation of 700 psia (Cycle 9 fuel would be the only potentially limiting fuel type for the GE14 co-resident in future cycles), significantly higher than the SPCB correlation low pressure boundary of [ ].

AREVA analyses indicate the PRFO event can reach pressures below 700 psia, during which, the safety limit must be maintained. Normally, crossing a critical power pressure boundary requires assuming that onset of dryout has occurred. This is not an acceptable outcome for the PRFO event. In this section, a method allowing application of the SPCB/GE14 to pressures lower than 700 psia (but remaining within the application range of SPCB) is described and justified. The bases for this justification are:

- Observations of critical power behavior with pressure from the open literature
- Test data observations of critical power behavior as a function of pressure for ATRIUM-10
- SPCB critical power correlation behavior as function of pressure

Collier & Thome (Reference 12) show the influence of pressure on critical heat flux. When the test section is at the critical heat flux, the integrated heat flux over the heated surface area is the critical power. Their figure (reproduced in Figure 4.1) shows the characteristic expected behavior in the range of BWR pressure from 40 to 100 bar (approximately 580 to 1450 psia). The dashed line with the inlet subcooling set to zero is the most representative of BWR application. The critical heat flux increases monotonically as the pressure decreases, reaching a maximum near 500 to 600 psia. The curve with the solid line represents an unusual case.

The inlet temperature is fixed to the specified value of 174 °C. This means that as the pressure is increased, the inlet subcooling increases; the decreased inlet subcooling as the pressure is lowered (leading to lower critical power) appears to compete with the effect of pressure, where the critical power increases as the pressure is lowered.

Lahey & Moody (Reference 13) show the influence of pressure on critical power of BWR fuel (reproduced in Figure 4.2). It also shows that decreasing the pressure increases the critical power. The data includes two different flow rates and several peaking factors. There is a note in Reference 13, page 113 that says that the behavior continues as the pressure decreases until the trend reverses at a pressure less than 600 psia. Thus, the effect noted by Collier and Thome is observed to be present in BWR fuel assemblies.

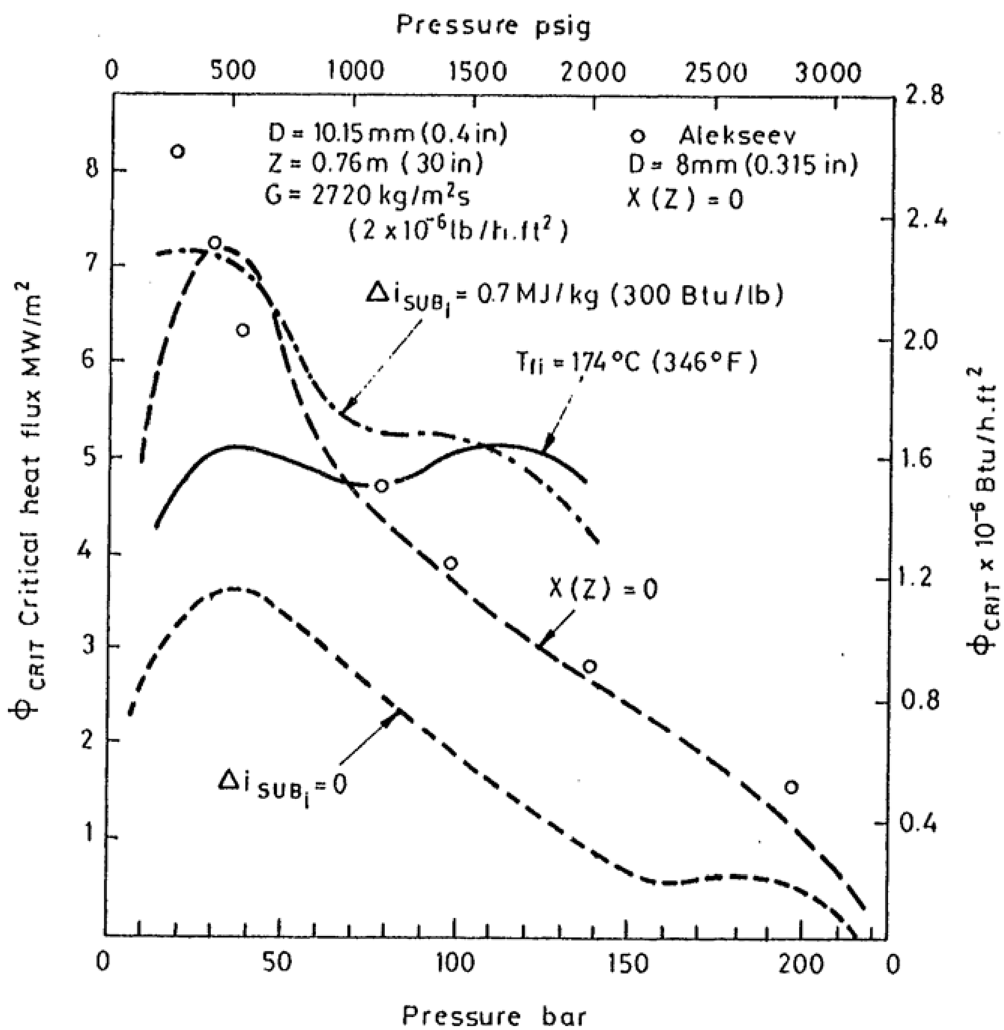
Pressure variation of ATRIUM-10 fuel design (test STS-17.8) with an inlet subcooling of approximately 20 Btu/lb and two flow rates are selected from Reference 7 and plotted in Figure 4.3. It shows the ATRIUM-10 critical power data trend with pressure is consistent with that of the open literature – critical power increases as the pressure is decreased.

The bases for the expected behavior of critical power with pressure have been established from the open literature and from BWR fuel critical power test data observations. Now consider the critical power correlation. The SPCB correlation critical power behavior as a function of pressure and flow rate is described in Reference 7, page 2-28. For the purpose of discussing the low pressure boundary of the SPCB correlation, the critical power is plotted as a function of pressure and mass flow rate with an inlet subcooling of 20 Btu/lb (Figure 4.4). The pressure is varied from 1000 psia to the lower boundary of the SPCB correlation. It shows that the SPCB correlation has the expected behavior – that as the pressure is decreased, the critical power increases.

The low pressure boundary of the SPCB/GE14 correlation (700 psia) is well within the range of the SPCB correlation. Thus, an alternative treatment for the low pressure boundary can be described. For pressures that are lower than the SPCB/GE14 700 psia correlation boundary, the critical power will be evaluated as though the pressure was at 700 psia (preserving the same inlet subcooling). The results of applying the SPCB/GE14 correlation to pressures lower than 700 psia is illustrated with dashed lines in Figure 4.5 and indicates that the alternative low pressure boundary treatment is conservative. By treating the boundary in this way, the

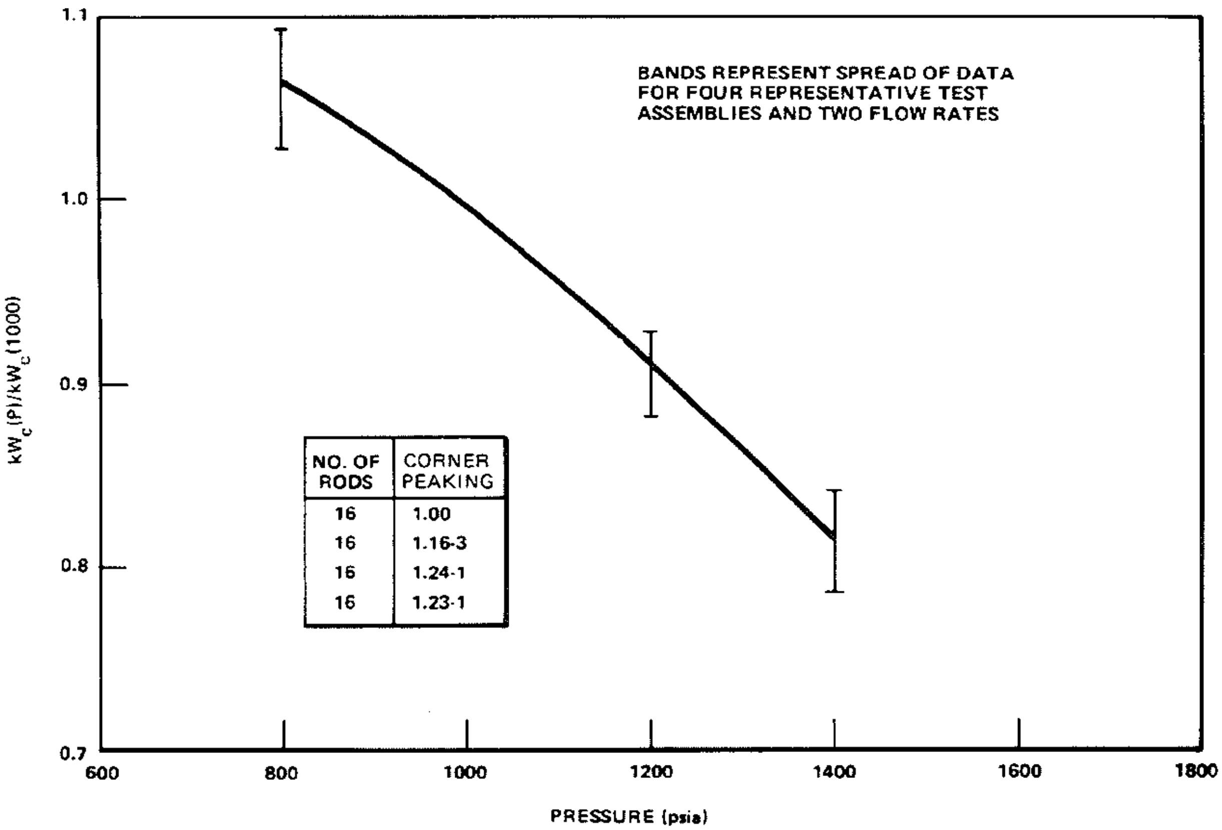
SPCB/GE14 correlation can be applied to system pressures as low as the SPCB correlation lower boundary on pressure.

This application of the SPCB/GE14 correlation to the SPCB lower boundary pressure [ ] supports the expected system pressure reduction associated with the PRFO event analysis.



Reproduced from Reference 12, Figure 8.13, page 362.

Figure 4.1 The Influence of System Pressure on Critical Heat Flux



Reproduced from Reference 13, Figure 4-36, page 116.

Figure 4.2 Normalized Critical Power versus Pressure

Figure 4.3 ATRIUM-10 Test STS-17.8 Critical Power versus Pressure

Figure 4.4 SPCB Correlation Critical Power as Function of Pressure and Flow Rate

[ ]

Figure 4.5 SPCB/GE14 Correlation With Alternative Treatment of Low Pressure Boundary

5.0 References

1. Letter, DB McBurney (AREVA) to BD McNelley (TVA), “Proposal for Evaluation of PRFO Low Pressure Technical Specification Value for Browns Ferry,” FAB11-2517, Proposal 2011001721, December 9, 2011.

2. 51-9107601-000, “Disposition of Events for Browns Ferry Unit 1,” AREVA NP, May 1, 2009.

3. Letter, T.A. Galioto (AREVA) to J.F. Lemons (TVA), “Licensing Basis Issues and Disposition of Events for BFN Unit 3 Cycle 12 – Revision 1,” TAG:03:140 FAB03-1387, December 22, 2003 (38-9107703-000).

4. Technical Specifications for Browns Ferry Nuclear Plant Unit 1, latest Revision.

5. Technical Specifications for Browns Ferry Nuclear Plant Unit 2, latest Revision.

6. Technical Specifications for Browns Ferry Nuclear Plant Unit 3, latest Revision.

7. EMF-2209(P)(A) Revision 3, SPCB Critical Power Correlation, AREVA NP, September 2009.

8. ANP-10298PA Revision 0, ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, March 2010.

9. Browns Ferry Nuclear Plant Final Safety Analysis Report, Amendment 24.

10. ANP-3107(P) Revision 1, Browns Ferry Unit 2 Cycle 18 Plant Parameters Document, AREVA NP, June 2012.

11. EMF-2245(P)(A) Revision 0, Application of Siemens Power Corporation’s Critical Power Correlations to Co-Resident Fuel, Siemens Power Corporation, August 2000.

12. J. G. Collier and J. R. Thome, “Convective Boiling and Condensation,” Third Edition, Oxford University Press, 1996.

13. R. T. Lahey, Jr., and F. J. Moody, “The Thermal-hydraulics of a Boiling Water Nuclear Reactor,” American Nuclear Society, 1977.

ATTACHMENT 7

Affidavit for Attachment 5

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

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

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

3. I am familiar with the AREVA NP information contained in the report ANP-3245P, Revision 1, "Browns Ferry Evaluation of PRFO Low Pressure Technical Specification Value," dated February 2014 and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP 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 AREVA NP 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 AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP'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 AREVA NP.

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

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

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 AREVA NP'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 AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP 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.

SUBSCRIBED before me this 7th

day of February, 2014.

Susan K. McCoy
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 1/14/2016

SUSAN K. MCCOY
COMMISSION EXPIRES
NOTARY
PUBLIC
01-14-16
STATE OF WASHINGTON