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

February 28, 2013

10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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: **Technical Specification Change TS-478 – Addition of Analytical Methodologies to Technical Specification 5.6.5 for Browns Ferry 1, 2, and 3, and Revision of Technical Specification 2.1.1.2 for Browns Ferry Unit 2, in Support of ATRIUM-10 XM Fuel Use at Browns Ferry**

In accordance with the provisions of Title 10 of the Code of Federal Regulations (10 CFR) 50.90, "Application for amendment of license, construction permit, or early site permit," the Tennessee Valley Authority (TVA) is submitting a request for amendments to the Technical Specifications (TS) for Browns Ferry Nuclear Plant, Units 1, 2 and 3.

The proposed amendment adds three additional AREVA NP analysis methodologies to the list of approved methods to be used in determining core operating limits in the Core Operating Limits Report. In addition, the amendment requests a change to the Safety Limit Minimum Critical Power Ratio value for Unit 2.

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. Attachment 1 of the enclosure to this letter identifies a regulatory commitment. Attachment 2 provides the existing Unit 1 TS pages marked-up to show the proposed changes. Attachment 3 shows the existing Unit 1 TS pages retyped to show the proposed changes. Attachment 4 provides Unit 1 TS Bases pages marked up to show the associated proposed changes. Attachment 5 provides Unit 1 TS Bases pages retyped to show the associated proposed changes.

In support of the proposed TS changes, certain technical information related to the transition core design and licensing analyses, as well as information related to the AREVA analysis methodologies, has been provided in Attachments 6 through 23 of this submittal. The information attached to this submittal is based on non Extended Power Uprate (EPU) conditions only.

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Attachments 6, 8, 10, 12, 14, 17, 19, 21, 23, 25 and 27 contain 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 29 provides the affidavits supporting this request. Attachments 7, 9, 11, 13, 15, 16, 18, 20, 22, 24, 26, and 28 contain the redacted versions of the proprietary attachments with the proprietary material removed, which are suitable for public disclosure.

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.

TVA requests approval of these TS changes by February 28, 2014.

There is one new regulatory commitment in this submittal as reflected in Attachment 1 of the enclosure.

Please direct any questions concerning this matter to Tom Hess at (423) 751-3487.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on the 28th day of February 2013.

Respectfully,

J. W. Shea
Vice President Nuclear Licensing

Enclosure: Addition of Analytical Methodologies to Technical Specification 5.6.5.b for Browns Ferry 1, 2, & 3, and Revision of Technical Specification 2.1.1.2 for Browns Ferry Unit 2, in Support of ATRIUM-10 XM Fuel Use at Browns Ferry

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

Browns Ferry Nuclear Plant (BFN)
Units 1, 2, 3

Technical Specifications (TS) Change TS-478

Addition of Analytical Methodologies to Technical Specification 5.6.5.b for Browns Ferry 1, 2, & 3, and Revision of Technical Specification 2.1.1.2 for Browns Ferry Unit 2, in Support of ATRIUM-10 XM Fuel Use at Browns Ferry

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 the Operating Licenses to add three additional AREVA analytical methodologies for all three BFN units, to support a planned transition to ATRIUM-10 XM fuel design (hereafter referred to as XM). Tennessee Valley Authority (TVA) intends to transition BFN to the XM fuel design commencing with the Unit 2 cycle 19 reload batch delivered in the spring of 2015. Unit 3 would transition to XM fuel in the spring of 2016, followed by Unit 1 in the fall of 2016. This evaluation supports the transition to XM fuel for non Extended Power Uprate (EPU) power conditions (i.e., 105% Original Licensed Power Level (OLTP)) only.

In addition, a change to the Unit 2 Safety Limit Minimum Critical Power Ratio (SLMCPR) Technical Specification (TS) values is being requested by TVA. The revised SLMCPR values reflect a conservative reduction from the current values, supported by the application of one of the new methodologies being proposed for addition to the Technical Specifications.

TVA requests approval of the proposed amendments by February 28, 2014, to support the Unit 2 cycle 19 reload design and fuel fabrication schedule milestone dates.

2.0 DETAILED DESCRIPTION

TVA intends to begin utilizing the XM design in all three BFN units, beginning in the spring of 2015. The XM product is a proven fuel design in use at European reactors since 2005, and is being successfully used at two domestic units. The XM fuel design complies with the requirements outlined in Reference 1. The NRC has audited the compliance of the XM fuel design against the Reference 1 requirements in November 2010, and did not identify any issues.

The initial XM reloads will continue to utilize Blended Low Enriched Uranium (BLEU) provided to TVA under a joint project with the Department of Energy. However, TVA may also elect to utilize XM fuel in the BFN cores with standard commercial grade uranium in future reloads. At this time, TVA is requesting approval for application of the three additional NRC approved methodologies for non EPU conditions (i.e., 105% OLTP). Once NRC approval is granted for the use of the three additional methodologies, TVA will implement the XM design under the 10CFR50.59 process.

TS-478 requests approval for the application of three additional methodologies discussed below, for all three BFN units. To support the NRC review of the usage of these methodologies at BFN, a transition XM core design has been developed. This design is for the lead cycle application, which is Unit 2 cycle 19. This Unit 2 transition design is representative of an XM transition for the other two units, since for all three units the balance of the core will be standard ATRIUM-10 fuel when the first XM reload is inserted. While Unit 1 currently has a mix of ATRIUM-10 and GE14 fuel, all of the GE14 fuel will be discharged in the fall 2016 outage, when XM would be introduced. Therefore, there are no mixed vendor core issues associated with this license amendment request (LAR).

In order to allow the use of the XM fuel design at BFN, changes to the TS are required. TS 5.6.5.b address the analytical methods that may be used to determine input to the Core Operating Limits Report (COLR). This Technical Specification currently includes all of the AREVA methodologies necessary to analyze the ATRIUM-10 fuel currently being used in all three units. The use of XM fuel requires the addition of three other NRC approved AREVA methodologies to this Technical Specification. The additional methodologies are:

- RODEX4 for fuel rod thermal mechanical analyses
- ACE correlation for critical power monitoring of XM fuel
- SAFLIM-3D for Safety Limit MCPR (SLMCPR) analyses

The application of the RODEX4 code is limited to fuel rod mechanical analysis. The RODEX2 code will still be used to support transient and LOCA analyses. The RODEX4 code provides a realistic evaluation of the fuel rod mechanical response during both normal operation and anticipated operational occurrences. RODEX4 accounts for the degradation of thermal conductivity with increasing fuel burnup. The RODEX2 code, as currently approved, does not account for degradation of pellet conductivity with increasing fuel burnup. A BFN specific evaluation of the effects of not accounting for conductivity degradation is included in the XM LAR as Attachments 19 and 20.

During the review of the LAR submittal discussed in section 4.2, the NRC raised a concern over oxide thickness and hydride formation. The specific concern is that excessive oxide and hydriding could create non uniform clad properties that the fuel rod thermal mechanical code does not account for. In approving the use of RODEX4 for this similar LAR, an upper bound on the calculated oxide was proposed by the licensee and subsequently accepted by the NRC. This upper bound was chosen based on the AREVA Boiling Water Reactor (BWR) fuel inspection database that indicates that oxide spallation has not been observed at oxide values at or below this upper bound value. BFN will apply the RODEX4 methodology using this same upper limit on calculated clad oxide thickness. The specific value is proprietary, and is discussed in Attachment 21. A commitment to use this upper bound calculated value is provided in Attachment 1.

The AREVA Critical Power Evaluator (ACE) critical power correlation will only be applied to XM fuel. The Siemens Power Corporation (SPCB) correlation currently being used at BFN will continue to be applied to the co-resident ATRIUM-10 fuel.

The current ACE correlation for XM fuel has an identified deficiency. The deficiency involves a non conservatism in the axial averaging process used to determine the K factor, which is an input to the correlation. AREVA has docketed a generic ACE supplement report with the NRC (reference 3), which provides a correction to the methodology. Given the uncertainty in the timing of the NRC approval of the generic ACE supplement, TVA is including a BFN specific

ACE supplement in Attachments 27 and 28. If the NRC approves the generic ACE supplement ahead of the approval of the Browns Ferry XM LAR, TVA will supplement the XM LAR with a request to have BFN Technical Specification 5.6.5.b revised to reference the generic ACE supplement. If the BFN XM LAR is approved prior to NRC approving the ACE generic supplement, TVA will request that the reference for ACE in TS 5.6.5.b be annotated to reference the Safety Evaluation for TS-478.

The value for the SLMCPR in Technical Specification 2.1.1.2 for BFN Unit 2, will also be modified. This change is driven by the introduction of the AREVA SAFLIM-3D SLMCPR methodology. Technical Specifications change TS-478 will only change the SLMCPR value for Unit 2, since the transition work submitted in this LAR is based on Unit 2 Cycle 19. The SLMCPR values proposed for Unit 2 are conservatively selected relative to the values discussed in Attachments 25 and 26, as the intention is to establish values that will not routinely require changing for future reloads on the unit. TVA plans to submit separate SLMCPR LARs for Units 1 and 3, commensurate with the design and licensing schedules for the initial XM reloads of those units.

The submittal also addresses the required changes to the Technical Specification Bases. The changes are related to adding information pertaining to AREVA analytical methodologies (including reference documents).

In a meeting with the NRC staff on June 18, 2012 (reference 2), the overall approach for the BFN XM fuel transition submittal was discussed. In the meeting, TVA stated that the XM LAR would include similar technical information as was provided in TS-473 for the Unit 1 ATRIUM-10 transition. Attachment 1 contains the regulatory commitment associated with this LAR. Attachments 2 thru 5 contain the changes to the Technical Specifications and Bases. The documents in Attachments 6 thru 16 are consistent with the reports submitted under TS-473. The Loss of Coolant Accident (LOCA) reports in Attachments 14 and 15 were developed using the modified LOCA method reviewed for BFN under TS-473. The Licensing Methodology Compendium in Attachment 16 has been updated to include the SAFLIM-3D and RODEX4 methodologies (ACE was included in the prior revision docketed under TS-473).

TVA has reviewed the NRC Request for Additional Information (RAIs) covered in the prior TS-473 LAR (AREVA report ANP-2860), and has readdressed the questions whose answers have a fuel type dependency, in the context of XM fuel. In addition, TVA has screened the RAIs from the Brunswick XM LAR for applicability (see section 4.2). A compendium of the responses to the applicable prior RAIs is included in Attachments 17 and 18.

The new methodologies utilized for XM fuel require additional analyses to be performed as part of the design and licensing of a core, compared to what is done for standard ATRIUM-10 reloads on Browns Ferry. The RODEX4 method involves a statistical evaluation of fuel rod power histories from the design cycle, as well as an equilibrium cycle. The fuel rod thermal mechanical analysis is documented in a separate report, and is included in Attachments 21 and 22. The fuel cycle design report for the equilibrium fuel cycle design used in the RODEX4 fuel rod mechanical evaluation is included in Attachments 23 and 24.

Attachments 25 and 26 document the plant specific application of the SAFLIM-3D method for the lead XM reload, Unit 2 cycle 19. This analysis supports the SLMCPR value for TS 2.1.1.2 as shown in Attachments 2 and 3.

As mentioned previously, TVA is also providing a BFN specific ACE supplement in this LAR, to address the issue with the K factor methodology. The BFN specific ACE supplement is included in Attachments 27 and 28.

The specific document titles and dates are shown in the table below for convenience.

Attachment	Title
1	Regulatory Commitments
2	Proposed Technical Specifications Changes (Mark-up)
3	Retyped Proposed Technical Specifications Pages
4	Proposed Technical Specification Bases Changes (Mark-up)
5	Retyped Proposed Technical Specification Bases Pages
6	Mechanical Design Report (*proprietary*) ANP-3150(P), Revision 0, Mechanical Design Report for Browns Ferry ATRIUM 10XM Fuel Assemblies, AREVA NP Inc., October 2012
7	Mechanical Design Report (*non-proprietary*) ANP-3150(NP), Revision 0, Mechanical Design Report for Browns Ferry ATRIUM 10XM Fuel Assemblies, AREVA NP Inc., October 2012
8	Thermal Hydraulic Design Report (*proprietary*) ANP-3082(P), Revision 1, Browns Ferry Thermal-Hydraulic Design Report for ATRIUM 10XM Fuel Assemblies, AREVA NP Inc., August 2012
9	Thermal Hydraulic Design Report (*non-proprietary*) ANP-3082(NP), Revision 1, Browns Ferry Thermal-Hydraulic Design Report for ATRIUM 10XM Fuel Assemblies, AREVA NP Inc., August 2012
10	Fuel Cycle Design Report (*proprietary*) ANP-3145(P), Revision 0, Browns Ferry Unit 2 Cycle 19 LAR Fuel Cycle Design, AREVA NP Inc., August 2012
11	Fuel Cycle Design Report (*non-proprietary*) ANP-3145(NP), Revision 0, Browns Ferry Unit 2 Cycle 19 LAR Fuel Cycle Design, AREVA NP Inc., August 2012
12	Reload Safety Analysis Report (*proprietary*) ANP-3167(P), Revision 0, Browns Ferry Unit 2 Cycle 19 Reload Analysis, AREVA NP Inc., November 2012
13	Reload Safety Analysis Report (*non-proprietary*) ANP-3167(NP), Revision 0, Browns Ferry Unit 2 Cycle 19 Reload Analysis, AREVA NP Inc., November 2012
14	LOCA Break Spectrum Analysis Report (*proprietary*) ANP-3152(P), Revision 0, Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel, AREVA NP Inc., October 2012
15	LOCA Break Spectrum Analysis Report (*non-proprietary*) ANP-3152(NP), Revision 0, Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel, AREVA NP Inc., October 2012

Attachment	Title
16	Boiling Water Reactor Licensing Methodology Compendium (*non-proprietary*) ANP-2637, Revision 4, Boiling Water Reactor Licensing Methodology Compendium, AREVA NP Inc., November 2012
17	RAI Compendium (proprietary) ANP-2860P Revision 2, Supplement 1P Revision 0, Browns Ferry Unit 1 Summary of Responses to Request for Additional Information - Extension for ATRIUM-10 XM, AREVA NP Inc., November 2012
18	RAI Compendium (non-proprietary) ANP-2860NP Revision 2, Supplement 1NP Revision 0, Browns Ferry Unit 1 Summary of Responses to Request for Additional Information - Extension for ATRIUM-10 XM, AREVA NP Inc., November 2012
19	Thermal Conductivity Degradation Report (*proprietary*) ANP-3170(P), Revision 0, Evaluation of Fuel Conductivity Degradation for ATRIUM 10XM Fuel for Browns Ferry Units 1, 2, and 3, AREVA NP Inc., November 2012
20	Thermal Conductivity Degradation Report (*non-proprietary*) ANP-3170(NP), Revision 0, Evaluation of Fuel Conductivity Degradation for ATRIUM 10XM Fuel for Browns Ferry Units 1, 2, and 3, AREVA NP Inc., November 2012
21	Fuel Rod Thermal Mechanical Evaluation Report (*proprietary*) ANP-3159(P), Revision 0, ATRIUM-10XM Fuel Rod Thermal-Mechanical Evaluation for Browns Ferry Unit 2 Cycle 19 Reload BFE2-19, AREVA NP Inc., October 2012
22	Fuel Rod Thermal Mechanical Evaluation Report (*non-proprietary*) ANP-3159(NP), Revision 0, ATRIUM-10XM Fuel Rod Thermal-Mechanical Evaluation for Browns Ferry Unit 2 Cycle 19 Reload BFE2-19, AREVA NP Inc., October 2012
23	Equilibrium Fuel Cycle Design Report (*proprietary*) ANP-3148(P), Revision 0, Browns Ferry ATRIUM 10XM Equilibrium Cycle Design Summary, AREVA NP Inc., August 2012
24	Equilibrium Fuel Cycle Design Report (*non-proprietary*) ANP-3148(NP), Revision 0, Browns Ferry ATRIUM 10XM Equilibrium Cycle Design Summary, AREVA NP Inc., August 2012
25	Browns Ferry Unit 2 Cycle 19 SLMCPR Analysis (*proprietary*) 51-9191258-001, Revision 1, Browns Ferry Unit 2 Cycle 19 MCPR Safety Limit Analysis With SAFLIM3D Methodology, October 2012
26	Browns Ferry Unit 2 Cycle 19 SLMCPR Analysis (*non-proprietary*) 51-9191259-001, Revision 1, Browns Ferry Unit 2 Cycle 19 MCPR Safety Limit Analysis With SAFLIM3D Methodology, October 2012
27	BFN ACE Supplement (*proprietary*) 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 Inc., August 2012

Attachment	Title
28	BFN ACE Supplement (*non proprietary*) ANP-3140(NP), Revision 0, Browns Ferry Units 1, 2 and 3, Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP Inc., August 2012
29	Affidavits

3.0 TECHNICAL EVALUATION

The XM fuel design utilizes a 10x10 array of fuel rods, with seventy nine (79) full length fuel rods and twelve (12) partial length fuel rods. The active fuel length of the partial length fuel rods is approximately one half the length of the full length fuel rods. The use of partial length rods improves fuel utilization in the high void upper region of the bundle, and also enhances cold shutdown margin, stability, and pressure drop performance. Relative to ATRIUM-10, the partial length fuel rods are shorter, and their radial placement within the assembly is different.

The fuel rod diameter for XM fuel is increased relative to ATRIUM-10, with a corresponding increase in fuel pellet diameter. The fuel spacers utilize an inconel alloy. The number of spacers in the bundle is increased, and the axial placement of the fuel spacers is optimized to enhance critical power performance. The central water channel of the XM design includes the use of water crowns to improve critical power performance, by redirecting flow off the central water channel back into the rodded area of the assembly.

Similar to the ATRIUM-10 design, XM does not utilize tie rods as the structural tie between the upper and lower tie plates. Instead, the XM design uses a central water channel, having a mechanical connection to the two tie plates. The central water channel carries the mechanical loads during fuel handling. It displaces a 3x3 array of fuel rods within the bundle and serves to improve fuel economy by improving internal neutron moderation. The lower ends of the fuel rods rest on top of the lower tie plate, with their lower ends laterally restrained by a spacer grid located just above the lower tie plate. No expansion springs are required on each fuel rod because a single, large reaction spring is used on the central water channel to hold the upper tie plate in the latched position. The XM design to be employed at BFN utilizes an improved debris resistant lower tie plate filter to limit introduction of foreign material into the assembly from below.

Three additional AREVA analytical methodologies (RODEX4, ACE, and SAFLIM-3D) are to be added to Technical Specification 5.6.5.b as part of this LAR. Each of these analytical methodologies has been previously reviewed and approved by the NRC.

The impact of the XM design on the Updated Final Safety Analysis Report (UFSAR) accident analyses will be accounted for by cycle specific reload and accident analyses. Limiting transients from UFSAR Chapter 14 categories of pressure increase events, vessel water temperature decrease events, control rod withdrawal error events, core flow increase events, and increase in vessel inventory events are evaluated each cycle. Limiting analyses results, for a representative transition cycle, are presented in Attachments 12 and 13. These attachments also contain a disposition of the transient and accident events for BFN, to determine which events are potentially limiting using XM fuel and AREVA analysis methods. Some of these events are screened out as non-limiting, others are evaluated one time on a cycle independent basis, while others are determined to require cycle specific analysis.

One event not discussed in the disposition of events section of Attachments 12 and 13 is the loss of stator cooling transient (LOSC). This event has always been considered non limiting for Browns Ferry, and the event is not discussed in the UFSAR. However, in August 2012, GE Hitachi issued Rapid Information Communication Services Information Letter (RICSIL) 094, on the subject of an emerging issue related to LOSC being a potentially limiting event for some plants. The issue concerns plants where a loss of stator cooling results in a dual recirculation pump trip with a subsequent turbine load runback, resulting in slow closure of the turbine control valves, a rise in reactor pressure, and termination of the event by a scram on high reactor pressure or high neutron flux. This scenario could represent a limiting, unanalyzed event if the plant hardware functions as described above. TVA independently evaluated the event scenario discussed in RICSIL 094 under its corrective action program, and has concluded that the LOSC event remains non-limiting for BFN. This conclusion is based on the fact that all three BFN units receive an automatic turbine trip signal off of a loss of stator cooling signal, resulting a direct reactor scram from the turbine trip. Per the recommendations in the RICSIL, plants like Browns Ferry with an automatic reactor trip in response to the LOSC signal do not require further evaluation, and no further actions are required.

Introduction of the XM design fuel will not adversely impact UFSAR accident analyses. AREVA evaluates the control rod drop accident (UFSAR section 14.6.2) on a cycle specific basis. Attachment 12 includes a cycle specific evaluation of the control rod drop accident for a representative transition cycle. The evaluation shows the number of rods calculated to fail in this event remains well below the value of 850 assumed in the UFSAR radiological evaluation of this event. The doses, from the control rod drop accident, remain within limits required by 10 CFR 50.67, "Accident source term," and Regulatory Guide 1.183 (Reference 6).

Regarding the LOCA analysis (UFSAR section 14.6.3), a baseline LOCA break spectrum analysis of XM fuel has been performed, covering all three BFN units at 105% OLTP power conditions; it is included as Attachments 14 and 15. Cycle specific fuel design Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits are analyzed consistent with assumptions used in the baseline LOCA analysis. Peak cladding temperature, cladding oxidation, and hydrogen generation analyses results of record are included in Attachment 12 for a representative transition cycle. The introduction of XM fuel will not challenge the peak clad temperature, cladding oxidation, or hydrogen generation limits specified in 10 CFR 50.46 "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," paragraph (b).

The XM design will not challenge the UFSAR basis of the refueling accident (UFSAR section 14.6.4). The BFN UFSAR accident is based on a bounding event using a 7x7 fuel design. While the number of rods calculated to fail in a fuel handling accident involving an XM bundle (163) is higher than the number calculated to fail in a 7x7 bundle (111), the activity is allocated over a greater number of rods. The XM bundle has the equivalent of 85 full length fuel rods (79 full length plus 12 partial length rods with approximately one half the full length), while the 7x7 bundle has 49 full length rods. Therefore, applying a simple ratio approach, TVA has concluded that the accident release with XM fuel would be approximately (163/111) x (49/85), or approximately 85% of the release from the design basis 7x7 fuel. Also, the isotopes of interest remain bounded by the current analysis except for Kr-85. However, since Kr-85 accounts for a negligible amount of the total dose and the other isotopes have decreased, the overall effect would be a lower Total Effective Dose Equivalent (TEDE). Consequently, the fuel handling accident described in the UFSAR remains bounding for ATRIUM-10XM fuel. The doses resulting from this event will remain within the limits specified in 10 CFR 50.67.

The main steam line break accident (UFSAR section 14.6.5) is not affected by a change in fuel design. As stated in the UFSAR, no fuel failures are expected to occur as a result of this accident. The radionuclide inventory, released from the primary coolant system, is that present in the coolant prior to the event; UFSAR section 14.6.5.2.1 provides details regarding the assumed accident inventory. Therefore, the fuel design change does not alter the dose consequences of a main steam line break accident.

It should also be noted that the similar plant discussed in section 4.2 did not request addition of the SAFLIM-3D methodology to their Technical Specifications at the time of their initial XM LAR, due to the fact that SAFLIM-3D was not NRC approved at that time. TVA is including the SAFLIM-3D methodology as part of the BFN XM LAR request, as the method has now been approved by the NRC. It should be noted that the similar plant has subsequently filed an LAR requesting the addition of the SAFLIM-3D methodology to their Technical Specifications (see section 4.2 item 3).

In summary, the XM fuel design fully complies with applicable fuel licensing criteria provided in Reference 4, as documented in Reference 1 as applied to the XM design. The analytical methodologies to be used for design and licensing of XM reloads are all NRC approved, and acceptable for establishing COLR limits. Application of these methods will be in compliance with the restrictions identified by the NRC staff during the August 2008 review of the AREVA analytical methods.

4.0 REGULATORY EVALUATION

4.1 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

The XM fuel design was developed using the thermal mechanical design bases and limits outlined in Reference 1. Compliance with Reference 1 ensures the fuel design meets the regulatory requirements for fuel system damage, fuel failure, and fuel coolability criteria identified in the Reference 4 Standard Review Plan. The NRC reviewed and approved (per Reference 5) the use of Reference 1 for making changes and improvements to fuel designs; specifically stating such changes and improvements do not require specific NRC review and approval, provided the criteria are satisfied. The XM design fully complies with the criteria of Reference 1, and therefore meets all of the required regulatory and licensing criteria in the Reference 4 Standard Review Plan.

4.2 PRECEDENT

The NRC has previously reviewed and approved the application of RODEX-4 and ACE (see items 1 and 2 below). These previous reviews confirmed the acceptability of transitioning from ATRIUM-10 fuel to XM, and the application of RODEX4 and ACE, to a similar plant. The scope of the technical analyses provided in support of the BFN XM submittal is consistent with, and surpasses that of the analyses provided in the submittals reviewed in items 1 thru 3 below. In addition, the plant in items 1 thru 3 below is very similar to BFN, in that both are BWR/4 plants with D lattice cores. The similar plant is licensed for EPU, while the BFN XM LAR is only requesting approval at the current power level of 105% of original licensed power. As such, TVA believes that any issues reviewed in the submittals in items 1 thru 3 below pertaining to EPU specific aspects do not apply to the review of the BFN XM LAR.

It should be noted that the similar plant submittals involved a core that contained limited amounts of GE14 legacy fuel in low importance areas of the core, while the XM transition at Browns Ferry will only have ATRIUM-10 legacy fuel in core with the XM fuel. TVA does not believe this minor difference should affect any conclusions drawn from the review of the similar plant submittals listed below.

1. "Brunswick Steam Electric Plant, Units 1 and 2 – Issuance of Amendments Regarding Addition of Analytical Methodology Topical Report to Technical Specification 5.6.5 (TAC Nos. ME3858 and ME3859) April 8, 2011. (ML11101A043)
2. "Brunswick Steam Electric Plant, Units 1 and 2 – Issuance of Amendments Regarding Addition of Analytical Methodology Topical Report to Technical Specification 5.6.5 (TAC Nos. ME3856 and ME3857) April 8, 2011. (ML111010234)
3. "Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Renewed Facility Operating License Nos. DPR-71 and DPR-62, Docket Nos. 50-325 and 50-324, Request for License Amendments – Addition of Analytical Methodology Topical Report to Technical Specification 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)" and Revision to Technical Specification 2.1.1.2 Minimum Critical Power Ratio Safety Limit" March 6, 2012 (ML12076A062)

4.3 SIGNIFICANT HAZARDS CONSIDERATIONS

This analysis addresses the proposed change to amend Operating Licenses DPR-33, DPR-52, and DPR-68 for BFN to allow the use of three additional AREVA fuel and analytical methodologies. In addition, Operating License DPR-52 changes the SLMCPR value to reflect the use of the improved AREVA safety limit MCPR methodology.

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 10CFR 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

Changing the fuel design, adding the additional approved methodologies to the Technical Specifications, and revising the unit 2 SLMCPR value in the Technical Specifications will not increase the probability of a LOCA. The fuel cannot increase the probability of a primary coolant system breach or rupture, as there is no interaction between the fuel and the system piping. The fuel will continue to meet the 10 CFR 50.46 limits for peak clad temperature, oxidation fraction, and hydrogen generation. Therefore, the consequences of a LOCA will not be increased.

Similarly, changing fuel type and revising the Technical Specifications as proposed cannot increase the probability of an abnormal operating occurrence (AOO). As a passive component, the fuel does not interact with plant operating or control systems. Therefore, the fuel change cannot affect the initiators of the previously evaluated AOO transient events. Thermal limits for the new fuel will be determined on a reload specific basis, ensuring the specified acceptable fuel design limits continue to be met. Therefore, the consequences of a previously evaluated AOO will not increase.

The refueling accident is potentially affected by a change in fuel design, due to the mechanical interaction between the fuel and the refueling equipment. However, the probability of the refueling accident with XM fuel is not increased because the upper bail handle is designed to be mechanically compatible with existing fuel handling equipment. The design weight of the XM design is similar to other designs in use at BFN, and is well within the design capability of the refueling equipment. The consequences of the refueling accident are similar to the current ATRIUM-10 fuel, remaining well within the design basis (7x7 fuel) evaluation in the UFSAR.

The probability of a control rod drop accident does not increase because the XM fuel channel is mechanically compatible with the co-resident ATRIUM-10 fuel, and the existing control blade designs. The mechanical interaction and friction forces between the XM fuel channel, and control blades, would not be higher than previous designs. In addition, routine plant testing includes confirmation of adequate control blade to control rod drive coupling. The probability of a rod drop accident is not increased with the use of XM fuel. Control rod drop accident consequences are evaluated on a cycle specific basis, confirming the number of calculated fuel rod failures remains with the UFSAR design basis.

The dose consequences of all the previously evaluated UFSAR accidents remain with the limits of 10 CFR 50.67.

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

Response: No

The XM fuel product has been designed to maintain neutronic, thermal-hydraulic, and mechanical compatibility with the NSSS vendor fuel designs. The XM fuel has been designed to meet fuel licensing criteria specified in NUREG-0800, "Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants." Compliance with these criteria ensures the fuel will not fail in an unexpected manner.

A change in fuel design and revising the Technical Specifications as proposed cannot create any new accident initiators because the fuel is a passive component, having no direct influence on the performance of operating plant systems and equipment. Hence, a fuel design change cannot create a new type of malfunction leading to a new or different kind of transient or accident.

Consequently, the proposed fuel design change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No

The XM fuel is designed to comply with the fuel licensing criteria specified in NUREG-0800. Reload specific and cycle independent safety analyses are performed ensuring no fuel failures will occur as the result of abnormal operational transients, and dose consequences for accidents remain with the bounds of 10 CFR 50.67. All regulatory margins and requirements are maintained.

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 use of XM fuel (using BLEU or commercial grade uranium), the adoption of the ACE, SAFLIM-3D, and RODEX4 analytical methodologies for BFN, and the proposed change to the Unit 2 SLMCPR value are acceptable based on the following:

- XM fuel has been designed to comply with the fuel related licensing criteria specified in the Standard Review Plan (Reference 4).
- The three new analytical methodologies being added to the Technical Specifications have been previously reviewed and approved by NRC.
- The remaining analytical methodologies already in the Technical Specifications have been reviewed by the NRC and found to be acceptable for application at BFN, with the caveat of two restrictions related to vessel overpressure margins. These two restrictions have been incorporated into the XM transition analyses.
- Transition core design analyses demonstrate the acceptability of using XM fuel in BFN.
- The revised SLMCPR value for Unit 2 is supported by the NRC approved SAFLIM-3D method, and the value has been chosen conservatively to ensure that no more than 0.1% of rods experience transition boiling.

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. ANF-89-98(P)(A) Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Designs," Advanced Nuclear Fuels Corporation, dated May 1995.

2. Letter from Ms. Eva A. Brown (NRC) to Douglas A. Broaddus (NRC), "Forthcoming Meeting with Tennessee Valley Authority (TVA) Regarding AREVA XM Fuel Transition Request" dated May 29, 2012.

3. ANP-10298PA, Revision 0, Supplement 1P, Revision 0. "Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation", December 2011.

4. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants:LWR Edition," Section 4.2, 'Fuel System Design,' Revision 3, dated March 2007.

5. Letter from R.C. Jones (NRC) to R. Copeland (Siemens Power Corporation), "Acceptance for Referencing of Topical Report ANF-89-98(P), Revision 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," (TAC No. M81070)," dated April 20, 1995.

6. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," U.S. Nuclear Regulatory Commission, dated July 2000.

ATTACHMENT 1

Browns Ferry Nuclear Plant (BFN)
Units 1, 2, and 3

Technical Specifications (TS) Change 478

Addition of Analytical Methodologies to Technical Specification 5.6.5.b for Browns Ferry 1, 2, & 3, and Revision of Technical Specification 2.1.1.2 for Browns Ferry Unit 2, in Support of ATRIUM-10 XM Fuel Use at Browns Ferry

Regulatory Commitments

Commitment	Completion Date
When using AREVA Topical Report BAW-10247PA, *Realistic Thermal Mechanical Fuel Rod Methodology for Boiling Water Reactors*, Revision 0, February 2008, to determine core operating limits, the fuel cladding peak oxide thickness calculated by RODEX4 will be limited to less than the proprietary value defined in section 3.2.7 of AREVA report ANP-3159P revision 0, dated October 2012.	Upon implementation of the Unit 1, 2, and 3 license amendments authorizing the incorporation of AREVA Topical Report BAW-10247PA into Technical Specification 5.6.5.b.

ATTACHMENT 2

Browns Ferry Nuclear Plant (BFN)
Units 1, 2, and 3

Technical Specifications (TS) Change 478

Addition of Analytical Methodologies to Technical Specification 5.6.5.b for Browns Ferry 1, 2, & 3, and Revision of Technical Specification 2.1.1.2 for Browns Ferry Unit 2, in Support of ATRIUM-10 XM Fuel Use at Browns Ferry

Proposed Technical Specifications Changes (Mark-up)

The following pages have been revised to reflect the proposed changes. On the affected pages a line has been drawn through the deleted text and new or revised text is shaded.

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model, Siemens Power Corporation, February 1998.

5. ANF-89-98(P)(A) Revision 1 and Supplement 1, Generic Mechanical Design Criteria for BWR Fuel Designs, Advanced Nuclear Fuels Corporation, May 1995.

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

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

8. EMF-2158(P)(A) Revision 0, Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURNB2, Siemens Power Corporation, October 1999.

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

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

11. **ANP-10307PA Revision 0, AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, AREVA NP, June 2011.**~~ANF-524(P)(A) Revision 2 and Supplements 1 and 2, ANF Critical Power Methodology for Boiling Water Reactors, Advanced Nuclear Fuels Corporation, November 1990.~~

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

20. **BAW-10247PA Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, AREVA NP, February 2008.**

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

22. **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.**

(continued)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

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

MCPR shall be ≥ 1.06~~1.08~~ for two recirculation loop operation or ≥ 1.08~~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.

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. XN-NF-80-19(P)(A) Volume 1, Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis.

7. XN-NF-80-19(P)(A) Volume 4, Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads.

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

9. XN-NF-80-19(P)(A) Volume 3, Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description.

10. XN-NF-84-105(P)(A) Volume 1, XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis.

11. **ANP-10307PA Revision 0, AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, AREVA NP, June 2011.**~~ANF-524(P)(A), ANF Critical Power Methodology for Boiling Water Reactors.~~

12. ANF-913(P)(A) Volume 1, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses.

13. ANF-1358(P)(A), The Loss of Feedwater Heating Transient in Boiling Water Reactors.

14. EMF-2209(P)(A), SPCB Critical Power Correlation.

15. EMF-2245(P)(A), Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel.

16. EMF-2361(P)(A), EXEM BWR-2000 ECCS Evaluation Model.

17. EMF-2292(P)(A), ATRIUM^{TM}-10: Appendix K Spray Heat Transfer Coefficients.

(continued)

5.6 Reporting Requirements (continued)

CORE OPERATING LIMITS REPORT (COLR) (continued)

18. **BAW-10247PA Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, AREVA NP, February 2008.**

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

20. **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.**

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. XN-NF-80-19(P)(A) Volume 1, Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis.

7. XN-NF-80-19(P)(A) Volume 4, Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads.

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

9. XN-NF-80-19(P)(A) Volume 3, Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description.

10. XN-NF-84-105(P)(A) Volume 1, XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis.

11. **ANP-10307PA Revision 0, AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, AREVA NP, June 2011.**~~ANF-524(P)(A), ANF Critical Power Methodology for Boiling Water Reactors.~~

12. ANF-913(P)(A) Volume 1, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses.

13. ANF-1358(P)(A), The Loss of Feedwater Heating Transient in Boiling Water Reactors.

14. EMF-2209(P)(A), SPCB Critical Power Correlation.

15. EMF-2245(P)(A), Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel.

16. EMF-2361(P)(A), EXEM BWR-2000 ECCS Evaluation Model.

17. EMF-2292(P)(A), ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

18. **BAW-10247PA Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, AREVA NP, February 2008.**
19. **ANP-10298PA Revision 0, ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, March 2010.**
20. **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.**

(continued)

ATTACHMENT 3

Browns Ferry Nuclear Plant (BFN)
Units 1, 2, and 3

Technical Specifications (TS) Change 478

Addition of Analytical Methodologies to Technical Specification 5.6.5.b for Browns Ferry 1, 2, & 3, and Revision of Technical Specification 2.1.1.2 for Browns Ferry Unit 2, in Support of ATRIUM-10 XM Fuel Use at Browns Ferry

Retyped Proposed Technical Specifications Pages

The following pages have been revised to reflect the proposed changes. These are the retyped pages relative to the markups found in Attachment 2.

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model, Siemens Power Corporation, February 1998.

5. ANF-89-98(P)(A) Revision 1 and Supplement 1, Generic Mechanical Design Criteria for BWR Fuel Designs, Advanced Nuclear Fuels Corporation, May 1995.

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

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

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

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

20. BAW-10247PA Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, AREVA NP, February 2008.

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

22. 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.

(continued)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

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

MCPR shall be ≥ 1.06 for two recirculation loop operation or ≥ 1.08 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.

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. XN-NF-80-19(P)(A) Volume 1, Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis.

7. XN-NF-80-19(P)(A) Volume 4, Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads.

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

9. XN-NF-80-19(P)(A) Volume 3, Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description.

10. XN-NF-84-105(P)(A) Volume 1, XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis.

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

12. ANF-913(P)(A) Volume 1, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses.

13. ANF-1358(P)(A), The Loss of Feedwater Heating Transient in Boiling Water Reactors.

14. EMF-2209(P)(A), SPCB Critical Power Correlation.

15. EMF-2245(P)(A), Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel.

16. EMF-2361(P)(A), EXEM BWR-2000 ECCS Evaluation Model.

17. EMF-2292(P)(A), ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients.

(continued)

5.6 Reporting Requirements (continued)

CORE OPERATING LIMITS REPORT (COLR) (continued)

18. BAW-10247PA Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, AREVA NP, February 2008.

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

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

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. XN-NF-80-19(P)(A) Volume 1, Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis.

7. XN-NF-80-19(P)(A) Volume 4, Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads.

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

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10. XN-NF-84-105(P)(A) Volume 1, XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis.

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

12. ANF-913(P)(A) Volume 1, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses.

13. ANF-1358(P)(A), The Loss of Feedwater Heating Transient in Boiling Water Reactors.

14. EMF-2209(P)(A), SPCB Critical Power Correlation.

15. EMF-2245(P)(A), Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel.

16. EMF-2361(P)(A), EXEM BWR-2000 ECCS Evaluation Model.

17. EMF-2292(P)(A), ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

18. BAW-10247PA Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, AREVA NP, February 2008.

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

20. 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.

(continued)

ATTACHMENT 4

Browns Ferry Nuclear Plant (BFN)
Units 1, 2, and 3

Technical Specifications (TS) Change 478

Addition of Analytical Methodologies to Technical Specification 5.6.5.b for Browns Ferry 1, 2, & 3, and Revision of Technical Specification 2.1.1.2 for Browns Ferry Unit 2, in Support of ATRIUM-10 XM Fuel Use at Browns Ferry

Proposed Technical Specification Bases Changes (Mark-up)

The following pages have been revised to reflect the proposed changes. On the affected pages a line has been drawn through the deleted text and new or revised text is shaded.

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.

~~The SPCB critical power correlation is used for AREVA fuel and is valid at pressures ≥ 700 psia and bundle mass fluxes ≥ 0.1 x~~ ~~10^6 lb_m/hr-ft^2 (≥12,000 lb_m/hr, i.e., ≥ 10% core flow, on a per bundle basis). For thermal margin monitoring at 25% power and higher, the hot channel flow rate will be >28,000 lb_m/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 SPCB critical power correlation. For operation at low pressures or low flows, another basis is used, as follows:~~

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

(continued)

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)

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.

~~Thus operation up to 25% of rated power with normal natural circulation available is conservatively acceptable even if reactor pressure is equal to or below the lower pressure limit of the SPCB correlation. If reactor power is significantly less than 25% of rated (e.g., below 10% of rated), the core flow and the channel flow supported by the available driving head may be less than 28,000 lb_m/hr (along the lower portion of the natural circulation flow characteristic on the P/F map). However, the critical power that can be supported by the core and hot channel flow with normal natural circulation paths available remains well~~

(continued)

~~above the actual power conditions. The inherent characteristics of BWR natural circulation make power and core flow follow the natural circulation line as long as normal water level is maintained.~~

~~Thus, operation with core thermal power below 25% of rated without thermal margin surveillance is conservatively acceptable even for reactor operations at natural circulation. Adequate fuel thermal margins are also maintained without further surveillance for the low power conditions that would be present if core natural circulation is below the lower flow limit of the SPCB correlation.~~

(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. ~~SPCB critical power correlation.~~ References 2, 3, 4, 5, and 64 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~~5~~). 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.~~ANF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors," (as identified in the COLR).~~
7. 10 CFR 50.67.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the fuel design limits identified in Reference 1 are not exceeded during abnormal operational transients and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1, 2, and 11. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), abnormal operational transients, and normal operation that determine the APLHGR limits are presented in ~~References 1, 2, 3, 4, and 7 for GE fuel;~~ -References 11, 12, 13, 14, 15, and 16~~5~~ for -AREVA fuel.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~GE Fuel~~

~~LOCA analyses are performed to ensure APLHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis code is provided in Reference 5. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~For single recirculation loop operation, an APLHGR multiplier is applied to the APLHGR limit (Ref. 5 and Ref. 10). The multiplier is documented in the COLR. This multiplier is due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe heatup during a LOCA.~~

AREVA Fuel

For AREVA fuel, the APLHGR limits are developed as a function of exposed and, along wit the LHGR limits, ensure adherence to fuel design limits during abnormal operational transients. No power- or flow-dependent corrections are applied to the APLHGR (referred to as the maximum APLHGR or MAPLHGR). AREVA APLHGR limits are intended to be bound by the LHGR limits.

The calculational procedure used to establish the AREVA fuel MAPLHGR limits is based on LOCA analyses as defined in 10 CFR 50.46, Appendix K. MAPLHGR limits are created to assure that the peak cladding temperature of AREVA fuel following a postulated design basis LOCA will not exceed the PCT and maximum oxidation limits specified in 10 CFR 50.46, Appendix K. The calculational models and methodology are described in References 11 and 12.

The AREVA fuel MAPLHGR limits for two-loop operation are specified in the COLR. For single-loop operation, a MAPLHGR multiplier is applied to the MAPLHGR limit (Ref. 11). The multiplier is documented in the COLR.

The APLHGR satisfies Criterion 2 of the NRC Policy Statement (Ref. 6).

(continued)

BASES

REFERENCES (continued)

11. EMF-2361(P)(A), Rev. 0, "EXEM BWR-2000 ECCS Evaluation Model, "Framatome ANP Inc., as supplemented by the site-specific approval in NRC safety evaluation dated April 27, 2012.

12. EMF-2292(P)(A), "ATRIUM^{TM}-10: Appendix K Spray Heat Transfer Coefficients," (as identified in the COLR).

13. XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," (as identified in the COLR).

14. XN-NF-80-19(P)(A), Volume 1, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," (as identified in the COLR).

15. XN-NF-80-19(P)(A), Volume 4, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," (as identified in the COLR).

16. BAW-10247PA Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA NP, February 2008.

BASES

REFERENCES (continued)

8. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Units 1, 2, and 3," April 1995.

9. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.

10. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.

11. ANP-10307PA Revision 0, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," AREVA NP, June 2011.~~ANF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors," (as identified in the COLR).~~

12. EMF-2158(P)(A), "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4?MICROBURN-B2," (as identified in the COLR).

13. XN-NF-80-19(P)(A) Volume 3, "Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description," (as identified in the COLR).

14. ANF-913(P)(A) Volume 1, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," (as identified in the COLR).

15. XN-NF-84-105(P)(A) Volume 1, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," (as identified in the COLR).

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.

~~The SPCB critical power correlation is used for AREVA fuel and is valid at pressures ≥ 700 psia and bundle mass fluxes ≥ 0.1 x~~ ~~10^6 $lb_m/hr\text{-}ft^2$ (≥12,000 lb_m/hr, i.e., ≥ 10% core flow, on a per bundle basis). For thermal margin monitoring at 25% power and higher, the hot channel flow rate will be >28,000 lb_m/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 SPCB critical power correlation. For operation at low pressures or low flows, another basis is used, as follows:~~

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

(continued)

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.

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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.

~~Thus operation up to 25% of rated power with normal natural circulation available is conservatively acceptable even if reactor pressure is equal to or below the lower pressure limit of the SPCB correlation. If reactor power is significantly less than 25% of rated (e.g., below 10% of rated), the core flow and the channel flow supported by the available driving head may be less than 28,000 lb_m/hr (along the lower portion of the natural circulation flow characteristic on the P/F map). However, the critical power that can be supported by the core and hot channel flow with normal natural circulation paths available remains well~~

(continued)

~~above the actual power conditions. The inherent characteristics of BWR natural circulation make power and core flow follow the natural circulation line as long as normal water level is maintained.~~

~~Thus, operation with core thermal power below 25% of rated without thermal margin surveillance is conservatively acceptable even for reactor operations at natural circulation. Adequate fuel thermal margins are also maintained without further surveillance for the low power conditions that would be present if core natural circulation is below the lower flow limit of the SPCB correlation.~~

(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~~SPCB critical power correlation. References 2, 3, and 4~~ 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~~5~~). 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.~~ANF-524(P)(A), ANF Critical Power Methodology for Boiling Water Reactors, (as identified in the COLR).~~
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~~10 CFR 50.67~~.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the fuel design limits are not exceeded during abnormal operational transients and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 2 and 11. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), abnormal operational transients, and normal operation that determine the APLHGR limits are presented in References 11, 12, 13, 14, 15, and 16~~5~~ for the AREVA fuel.

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BASES

REFERENCES (continued)

13. XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," (as identified in the COLR).
14. XN-NF-80-19(P)(A) Volume 1, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," (as identified in the COLR).
15. XN-NF-80-19(P)(A) Volume 4, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," (as identified in the COLR).
16. BAW-10247PA Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, AREVA NP, February 2008.

BASES (continued)

REFERENCES

1. NUREG-0562, "Fuel Rod Failure As a Consequence of Departure from Nucleate Boiling or Dryout," June 1979.

2. (Deleted)

3. FSAR, Chapter 3.

4. FSAR, Chapter 14.

5. FSAR, Appendix N.

6. (Deleted)

7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

8. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Units 1, 2, and 3," April 1995.

9. (Deleted)

10. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.

11. ANP-10307PA Revision 0, AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, AREVA NP, June 2011.~~ANF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors," (as identified in the COLR).~~

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

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BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity

Critical power correlations are valid over a wide range of ~~The SPCB critical power correlation is used for AREVA fuel and is~~ 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:~~valid at pressures ≥ 700 psia and bundle mass fluxes ≥ 0.1 x 10^6 lb_m/hr-ft^2 (≥12,000 lb_m/hr, i.e., ≥10% core flow, on a per bundle basis). For thermal margin monitoring at 25% power and higher, the hot channel flow rate will be >28,000 lb_m/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 SPCB critical power correlation. For operation at low pressures or low flows, 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

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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.

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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.

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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.

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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~~5~~). 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.~~ANF-524(P)(A), ANF Critical Power Methodology for Boiling Water Reactors, (as identified in the COLR).~~
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.~~10 CFR 50.67.~~

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND	The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the fuel design limits are not exceeded during abnormal operational transients and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.
APPLICABLE SAFETY ANALYSES	The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 2 and 11. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), abnormal operational transients, and normal operation that determine the APLHGR limits are presented in References 11, 12, 13, 14, ~~and~~ 15, and 16 for the AREVA fuel.

(continued)

BASES

REFERENCES (continued)

13. XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal- "Mechanical Response Evaluation Model," (as identified in the COLR).
14. XN-NF-80-19(P)(A) Volume 1, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," (as identified in the COLR).
15. XN-NF-80-19(P)(A) Volume 4, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," (as identified in the COLR).
16. BAW-10247PA Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA NP, February 2008.

BASES (continued)

REFERENCES

1. NUREG-0562, "Fuel Rod Failure As a Consequence of Departure from Nucleate Boiling or Dryout," June 1979.

2. (Deleted)

3. FSAR, Chapter 3.

4. FSAR, Chapter 14.

5. FSAR, Appendix N.

6. (Deleted)

7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

8. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Units 1, 2, and 3," April 1995.

9. (Deleted)

10. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.

11. ANP-10307PA Revision 0, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," AREVA NP, June 2011.~~ANF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors," (as identified in the COLR).~~

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

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ATTACHMENT 5

Browns Ferry Nuclear Plant (BFN)
Units 1, 2, and 3

Technical Specifications (TS) Change 478

Addition of Analytical Methodologies to Technical Specification 5.6.5.b for Browns Ferry 1, 2, & 3, and Revision of Technical Specification 2.1.1.2 for Browns Ferry Unit 2, in Support of ATRIUM-10 XM Fuel Use at Browns Ferry

Retyped Proposed Technical Specification Bases Pages

The following pages have been revised to reflect the proposed changes. These are the retyped pages relative to the markups found in Attachment 4.

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.

(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.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the fuel design limits identified in Reference 1 are not exceeded during abnormal operational transients and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1, 2, and 11. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), abnormal operational transients, and normal operation that determine the APLHGR limits are presented in References 11, 12, 13, 14, 15, and 16 for AREVA fuel.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

AREVA Fuel

For AREVA fuel, the APLHGR limits are developed as a function of exposed and, along wit the LHGR limits, ensure adherence to fuel design limits during abnormal operational transients. No power- or flow-dependent corrections are applied to the APLHGR (referred to as the maximum APLHGR or MAPLHGR). AREVA APLHGR limits are intended to be bound by the LHGR limits.

The calculational procedure used to establish the AREVA fuel MAPLHGR limits is based on LOCA analyses as defined in 10 CFR 50.46, Appendix K. MAPLHGR limits are created to assure that the peak cladding temperature of AREVA fuel following a postulated design basis LOCA will not exceed the PCT and maximum oxidation limits specified in 10 CFR 50.46, Appendix K. The calculational models and methodology are described in References 11 and 12.

The AREVA fuel MAPLHGR limits for two-loop operation are specified in the COLR. For single-loop operation, a MAPLHGR multiplier is applied to the MAPLHGR limit (Ref. 11). The multiplier is documented in the COLR.

The APLHGR satisfies Criterion 2 of the NRC Policy Statement (Ref. 6).

(continued)

BASES

REFERENCES (continued)

11. EMF-2361(P)(A), Rev. 0, "EXEM BWR-2000 ECCS Evaluation Model, "Framatome ANP Inc., as supplemented by the site-specific approval in NRC safety evaluation dated April 27, 2012.

12. EMF-2292(P)(A), "ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients," (as identified in the COLR).

13. XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," (as identified in the COLR).

14. XN-NF-80-19(P)(A), Volume 1, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," (as identified in the COLR).

15. XN-NF-80-19(P)(A), Volume 4, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," (as identified in the COLR).

16. BAW-10247PA Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA NP, February 2008.

BASES

REFERENCES (continued)

8. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Units 1, 2, and 3," April 1995.

9. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.

10. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.

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

12. EMF-2158(P)(A), "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4?MICROBURN-B2," (as identified in the COLR).

13. XN-NF-80-19(P)(A) Volume 3, "Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description," (as identified in the COLR).

14. ANF-913(P)(A) Volume 1, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," (as identified in the COLR).

15. XN-NF-84-105(P)(A) Volume 1, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," (as identified in the COLR).

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 $\leq$ 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.

(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.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the fuel design limits are not exceeded during abnormal operational transients and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 2 and 11. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), abnormal operational transients, and normal operation that determine the APLHGR limits are presented in References 11, 12, 13, 14, 15, and 16 for the AREVA fuel.

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BASES

REFERENCES (continued)

13. XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," (as identified in the COLR).

14. XN-NF-80-19(P)(A) Volume 1, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," (as identified in the COLR).

15. XN-NF-80-19(P)(A) Volume 4, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," (as identified in the COLR).

16. BAW-10247PA Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, AREVA NP, February 2008.

BASES (continued)

REFERENCES

1. NUREG-0562, "Fuel Rod Failure As a Consequence of Departure from Nucleate Boiling or Dryout," June 1979.
2. (Deleted)
3. FSAR, Chapter 3.
4. FSAR, Chapter 14.
5. FSAR, Appendix N.
6. (Deleted)
7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
8. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Units 1, 2, and 3," April 1995.
9. (Deleted)
10. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.
11. ANP-10307PA Revision 0, AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, AREVA NP, June 2011.
12. EMF-2158(P)(A), "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," (as identified in the COLR).

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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.

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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.

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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.

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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.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND	The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the fuel design limits are not exceeded during abnormal operational transients and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.
APPLICABLE SAFETY ANALYSES	The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 2 and 11. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), abnormal operational transients, and normal operation that determine the APLHGR limits are presented in References 11, 12, 13, 14, 15, and 16 for the AREVA fuel.

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BASES

REFERENCES (continued)

13. XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-"Mechanical Response Evaluation Model," (as identified in the COLR).

14. XN-NF-80-19(P)(A) Volume 1, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," (as identified in the COLR).

15. XN-NF-80-19(P)(A) Volume 4, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," (as identified in the COLR).

16. BAW-10247PA Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA NP, February 2008.

BASES (continued)

REFERENCES

1. NUREG-0562, "Fuel Rod Failure As a Consequence of Departure from Nucleate Boiling or Dryout," June 1979.
2. (Deleted)
3. FSAR, Chapter 3.
4. FSAR, Chapter 14.
5. FSAR, Appendix N.
6. (Deleted)
7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
8. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Units 1, 2, and 3," April 1995.
9. (Deleted)
10. NEDO-24236, "Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation," May 1981.
11. ANP-10307PA Revision 0, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," AREVA NP, June 2011.
12. EMF-2158(P)(A), "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," (as identified in the COLR).

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