TVA

Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402-2801

November 17, 2009

10 CFR 50.90

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

Browns Ferry Nuclear Plant, Unit 1
Facility Operating License No. DPR-33
NRC Docket No. 50-259

Subject: **Technical Specification Change TS-467-S - Utilization of AREVA Fuel and Associated Analysis Methodologies - Non EPU Supplement**

References:
1. Letter from NRC to TVA, "Summary of January 28, 2009, Meeting with the Tennessee Valley Authority Regarding Proposed Fuel Transition Amendment (TAC No. ME0438)," dated March 23, 2009
2. Letter from NRC to TVA, "Summary of March 16, 2009, Meeting with the Tennessee Valley Authority Regarding Proposed Fuel Transition Amendment (TAC No. ME0438)," dated June 3, 2009
3. Letter from TVA to NRC, "Technical Specification Change TS-467 - Utilization of AREVA Fuel and Associated Analysis Methodologies," October 23, 2009
4. Letter from TVA to NRC "Browns Ferry Nuclear Plant (BFN) Unit 1, Proposed Technical Specifications (TS) Change - 431, Request for License Amendment - Extended Power Uprate (EPU) Operation," dated June 28, 2004

By letter dated October 23, 2009 (Reference 3), the Tennessee Valley Authority (TVA) submitted a request for amendments to the Technical Specifications (TS) for Browns Ferry Nuclear Plant, Unit 1. The amendment request (TS-467) proposed to add the AREVA NP analysis methodologies to the list of approved methods to be used in determining the core operating limits in the Core Operating Limits Report (COLR). Additional Technical Specification changes are also requested to reflect the AREVA NP specific methods for monitoring and enforcing of the thermal limits. As indicated in the Reference 3 letter, TS-467 is based on Extended Power Uprate (EPU) power conditions, and provides the necessary core methods and fuels

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analyses to support the NRC review of TS-431 (Reference 4) using AREVA NP analysis methodologies at EPU.

Given the delays in EPU approval for Browns Ferry Nuclear Plant, Unit 1, TVA is supplementing the original submitted TS-467 with additional information to support operation of Unit 1 for Cycle 9 at the current licensed power (i.e., 105% Original Licensed Thermal Power (OLTP)). Accordingly, TVA requests that this supplement to TS-467 be approved for Unit 1 Cycle 9 based on a licensed power level of 105% OLTP.

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 NP analysis methodologies, were provided in Attachments 6 through 23 of the Reference 3 submittal. These attachments also provided the information requested during meetings, summarized in References 1 and 2, between TVA and NRC representatives. The information in the Reference 3 submittal is based on Extended Power Uprate (EPU) conditions. This supplement provides an update to the enclosure and some of the original attachments provided in TS-467 (Reference 3) to reflect Unit 1 current licensed power conditions. The submittal of this additional supplemental information was previously agreed to between NRC and TVA representatives during a teleconference on June 2, 2009.

For the Unit 1 current licensed power conditions, Attachments 1, 4, 5, 14, 15, 16, 17, 18, 19, 20, and 21 of the Reference 3 submittal are applicable. The Reference 3 submittal included proposed Technical Specification changes (Attachments 2 and 3) to TS 2.1.1.2, "Minimum Critical Power Ratio Safety Limit," to reflect the results of the cycle specific analyses of the Safety Limit Minimum Critical Power Ratio (SLMCPR) at EPU power. For non-EPU conditions (i.e., 105% OLTP), the current Unit 1 TS 2.1.1.2 SLMCPR values are supported by the results of the cycle specific analyses. Therefore, changes to TS 2.1.1.2 are not required for non EPU conditions and Attachments 2 and 3 of the Reference 3 submittal are revised in this supplement to remove the TS 2.1.1.2 related changes. While the documents provided in Attachments 14 and 15 of the Reference 3 submittal are EPU based, the results are bounding for non EPU operation, and are applicable for the Unit 1 current licensed power (i.e., 105% OLTP). The information in Attachments 16 through 21 of the Reference 3 submittal is unchanged in this supplement. The documents provided in Attachments 6, 7, 8, 9, 10, 11, 12 and 13 of the Reference 3 submittal have been updated in this supplement to be based on the Unit 1 current licensed power (i.e., 105% OLTP). Attachments 22 and 23 of the Reference 3 submittal provide the summary of the SLMCPR analyses for EPU conditions and are not applicable to non EPU conditions. Therefore, Attachments 22 and 23 of the Reference 3 letter are not applicable to this supplement. The information for the non EPU SLMCPR analysis is contained in Attachment 12 (105% OLTP) and Attachment 13 (105% OLTP), which are included in this supplement.

The above information is summarized as follows.

| TS-467 (Reference 3) Attachment | Title | TS-467 Supplement Status |
|---|---|---|
| 1 | List of Regulatory Commitments | Applicable |
| 2 | Proposed Technical Specifications Changes (Mark-up) | Revised in Attachment 2 (105% OLTP) |
| 3 | Retyped Proposed Technical Specifications Pages | Revised in Attachment 3 (105% OLTP) |
| 4 | Proposed Technical Specification Bases Changes (Mark-up) | Applicable |
| 5 | Retyped Proposed Technical Specification Bases Pages | Applicable |
| 6 | Mechanical Design Report (*proprietary*) ANP-2833(P), Revision 0, Mechanical Design Report for Browns Ferry Unit 1 Reload BFE1-9 ATRIUM-10 Fuel Assemblies, AREVA NP Inc., September 2009. | Revised in Attachment 6 (105% OLTP) |
| 7 | Mechanical Design Report (*non-proprietary*) ANP-2833(NP), Revision 0, Mechanical Design Report for Browns Ferry Unit 1 Reload BFE1-9 ATRIUM-10 Fuel Assemblies, AREVA NP Inc., September 2009. | Revised in Attachment 7 (105% OLTP) |
| 8 | Thermal Hydraulic Design Report (*proprietary*) ANP-2807(P), Revision 0, Browns Ferry Unit 1 Thermal-Hydraulic Design Report for ATRIUM-10 Fuel Assemblies, AREVA NP Inc., June 2009. | Revised in Attachment 8 (105% OLTP) |
| 9 | Thermal Hydraulic Design Report (*non-proprietary*) ANP-2807(NP), Revision 0, Browns Ferry Unit 1 Thermal-Hydraulic Design Report for ATRIUM-10 Fuel Assemblies, AREVA NP Inc., June 2009. | Revised in Attachment 9 (105% OLTP) |
| 10 | Fuel Cycle Design Report (*proprietary*) ANP-2850(P), Revision 0, Browns Ferry Unit 1 Fuel Cycle Design, AREVA NP Inc., July 2009. | Revised in Attachment 10 (105% OLTP) |
| 11 | Fuel Cycle Design Report (*non-proprietary*) ANP-2850(NP), Revision 0, Browns Ferry Unit 1 Fuel Cycle Design, AREVA NP Inc., August 2009. | Revised in Attachment 11 (105% OLTP) |
| 12 | Reload Safety Analysis Report (*proprietary*) ANP-2864(P), Revision 2, Browns Ferry Unit 1 Cycle Reload Safety Analysis, AREVA NP Inc., October 2009. | Revised in Attachment 12 (105% OLTP) |
| 13 | Reload Safety Analysis Report (*non-proprietary*) ANP-2864(NP), Revision 2, Browns Ferry Unit 1 Cycle Reload Safety Analysis, AREVA NP Inc., October 2009. | Revised in Attachment 13 (105% OLTP) |
| 14 | LOCA Break Spectrum Analysis Report (*proprietary*) EMF-2950(P), Revision 2, Browns Ferry Units 1, 2, and 3 Extended Power Uprate LOCA Break Spectrum Analysis, AREVA NP Inc., August 2009. | Applicable |

| TS-467 (Reference 3) Attachment | Title | TS-467 Supplement Status |
|---|---|---|
| 15 | LOCA Break Spectrum Analysis Report (*non-proprietary*)<br>EMF-2950(NP), Revision 0, Browns Ferry Units 1, 2, and 3 Extended Power Uprate LOCA Break Spectrum Analysis, AREVA NP Inc., August 2009. | Applicable |
| 16 | Response to NRC Comments Regarding Browns Ferry Unit 1 Proposed Fuel Transition Amendment (*non-proprietary*)<br>51-9121503-002, Response to NRC Comments Regarding Browns Ferry Unit 1 Proposed Fuel Transition Amendment, AREVA NP Inc., October 2009. | Applicable |
| 17 | Boiling Water Reactor Licensing Methodology Compendium (*non-proprietary*)<br>ANP-2637, Revision 2, Boiling Water Reactor Licensing Methodology Compendium, AREVA NP Inc., December 2007. | Applicable |
| 18 | Applicability of AREVA NP BWR Methods to Extended Power Uprate Conditions (*proprietary*)<br>ANP-2638(P), Revision 2, Applicability of AREVA NP BWR Methods to Extended Power Uprate Conditions, AREVA NP Inc., October 2009. | Applicable |
| 19 | Applicability of AREVA NP BWR Methods to Extended Power Uprate Conditions (*non-proprietary*)<br>ANP-2638(NP), Revision 2, Applicability of AREVA NP BWR Methods to Extended Power Uprate Conditions, AREVA NP Inc., October 2009. | Applicable |
| 20 | Part 1: Previous NRC Requests for Additional Information Matrix and Text<br>Part 2: Browns Ferry Unit 1 – Summary of Response to Requests for Additional Information (*proprietary*)<br>ANP-2860(P), Revision 2, Browns Ferry Unit 1 – Summary of Responses to Requests for Additional Information, AREVA NP Inc., October 2009. | Applicable |
| 21 | Part 1: Previous NRC Requests for Additional Information Matrix and Text<br>Part 2: Browns Ferry Unit 1 – Summary of Response to Requests for Additional Information (*non-proprietary*)<br>ANP-2860(NP), Revision 2, Browns Ferry Unit 1 – Summary of Responses to Requests for Additional Information, AREVA NP Inc., October 2009. | Applicable |
| 22 | Safety Limit Minimum Critical Power Ratio (*proprietary*)<br>51-9119738-000, Browns Ferry Unit 1 Cycle 9 MCPR Safety Limit Analysis (120% OLTP), AREVA NP Inc., September 2009. | Not Applicable SLMCPR for non EPU conditions addressed in Attachment 12 (105% OLTP) |

| TS-467 (Reference 3) Attachment | Title | TS-467 Supplement Status |
|---|---|---|
| 23 | Safety Limit Minimum Critical Power Ratio (*non-proprietary*)<br>51-9121246-000, Browns Ferry Unit 1 Cycle 9 MCPR Safety Limit Analysis (120% OLTP), AREVA NP Inc., September 2009. | Not Applicable SLMCPR for non EPU conditions addressed in Attachment 13 (105% OLTP) |
| 24 | Affidavits | Applicable Additional Affidavits provided in Attachment 24 (105% OLTP) |

Attachments 6, 8, 10, 12, 14, 18, 20 and 22 of the Reference 3 letter 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), in the Reference 3 submittal, TVA requested that such information be withheld from public disclosure. Attachment 24 of the Reference 3 submittal provided the affidavits supporting this request. These affidavits remain applicable.

In addition, Attachments 6 (105% OLTP), 8 (105% OLTP), 10 (105% OLTP), and 12 (105% OLTP) of this supplement contain information that AREVA NP considers to be proprietary in nature and subsequently, pursuant to 10 CFR 2.390, paragraph (a)(4), it is requested that such information be withheld from public disclosure. Attachment 24 (105% OLTP) of this supplement provides the affidavits supporting this request. Attachments 7 (105% OLTP), 9 (105% OLTP), 11 (105% OLTP), and 13 (105% OLTP) of this supplement contain the redacted versions of the proprietary attachments with the proprietary material removed, which are suitable for public disclosure.

TVA has determined that the supplemental information provided by this letter does not affect the no significant hazards considerations associated with the proposed TS changes. The proposed TS changes still 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 October 22, 2010, and that the implementation of the revised TS be made prior to the startup of Unit 1 for Cycle 9.

There are no regulatory commitments in this submittal as reflected in Attachment 1.

Please direct any questions concerning this matter to Dan Green at (423) 751-8423.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on the 17th day of November 2009.

Respectfully,

R. M. Krich
Vice President
Nuclear Licensing

Enclosure: Revision of Technical Specifications to Allow Utilization of AREVA Fuel and Associated Analytical Methodologies – Non EPU Supplement

Attachments:

| | |
|---|---|
| 2 (105% OLTP) | Proposed Technical Specifications Changes (Mark-up) |
| 3 (105% OLTP) | Retyped Proposed Technical Specifications Pages |
| 6 (105% OLTP) | Mechanical Design Report (*proprietary*) ANP-2877(P), Revision 0, Mechanical Design Report for Browns Ferry Unit 1 Reload BFE1-9 ATRIUM-10 Fuel Assemblies (105% OLTP), AREVA NP Inc., November 2009. |
| 7 (105% OLTP) | Mechanical Design Report (*non-proprietary*) ANP-2877(NP), Revision 0, Mechanical Design Report for Browns Ferry Unit 1 Reload BFE1-9 ATRIUM-10 Fuel Assemblies (105% OLTP), AREVA NP Inc., November 2009. |
| 8 (105% OLTP) | Thermal Hydraulic Design Report (*proprietary*) ANP-2821(P), Revision 0, Browns Ferry Unit 1 Thermal-Hydraulic Design Report for ATRIUM-10 Fuel Assemblies (105% OLTP), AREVA NP Inc., June 2009. |
| 9 (105% OLTP) | Thermal Hydraulic Design Report (*non-proprietary*) ANP-2821(NP), Revision 0, Browns Ferry Unit 1 Thermal-Hydraulic Design Report for ATRIUM-10 Fuel Assemblies (105% OLTP), AREVA NP Inc., June 2009. |
| 10 (105% OLTP) | Fuel Cycle Design Report (*proprietary*) ANP-2859(P), Revision 0, Browns Ferry Unit 1 Cycle 9 Fuel Cycle Design (105% OLTP), AREVA NP Inc., September 2009. |
| 11 (105% OLTP) | Fuel Cycle Design Report (*non-proprietary*) ANP-2859(NP), Revision 0, Browns Ferry Unit 1 Cycle 9 Fuel Cycle Design (105% OLTP), AREVA NP Inc., September 2009. |

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|---|---|
| 12 (105% OLTP) | Reload Safety Analysis Report (*proprietary*) ANP-2863(P), Revision 0, Browns Ferry Unit 1 Cycle 9 Reload Safety Analysis for 105% OLTP, AREVA NP Inc., November 2009. |
| 13 (105% OLTP) | Reload Safety Analysis Report (*non-proprietary*) ANP-2863(NP), Revision 0, Browns Ferry Unit 1 Cycle 9 Reload Safety Analysis for 105% OLTP, AREVA NP Inc., November 2009. |
| 24 (105% OLTP) | Affidavits |

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)**
**Unit 1**

**Technical Specifications (TS) Change 467-S**

**Revision of Technical Specifications to Allow Utilization of AREVA Fuel and Associated Analytical Methodologies**

Non EPU Supplement

## 1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Operating License DPR-33 for BFN Unit 1. The proposed changes would revise the Operating License to allow the use of AREVA fuel and analytical methodologies for BFN Unit 1. Unit 1 will transition from using Global Nuclear Fuel's (GNF) GE14 design, to using the AREVA ATRIUM-10 fuel design commencing with the reload batch delivered in the fall of 2010. This supplement updates the initial TS-467 submittal (Reference 14) to provide additional information based on non EPU power conditions (i.e., 105% Original Licensed Power Level (OLTP)).

## 2.0 DETAILED DESCRIPTION

The Tennessee Valley Authority (TVA) intends to begin utilizing the ATRIUM-10 design in BFN Unit 1 Cycle 9. The first reload of ATRIUM-10 targeted for insertion into the core is the fall 2010 outage. The ATRIUM-10 product is an industry proven fuel design in use at BFN 2 and BFN 3 since 2005 and 2004, respectively. The initial Unit 1 reload, and at least one follow on reload, will 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 ATRIUM-10 fuel in Unit 1 with standard commercial grade uranium in future reloads.

In order to extend the use of this fuel design to BFN Unit 1, several changes to the Technical Specifications (TS) are required. TS 5.6.5.b address the analytical methods which may be used to determine input to the Core Operating Limits Report (COLR). Currently, the BFN Unit 1 specification only includes GNF analytical methods. Unit 1 TS 5.6.5.b will be revised to add appropriate NRC approved AREVA analytical methodologies.

Also, TS 3.2.3 (Linear Heat Generation Rate (LHGR)) requires an administrative correction. Word processing of a previous change caused the header to incorrectly state "APRM Gain and Setpoints," (instead of LHGR). The header and section number are corrected. The change is editorial in nature and has no impact on public health and safety and no impact on the environment.

In addition, two other TS changes will be made to reflect the manner by which AREVA methodologies monitor and enforce thermal limits. The affected TS sections are 3.3.4.1 (End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation) and 3.7.5 (Main Turbine Bypass System); both are modified to require a linear heat generation rate limit adjustment when operating with EOC-RPT out of service -and operating with turbine bypass out of service, respectively.

The submittal also addresses the required changes to the Technical Specification Bases. Changes are related to adding information pertaining to AREVA analytical methodologies (including Reference documents) and information related to AREVA specific monitoring and enforcement of fuel thermal limits.

The previous AREVA fuel transition submittal for BFN (Reference 1) addressed Unit 1 to the extent of providing a description of the AREVA fuel (TS 4.2.1, Reactor Core - Fuel Assemblies), and to modify the fuel storage criticality requirement to a k-effective basis (TS 4.3.1, Fuel Storage - Criticality). Unit 1 was included in this prior change to allow for the possibility of storing AREVA fuel bundles in the BFN Unit 1 spent fuel pool. Consequently, these two TSs do not require alteration, and are not included in the current change request.

In a meeting with the NRC staff on January 28, 2009, the overall approach for the BFN Unit 1 fuel transition submittal was discussed. In addition to providing guidance on submittal timing, the NRC provided a list of eleven technical items to be addressed in the submittal, per Reference 2. A follow-up meeting was held on March 16, 2009 in which the specific contents of the transition submittal were agreed upon per Reference 3. In addition to the eleven items mentioned above, the NRC requested certain AREVA reload documents pertaining to the design and licensing analyses of the transition cycle, as well as selected generic reports related to methodologies, be included in the submittal. The NRC also provided a specific list of prior Requests for Additional Information (RAIs), which should be answered for Unit 1 (addressing the co-resident GNF fuel impacts as appropriate). Responses to prior RAIs, are addressed in Attachments 12 and 12 (105% OLTP), 18, and 20. Attachment 16 provides responses to technical items identified in Reference 2, along with information on BLEU material.

The specific information requested by the NRC is included in the following attachments.

| Attachment | Description | Location (TS-467 (Reference 14) or this Supplement (TS-467-S)) |
|---|---|---|
| 1 | List of Regulatory Commitments | TS-467 |
| 2 (105% OLTP) | Proposed Technical Specifications Changes (Mark-up) | TS-467-S |
| 3 (105% OLTP) | Retyped Proposed Technical Specifications Pages | TS-467-S |
| 4 | Proposed Technical Specification Bases Changes (Mark-up) | TS-467 |
| 5 | Retyped Proposed Technical Specification Bases Pages | TS-467 |
| 6 (105% OLTP) | Mechanical Design Report *(proprietary)*<br>ANP-2877(P), Revision 0, Mechanical Design Report for Browns Ferry Unit 1 Reload BFE1-9 ATRIUM-10 Fuel Assemblies (105% OLTP), AREVA NP Inc., November 2009. | TS-467-S |

| Attachment | Description | Location (TS-467 (Reference 14) or this Supplement (TS-467-S)) |
|---|---|---|
| 7 (105% OLTP) | Mechanical Design Report *(non-proprietary)* ANP-2877(NP), Revision 0, Mechanical Design Report for Browns Ferry Unit 1 Reload BFE1-9 ATRIUM-10 Fuel Assemblies (105% OLTP), AREVA NP Inc., November 2009. | TS-467-S |
| 8 (105% OLTP) | Thermal Hydraulic Design Report *(proprietary)* ANP-2821(P), Revision 0, Browns Ferry Unit 1 Thermal-Hydraulic Design Report for ATRIUM-10 Fuel Assemblies (105% OLTP), AREVA NP Inc., June 2009. | TS-467-S |
| 9 (105% OLTP) | Thermal Hydraulic Design Report *(non-proprietary)* ANP-2821(NP), Revision 0, Browns Ferry Unit 1 Thermal-Hydraulic Design Report for ATRIUM-10 Fuel Assemblies (105% OLTP), AREVA NP Inc.; June 2009. | TS-467-S |
| 10 (105% OLTP) | Fuel Cycle Design Report *(proprietary)* ANP-2859(P), Revision 0, Browns Ferry Unit 1 Cycle 9 Fuel Cycle Design (105% OLTP), AREVA NP Inc., September 2009. | TS-467-S |
| 11 (105% OLTP) | Fuel Cycle Design Report *(non-proprietary)* ANP-2859(NP), Revision 0, Browns Ferry Unit 1 Cycle 9 Fuel Cycle Design (105% OLTP), AREVA NP Inc., September 2009. | TS-467-S |
| 12 (105% OLTP) | Reload Safety Analysis Report *(proprietary)* ANP-2863(P), Revision 0, Browns Ferry Unit 1 Cycle 9 Reload Safety Analysis for 105% OLTP, AREVA NP Inc., November 2009. | TS-467-S |
| 13 (105% OLTP) | Reload Safety Analysis Report *(non-proprietary)* ANP-2863(NP), Revision 0, Browns Ferry Unit 1 Cycle 9 Reload Safety Analysis for 105% OLTP, AREVA NP Inc., November 2009. | TS-467-S |
| 14 | LOCA Break Spectrum Analysis Report (*proprietary*) EMF-2950(P), Revision 2, Browns Ferry Units 1, 2, and 3 Extended Power Uprate LOCA Break Spectrum Analysis, AREVA NP Inc., August 2009. | TS-467 |
| 15 | LOCA Break Spectrum Analysis Report (*non-proprietary*) EMF-2950(NP), Revision 0, Browns Ferry Units 1, 2, and 3 Extended Power Uprate LOCA Break Spectrum Analysis, AREVA NP Inc., August 2009. | TS-467 |
| 16 | Response to NRC Comments Regarding Browns Ferry Unit 1 Proposed Fuel Transition Amendment (*non-proprietary*) 51-9121503-002, Response to NRC Comments Regarding Browns Ferry Unit 1 Proposed Fuel Transition Amendment, AREVA NP Inc., October 2009. | TS-467 |
| 17 | Boiling Water Reactor Licensing Methodology Compendium (*non-proprietary*) ANP-2637, Revision 2, Boiling Water Reactor Licensing Methodology Compendium, AREVA NP Inc., December 2007. | TS-467 |
| 18 | Applicability of AREVA NP BWR Methods to Extended Power Uprate Conditions (*proprietary*) ANP-2638(P), Revision 2, Applicability of AREVA NP BWR Methods to Extended Power Uprate Conditions, AREVA NP Inc., October 2009. | TS-467 |
| 19 | Applicability of AREVA NP BWR Methods to Extended Power Uprate Conditions (*non-proprietary*) ANP-2638(NP), Revision 2, Applicability of AREVA NP BWR Methods to Extended Power Uprate Conditions, AREVA NP Inc., October 2009. | TS-467 |
| 20 | Part 1: Previous NRC Requests for Additional Information Matrix and Text Part 2: Browns Ferry Unit 1 – Summary of Response to Requests for Additional Information (*proprietary*) | TS-467 |

| Attachment | Description | Location (TS-467 (Reference 14) or this Supplement (TS-467-S)) |
|---|---|---|
| | ANP-2860(P), Revision 2, Browns Ferry Unit 1 – Summary of Responses to Requests for Additional Information, AREVA NP Inc., October 2009. | |
| 21 | Part 1: Previous NRC Requests for Additional Information Matrix and Text<br>Part 2: Browns Ferry Unit 1 – Summary of Response to Requests for Additional Information (*non-proprietary*)<br>ANP-2860(NP), Revision 2, Browns Ferry Unit 1 – Summary of Responses to Requests for Additional Information, AREVA NP Inc., October 2009. | TS-467 |
| 22 | Not used. | Not Applicable<br>SLMCPR for non EPU conditions addressed in Attachment 12 (105% OLTP) |
| 23 | Not used. | Not Applicable<br>SLMCPR for non EPU conditions addressed in Attachment 13 (105% OLTP) |
| 24 and<br>24 (105% OLTP) | Affidavits | TS-467 and TS-467-S |

Attachment information in the original TS-467 submittal (Reference 14) is based on Extended Power Uprate (EPU, 120% OLTP operations), consistent with previous meetings between TVA and NRC representatives, documented in References 2 and 3. As part of the meeting documented in Reference 3, representatives also discussed the potential to provide supplemental information supporting a 105% OLTP core design option. At the time of the Reference 3 meeting, TVA proposed to have all submittal material to NRC by the end of December. TVA and AREVA took actions to determine if an earlier date could be supported. A subsequent proposal to NRC indicated TVA would provide the EPU portion of the submittal first, followed up shortly thereafter with additional material supporting a 105% OLTP core design option.

A related public meeting was held between TVA and NRC representatives on August 11, 2009 to discuss the status of the TVA's pending EPU submittal. During the meeting, representatives discussed the potential impact of delaying the EPU submittal approval, in the context of the BFN Unit 1 fuel transition. To address the possibility of a 105% OLTP based fuel transition, TVA stated that a 120% OLTP based submittal would be made initially, with a submittal supplement to be provided by the end of December 2009 supporting 105% OLTP. NRC concurred with TVA's position to provide supplemental information based on the current licensed thermal power (105% OLTP). This supplement provides the additional information supporting the Unit 1 current licensed power (105% OLTP).

Much of the information in the Reference 14 submittal applies to both 120% and 105% OLTP core designs. For the Unit 1 current licensed power conditions, Attachments 1, 4, 5, 14, 15, 16, 17, 18, 19, 20, and 21 of the Reference 14 submittal are applicable. The Reference 14 submittal included proposed Technical Specification changes (Attachments 2 and 3) to TS 2.1.1.2, "Minimum Critical Power Ratio Safety Limit," to reflect the results of the cycle specific analyses of the Safety Limit Minimum Critical Power Ratio (SLMCPR) at EPU power. For non-EPU conditions (i.e., 105% OLTP), the current Unit 1 TS 2.1.1.2 SLMCPR values are supported by the results of the cycle specific analyses. Therefore, changes to TS 2.1.1.2 are not required for non EPU conditions and Attachments 2 and 3 of the Reference 14 submittal are revised in this supplement to remove the TS 2.1.1.2 related changes. While the documents provided in Attachments 14 and 15 of the Reference 14 submittal are EPU based, the results are bounding for non EPU operation, and are applicable for the Unit 1 current licensed power (i.e., 105% OLTP). The information in Attachments 16 through 21 of the Reference 14 submittal is unchanged in this supplement. The documents provided in Attachments 6, 7, 8, 9, 10, 11, 12 and 13 of the Reference 14 submittal have been updated and reissued with new document numbers in this supplement to be based on the Unit 1 current licensed power (i.e., 105% OLTP). Attachments 22 and 23 of the Reference 14 submittal provide the summary of the SLMCPR analyses for EPU conditions and are not applicable to non EPU conditions. Therefore, Attachments 22 and 23 of the Reference 14 letter are not applicable to this supplement. The information for the non EPU SLMCPR analysis is contained in Attachment 12 (105% OLTP) and Attachment 13 (105% OLTP), which are included in this supplement.

All critical power results provided in the submittal (Attachments 8 (105% OLTP), 9 (105% OLTP), 10 (105% OLTP), 11 (105% OLTP), 12 (105% OLTP), and 13 (105% OLTP)) are based on corrected additive constants approved by NRC in the addendum discussed above. Upon issuance of a revision to EMF-2209(P)(A), TVA will provide a revision to ANP-2637(Attachment 17), referencing the new version. ANP-2637 (Attachment 17) identifies Reference 8 as an approved methodology report. Reference 8 is the approved version at the time of this submittal. There was an outstanding 10 CFR 21, "Reporting of Defects and Noncompliance," issue related to Reference 8. The issue had to do with reported ATRIUM-10 additive constants values having been found non-conservative. AREVA submitted the Reference 9 addendum to Reference 8, correcting ATRIUM-10 additive constants values. The NRC has approved the Reference 9 correction per Reference 13.

A second 10 CFR 21 issue was recently identified in Reference 10. The issue of operating limit error is related to the fact that LaSalle operates with Zinc levels well beyond the industry standard set by Electric Power Research Institute (EPRI) guidance in References 11 and 12. LaSalle measured unusually high liftoff levels, which were attributed to operating water chemistry with high levels of Zinc. All BFN units operate within the EPRI water chemistry guidance, and measured BFN liftoff levels remain consistent with AREVA methodology assumptions. Therefore, this particular 10 CFR 21 issue is not applicable to any BFN unit.

## 3.0 TECHNICAL EVALUATION

The fuel design to be introduced into Unit 1 in 2010 is the AREVA ATRIUM-10 product. This design utilizes a 10x10 array of fuel rods, with eighty-three full length fuel rods and eight partial length fuel rods. The partial length fuel rods are approximately two-thirds 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.

The ATRIUM-10 design does not utilize tie rods as the structural tie between the upper and lower tie plates. Instead, the 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 ATRIUM-10 design uses a total of eight fuel rod spacers to provide lateral support for the fuel rods and to enhance thermal hydraulic performance. The ATRIUM-10 design to be employed at Unit 1 utilizes a debris resistant lower tie plate to limit introduction of foreign material into the assembly from below.

The ATRIUM-10 design was developed using the thermal mechanical design bases and limits outlined in Reference 4. Compliance with Reference 4 ensures the fuel design meets the fuel system damage, fuel failure, and fuel coolability criteria identified in the Reference 5 Standard Review Plan. The NRC reviewed and approved (per Reference 6) the use of Reference 4 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 ATRIUM-10 design fully complies with the criteria of Reference 4, and therefore meets all of the required fuel licensing criteria in the Reference 5 Standard Review Plan.

Changes to the Updated Final Safety Analysis Report (UFSAR), required as a result of implementing AREVA ATRIUM-10 fuel, were previously addressed during the BFN Units 2 and 3 AREVA fuel transition. Changes to the following UFSAR sections were made during the initial implementation for BFN Units 2 and 3:

- Section 3.2 — Fuel Mechanical Design
- Section 3.3 — Reactor Vessel Internals Mechanical Design
- Section 3.6 — Nuclear Design
- Section 3.7 — Thermal and Hydraulic Design
- Section 6.5 — Safety Evaluation
- Section 13.10 — Refueling Test Program
- Section 14.4 — Approach to Safety Analysis
- Section 14.5 — Analyses of Abnormal Operational Transients
- Section 14.6 — Analysis of Design Basis Accidents

Given the UFSAR applies to all three units, and the AREVA fuel product for Unit 1 is the same design used in Units 2 and 3, introduction of AREVA ATRIUM-10 fuel into Unit 1 does not require any changes to the UFSAR.

The AREVA analytical methods and topical reports to be added to Technical Specification 5.6.5.b are those utilized to evaluate the fuel mechanical design, along with both cycle dependent and independent safety analyses, used to establish limits identified in the COLR. Additionally, Reference 4 is also being added to the Technical Specifications as the basis for acceptance of the ATRIUM-10 fuel design.

Each analytical methodology being added to Technical Specification 5.6.5.b has been previously reviewed and approved by the NRC. In August 2008, the NRC staff performed additional reviews of the AREVA analytical methods, specifically with EPU application in mind. The review concluded AREVA methodologies are adequate for application to EPU conditions, with two exceptions. These exceptions are related to the impact on calculated vessel overpressure arising from potential void quality correlation uncertainties. Information on how this concern is addressed is contained in Attachment 18. It should be noted that AREVA has chosen to apply the corrective action from this issue to non EPU conditions; the overpressure results in Attachments 12 (105% OLTP) and 13 (105% OLTP) of this supplement reflect this corrective action.

The impact of the ATRIUM-10 design on the 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 the transition cycle, are presented in Attachment 12 (105% OLTP).

Introduction of the ATRIUM-10 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 (105% OLTP) includes the cycle specific evaluation of the control rod drop accident for the 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 7).

Regarding the LOCA analysis (UFSAR section 14.6.3), a baseline LOCA break spectrum analysis of ATRIUM-10 fuel was previously performed, covering all three BFN units; it is included as Attachment 14. Cycle specific fuel design 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 (105% OLTP). The introduction of ATRIUM-10 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). As noted previously, this EPU based report is bounding for non EPU power.

The ATRIUM-10 design will also 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 for an ATRIUM-10 bundle (154) is higher than the number calculated to fail in a 7x7 bundle (111), the activity is allocated over a greater number of rods. The ATRIUM-10 bundle has the equivalent of 88.33 fuel length rods (83 full length plus 8 partial length rods with approximately two thirds the full length), while the 7x7 bundle has 49 full length rods. Therefore, the accident release with ATRIUM-10 fuel would be approximately (154/111) x (49/88.33), or 77% of the release from the design basis 7x7 fuel. Consequently, the fuel handling accident described in the UFSAR remains bounding for ATRIUM-10 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 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 consequences of a main steam line break accident.

The NRC has previously reviewed and approved transitions from GE14 to ATRIUM-10 (see section 4.1 below). Previous reviews confirmed the acceptability of transitioning from GE14 to ATRIUM-10. The scope of the technical analyses provided in support of the Unit 1 submittal is consistent with, and surpasses, the technical analyses provided with the precedent submittals.

In summary, the ATRIUM-10 fuel design fully complies with applicable fuel licensing criteria provided in Reference 5, as documented in Reference 4. The analytical methodologies to be used for design and licensing of ATRIUM-10 reloads are 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. The proposed changes to Technical Specifications 5.6.5.b, 3.3.4.1, and 3.7.5, are necessary and appropriate to implement the AREVA fuel design, and associated analytical methodologies. Given the prior transition to ATRIUM-10 fuel on BFN Units 2 & 3, the required changes to the UFSAR have already been completed.

## 4.0 REGULATORY EVALUATION

### 4.1 PRECEDENT

A search of NRC actions on Technical Specification changes revealed the Nuclear Regulatory Commission has previously approved similar changes for the following plants:

- "Browns Ferry Nuclear Plant, Units 2 and 3 - Issuance of Amendments Regarding Core Operating Limits (TAC Nos. MB8433 and MB8434)," December 30, 2003. (ML033650142)

- "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment Nos. 246 and 274 to Renewed Facility Operating Licenses Nos. DPR-71 and DPR-62, Carolina Power & Light Company, Brunswick Steam Electric Plant, Units 1 and 2, Docket Nos. 50-325 and 50-324," March 27, 2008. (ML080870546)

### 4.2 SIGNIFICANT HAZARDS CONSIDERATIONS

This analysis addresses the proposed change to amend Operating License DPR-33 for BFN Unit 1 to allow the use of AREVA fuel and analytical methodologies.

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 fuel designs and making an editorial change to TS will not increase the probability of a loss of coolant accident. 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 the fuel design and making an editorial change to TS 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 ATRIUM-10 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 ATRIUM-10 design is similar to other designs in use at Browns Ferry, and is well within the design capability of the refueling equipment. The consequences of the refueling accident are similar to the current GE14 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 ATRIUM-10 fuel channel is mechanically compatible with the co-resident fuel, and existing control blade designs. The mechanical interaction and friction forces between the ATRIUM-10 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 ATRIUM-10 fuel. Control rod drop accident consequences are evaluated on a cycle specific basis, confirming the number of calculated 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 ATRIUM-10 fuel product has been designed to maintain neutronic, thermal-hydraulic, and mechanical compatibility with the NSSS vendor fuel designs. The ATRIUM-10 fuel has been designed to meet fuel licensing criteria specified in NUREG-08000, "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 an editorial change to TS 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 ATRIUM-10 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 10CFR50.67. 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.

## 4.3 CONCLUSION

The proposed use of ATRIUM-10 fuel (using BLEU or commercial grade uranium), and the adoption of AREVA analytical methodologies for BFN Unit 1 are acceptable based on the following:

- ATRIUM-10 fuel has been designed to comply with the fuel related licensing criteria specified in the Standard Review Plan (Reference 5).
- Analytical methodologies being added to the Technical Specifications have been previously reviewed and approved by NRC.
- Analytical methodologies have been reviewed by the NRC and found to be acceptable, with the caveat of two restrictions related to vessel overpressure margins. These two restrictions have been incorporated into Unit 1 transition analyses.
- Transition core design analyses demonstrate acceptability of using ATRIUM-10 in Unit 1, including mixed core compatibility with co-resident GE14 fuel.
- The impacts of BLEU material do not adversely impact the neutronic, thermal-hydraulic, or mechanical performance of the fuel, including analytical methods used to perform these evaluations.

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)(10). 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. Letter from R. G. Jones (TVA) to NRC, "Browns Ferry Nuclear Plant (BFN) - Units 1, 2, and 3 - Technical Specifications (TS) Change 421 - Framatome Fuel Design and Storage," dated February 13, 2003.

2. Letter from Ms. Eva A. Brown (NRC) to TVA, "Summary of January 28, 2009, Meeting with the Tennessee Valley Authority Regarding Proposed Fuel Transition Amendment (TAC No. ME0438)," dated March 23, 2009.

3. Letter from Ms. Eva A. Brown (NRC) to TVA, "Summary of March 16, 2009, Meeting with the Tennessee Valley Authority Regarding Proposed Fuel Transition Amendment (TAC No. ME0438)," dated June 3, 2009.

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

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

6. 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.

7. 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.

8. EMF-2209(P)(A), Revision 2, "SPCB Critical Power Correlation," Framatome ANP, September 2003.

9. EMF-2209(P)(A), Revision 2, Addendum 1 Revision 0, "SPCB Additive Constants for ATRIUM-10 Fuel," AREVA NP, April 2008.

10. Letter from R. L. Gardener (AREVA NP, Inc.) to Document Control Desk (NRC), NRC:09:092, "10 CFR Part 21 Notification of an Error in LaSalle Units 1 & 2 Power Dependent MCPR and LHGR Operating Limits Calculation Due to High Measured Liftoff," AREVA NP Inc., dated August 27, 2009.

11. 1008192, "BWRVIP-130: BWR Vessel and Internals Project, BWR Water Chemistry Guidelines - 2004," EPRI, October 2004.

12. 1016579, "BWRVIP-190: BWR Vessel and Internals Project, BWR Water Chemistry Guidelines - 2008," EPRI, October 2008.

13. Letter from Thomas B. Blount (NRC) to R. L. Gardener (AREVA NP, Inc.), Final Safety Evaluation for AREVA NP, Inc. (AREVA) Topical Reports (TR) EMF-2209P, Revision 2, Addendum 1, "SPCB Additive Constants For ATRIUM-10 Fuel," and ANP-10249 (P), Revision 0, Supplement 1, "ACE Additive Constants For ATRIUM -10 Fuel," (TAC Nos. MD8754 AND ME0162), dated September 23, 2009.

14. Letter from TVA to NRC, "Technical Specifications Change Request TS-467 – Utilization of AREVA Fuel and Associated Analysis Methodologies," October 23, 2009.

**ATTACHMENT 2 (105% OLTP)**

**Browns Ferry Nuclear Plant (BFN)**
**Unit 1**

**Technical Specifications (TS) Change 467-S**

**Revision of Technical Specifications to allow utilization of AREVA NP fuel and associated analysis methodologies**

**Proposed Technical Specifications Changes (Mark-up)**

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**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.**

## 3.2 POWER DISTRIBUTION LIMITS

### 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

LCO 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. Any LHGR not within limits. | A.1 | Restore LHGR(s) to within limits. | 2 hours |
| B. Required Action and associated Completion Time not met. | B.1 | Reduce THERMAL POWER to < 25% RTP. | 4 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.2.3.1 | Verify all LHGRs are less than or equal to the limits specified in the COLR. | Once within 12 hours after ≥ 25% RTP<br><br><u>AND</u><br><br>24 hours thereafter |

## 3.3 INSTRUMENTATION

### 3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

LCO 3.3.4.1

a. Two channels per trip system for each EOC-RPT instrumentation Function listed below shall be OPERABLE:

1. Turbine Stop Valve (TSV) - Closure; and

2. Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure - Low.

OR

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for inoperable EOC-RPT as specified in the COLR are made applicable; and

c. LCO 3.2.3, “LINEAR HEAT GENERATION RATE (LHGR),” limits for an inoperable EOC-RPT, as specified in the COLR, are made applicable.

APPLICABILITY: THERMAL POWER ≥ 30% RTP.

ACTIONS

----------------------------------------------------NOTE----------------------------------------------------
Separate Condition entry is allowed for each channel.
------------------------------------------------------------------------------------------------------------------

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more channels inoperable. | A.1 Restore channel to OPERABLE status.<br><br><u>OR</u><br><br>A.2 -------------NOTE-------------<br>Not applicable if inoperable channel is the result of an inoperable breaker.<br>----------------------------------<br><br>Place channel in trip. | 72 hours<br><br><br><br><br><br><br><br>72 hours |
| B. One or more Functions with EOC-RPT trip capability not maintained.<br><br><u>AND</u><br><br>MCPR and LHGR limits for inoperable EOC-RPT not made applicable. | B.1 Restore EOC-RPT trip capability.<br><br><u>OR</u><br><br>B.2 Apply ~~the~~ MCPR and LHGR limits for inoperable EOC-RPT as specified in the COLR. | 2 hours<br><br><br><br>2 hours |
| C. Required Action and associated Completion Time not met. | C.1 Reduce THERMAL POWER to < 30% RTP. | 4 hours |

## 3.7 PLANT SYSTEMS

### 3.7.5 Main Turbine Bypass System

LCO 3.7.5 The Main Turbine Bypass System shall be OPERABLE.

OR

The following limits are made applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR.

APPLICABILITY: THERMAL POWER $\geq$ 25% RTP.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. Requirements of the LCO not met. | A.1 | Satisfy the requirements of the LCO. | 2 hours |
| B. Required Action and associated Completion Time not met. | B.1 | Reduce THERMAL POWER to < 25% RTP. | 4 hours |

5.6.4 (Deleted).

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

(1) The APLHGRs for Specification 3.2.1;

(2) The LHGR for Specification 3.2.3;

(3) The MCPR Operating Limits for Specification 3.2.2; and

(4) The RBM setpoints and applicable reactor thermal power ranges for each of the setpoints for Specification 3.3.2.1, Table 3.3.2.1-1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents (*latest approved versions applicable to BFN*): ~~NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved version for BFN).~~

1. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel.

2. XN-NF-81-58(P)(A), RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model.

3. XN-NF-85-67(P)(A), Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel.

4. EMF-85-74(P)(A), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model.

5. ANF-89-98(P)(A), Generic Mechanical Design Criteria for BWR Fuel Designs.

(continued)

5.6 Reporting Requirements (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. 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.

*The COLR will contain the complete identification for each of the Technical Specification referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date and any supplements)*

(continued)

**ATTACHMENT 3 (105% OLTP)**

**Browns Ferry Nuclear Plant (BFN)**
**Unit 1**

**Technical Specifications (TS) Change 467-S**

**Revision of Technical Specifications to allow utilization of AREVA NP fuel and associated analysis methodologies**

**Retyped Proposed Technical Specifications Pages**

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**The following pages have been revised to reflect the proposed changes. These are the retyped pages relative to the markups found in Attachment 2 (105% OLTP).**

## 3.2 POWER DISTRIBUTION LIMITS

### 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

LCO 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. Any LHGR not within limits. | A.1 | Restore LHGR(s) to within limits. | 2 hours |
| B. Required Action and associated Completion Time not met. | B.1 | Reduce THERMAL POWER to < 25% RTP. | 4 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.2.3.1 | Verify all LHGRs are less than or equal to the limits specified in the COLR. | Once within 12 hours after ≥ 25% RTP<br><br><u>AND</u><br><br>24 hours thereafter |

## 3.3 INSTRUMENTATION

### 3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

LCO 3.3.4.1

a. Two channels per trip system for each EOC-RPT instrumentation Function listed below shall be OPERABLE:

1. Turbine Stop Valve (TSV) - Closure; and

2. Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure - Low.

<u>OR</u>

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for inoperable EOC-RPT as specified in the COLR are made applicable; and

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for an inoperable EOC-RPT, as specified in the COLR, are made applicable.

APPLICABILITY: THERMAL POWER ≥ 30% RTP.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.
-----

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. One or more channels inoperable. | A.1 | Restore channel to OPERABLE status. | 72 hours |
| | <u>OR</u> | | |
| | A.2 | -----NOTE-----<br>Not applicable if inoperable channel is the result of an inoperable breaker.<br>-----<br><br>Place channel in trip. | 72 hours |
| B. One or more Functions with EOC-RPT trip capability not maintained.<br><br><u>AND</u><br><br>MCPR and LHGR limits for inoperable EOC-RPT not made applicable. | B.1 | Restore EOC-RPT trip capability. | 2 hours |
| | <u>OR</u> | | |
| | B.2 | Apply MCPR and LHGR limits for inoperable EOC-RPT as specified in the COLR. | 2 hours |
| C. Required Action and associated Completion Time not met. | C.1 | Reduce THERMAL POWER to < 30% RTP. | 4 hours |

## 3.7 PLANT SYSTEMS

### 3.7.5 Main Turbine Bypass System

LCO 3.7.5 The Main Turbine Bypass System shall be OPERABLE.

OR

The following limits are made applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR; and

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. Requirements of the LCO not met. | A.1 | Satisfy the requirements of the LCO. | 2 hours |
| B. Required Action and associated Completion Time not met. | B.1 | Reduce THERMAL POWER to < 25% RTP. | 4 hours |

5.6.4 (Deleted).

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

(1) The APLHGRs for Specification 3.2.1;

(2) The LHGR for Specification 3.2.3;

(3) The MCPR Operating Limits for Specification 3.2.2; and

(4) The RBM setpoints and applicable reactor thermal power ranges for each of the setpoints for Specification 3.3.2.1, Table 3.3.2.1-1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents (*latest approved versions applicable to BFN*):

1. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel.

2. XN-NF-81-58(P)(A), RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model.

3. XN-NF-85-67(P)(A), Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel.

4. EMF-85-74(P)(A), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model.

5. ANF-89-98(P)(A), Generic Mechanical Design Criteria for BWR Fuel Designs.

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

(continued)

5.6 Reporting Requirements (continued)

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. 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.

*The COLR will contain the complete identification for each of the Technical Specification referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date and any supplements)*

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