TVA

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April 16, 2010

10 CFR 50.4
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-473 - AREVA Fuel Transition**

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," dated October 23, 2009

4. Letter from TVA to NRC, "Technical Specification Change TS-467-S - Utilization of AREVA Fuel and Associated Analysis Methodologies – Non EPU Supplement," dated November 17, 2009

5. Letter from NRC to TVA, "Browns Ferry Nuclear Plant Unit 1 - Nonacceptance of Utilization of AREVA Fuel and Associated Analysis Methodologies (TAC No. ME2451)(TS-467)," dated December 23, 2009

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6. Letter from TVA to NRC, "Response to NRC Request for Supplemental Information Regarding Technical Specification Change TS-467 – Utilization of AREVA Fuel and Associated Analysis Methodologies," dated January 15, 2010

7. Letter from TVA to NRC, "Technical Specification Change TS-467 - Utilization of AREVA Fuel and Associated Analysis Methodologies; Withdrawal of Request," dated February 2, 2010

8. Letter from NRC to TVA, "Browns Ferry Nuclear Plant Unit 1 - Withdrawal of an Amendment Request to Utilize AREVA Fuel and Associated Methodologies (TAC No. ME2451)," dated February 25, 2010

By letter dated October 23, 2009 (Reference 3) and as supplemented by letter dated November 17, 2009 (Reference 4), the Tennessee Valley Authority (TVA) submitted a request for amendment to the Technical Specifications (TS) for Browns Ferry Nuclear Plant (BFN), Unit 1. The amendment request 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 TS changes were also requested to reflect the AREVA NP specific methods for monitoring and enforcing of the thermal limits. In a letter dated December 23, 2009 (Reference 5), the NRC requested that supplemental information be provided by January 15, 2010. By letter dated January 15, 2010 (Reference 6), TVA provided responses to the NRC request for supplemental information.

In the Reference 6 response, TVA identified that certain evaluations were being conducted and would not be available until March 2010. Accordingly, by letter dated February 2, 2010 (Reference 7), TVA withdrew Technical Specification Change TS-467 (Reference 3) and the associated non Extended Power Uprate (EPU) Supplement, TS-467-S (Reference 4).

The evaluations discussed in the Reference 6 response have now been completed. As a result, in accordance with the provisions of 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," TVA is submitting a request for amendment to the TS for BFN, Unit 1.

To support the transition to AREVA fuel, the proposed amendment adds the AREVA NP analysis methodologies to the list of approved methods to be used in determining the core operating limits in the COLR. Additional TS changes are requested to reflect the AREVA NP specific methods for monitoring and enforcing of the thermal limits. At this time, TVA is requesting approval for the transition to AREVA fuel for BFN, Unit 1 for non EPU conditions (i.e., 105% Original Licensed Thermal Power level) only.

The enclosure provides a description of the proposed changes, technical evaluation of the proposed changes, regulatory evaluation, and a discussion of environmental considerations. Attachment 1 identifies regulatory commitments. 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 27 of this submittal. These attachments also provide the information requested during meetings, summarized in References 1 and 2, between TVA and NRC representatives. In addition, the information previously submitted in Reference 6 has been modified to address the additional issues identified by the NRC in Reference 8 and is provided in the attachments.

Attachments 6, 8, 10, 12, 16, 24, and 26 to this 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), it is requested that such information be withheld from public disclosure. Attachment 14 to this letter contains information that Global Nuclear Fuel – Americas 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. Attachments 18 and 20 contain information that GE Hitachi Nuclear Energy Americas 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 28 provides the affidavits supporting these requests. Attachments 7, 9, 11, 13, 15, 17, 19, 21, 25, and 27 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, the enclosure, and the non-proprietary attachments to the Alabama State Department of Public Health.

TVA requests approval of these TS changes by April 18, 2011 in order to support fuel design and fabrication activities for Unit 1 Cycle 10. TVA also requests that the implementation of the revised TS be made prior to the startup of Unit 1 for Cycle 10, which is currently scheduled for the fall of 2012.

There are regulatory commitments in this submittal as reflected in Attachment 1.

Please direct any questions concerning this matter to Terry Cribbe at (423) 751-3850.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on the 16th day of April 2010.

Respectfully,

R. M. Krich

Enclosure:

AREVA Fuel Transition

Attachments:

1	List of 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-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	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
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10	Fuel Cycle Design Report (*proprietary*) ANP-2859(P), Revision 0, Browns Ferry Unit 1 Fuel Cycle Design (105% OLTP), AREVA NP Inc., September 2009
11	Fuel Cycle Design Report (*non-proprietary*) ANP-2859(NP), Revision 0, Browns Ferry Unit 1 Fuel Cycle Design (105% OLTP), AREVA NP Inc., September 2009
12	Reload Safety Analysis Report (*proprietary*) ANP-2863(P), Revision 1, Browns Ferry Unit 1 Cycle 9 Reload Safety Analysis for 105% OLTP, AREVA NP Inc., March 2010

13	Reload Safety Analysis Report (*non-proprietary*) ANP-2863(NP), Revision 1, Browns Ferry Unit 1 Cycle 9 Reload Safety Analysis for 105% OLTP, AREVA NP Inc., March 2010
14	GE14 Fuel Thermal-Mechanical Information (*proprietary*) GNF 0000-0111-8036-R0-P, Revision 0, GE14 Fuel Thermal-Mechanical Information, Global Nuclear Fuel, January 2010
15	GE14 Fuel Thermal-Mechanical Information (*non-proprietary*) GNF 0000-0111-8036-R0-NP, Revision 0, GE14 Fuel Thermal-Mechanical Information, Global Nuclear Fuel, January 2010
16	LOCA Break Spectrum Analysis Report (*proprietary*) ANP-2908(P), Revision 0, Browns Ferry Units 1, 2, and 3 105% OLTP LOCA Break Spectrum Analysis, AREVA NP Inc., March 2010
17	LOCA Break Spectrum Analysis Report (*non-proprietary*) ANP-2908(NP), Revision 0, Browns Ferry Units 1, 2, and 3 105% OLTP LOCA Break Spectrum Analysis, AREVA NP Inc., March 2010
18	SAFER/GESTR-LOCA Analysis Report (*proprietary*) NEDC-32484P, Revision 7, Browns Ferry Nuclear Plant Units 1, 2, & 3 SAFER/GESTR-LOCA Loss-Of-Coolant Accident Analysis, GE Hitachi Nuclear Energy, January 2010
19	SAFER/GESTR-LOCA Analysis Report (*non-proprietary*) NEDC-32484, Revision 7, Browns Ferry Nuclear Plant Units 1, 2, & 3 SAFER/GESTR-LOCA Loss-Of-Coolant Accident Analysis, GE Hitachi Nuclear Energy, January 2010
20	Supplementary Report Regarding ECCS-LOCA Additional Single Failure Evaluation (*proprietary*) NEDC-32484P, Supplement 1, Revision 0, Browns Ferry Nuclear Plant Unit 1 Supplementary Report Regarding ECCS-LOCA Additional Single Failure Evaluation at Current Licensed Thermal Power, GE Hitachi Nuclear Energy, April 2010
21	Supplementary Report Regarding ECCS-LOCA Additional Single Failure Evaluation (*non-proprietary*) NEDO-32484, Supplement 1, Revision 0, Browns Ferry Nuclear Plant Unit 1 Supplementary Report Regarding ECCS-LOCA Additional Single Failure Evaluation at Current Licensed Thermal Power, GE Hitachi Nuclear Energy, April 2010
22	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

23	Boiling Water Reactor Licensing Methodology Compendium (*non-proprietary*) ANP-2637, Revision 3, Boiling Water Reactor Licensing Methodology Compendium, AREVA NP Inc., March 2010
24	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
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26	Part 1: Previous NRC Requests for Additional Information Matrix and Text (*proprietary*) Part 2: Browns Ferry Unit 1 – Summary of Response to Requests for Additional Information (*proprietary*) ANP-2860(P), Revision 2, Browns Ferry Unit 1 – Summary of Responses to Requests for Additional Information, AREVA NP Inc., October 2009 Part 3: Response to Previous NRC Requests for Supplemental Information Regarding Utilization of AREVA Fuel and Associated Methodologies (*proprietary*)
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28	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 473

AREVA Fuel Transition

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. BFN, 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 2012. This evaluation supports the transition to AREVA fuel for non Extended Power Uprate (EPU) power conditions (i.e., 105% Original Licensed Thermal Power (OLTP) level).

2.0 DETAILED DESCRIPTION

The Tennessee Valley Authority (TVA) intends to begin utilizing the ATRIUM-10 design in BFN, Unit 1 Cycle 10. The first reload of ATRIUM-10 targeted for insertion into the core is the fall 2012 outage. The ATRIUM-10 product is an industry proven fuel design in use at BFN, Unit 2 and BFN, Unit 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 BFN, Unit 1 with standard commercial grade uranium in future reloads. At this time, TVA is requesting approval for the transition to AREVA fuel for BFN, Unit 1 for non EPU conditions (i.e., 105% OLTP) only.

In order to extend the use of this fuel design to BFN, Unit 1, several changes to the Technical Specifications (TS) are required. Technical Specification 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 Technical Specification 5.6.5.b will be revised to add appropriate NRC approved AREVA analytical methodologies.

Also, Technical Specification 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 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.

TS changes will also be made to Technical Specification 3.3.1.1 (Reactor Protection Systems Instrumentation) for the Oscillation Power Range Monitor (OPRM) Upscale Function to indicate that OPRM period based detection algorithm setpoint limits are included in Core Operating Limits Report (COLR) since these limits are evaluated each reload using analytical methodologies included in the COLR. Corresponding changes will also be made to TS 5.6.5.a to include the OPRM setpoint as a COLR item.

The submittal also addresses the required changes to the TS 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 TS do not require alteration, and are not included in the current change request. Additional information regarding the prior change with respect to spent fuel pool storage is provided in Attachment 26 (Part 3).

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 24, and 26 (Parts 1 and 2). Some of the prior RAIs were related to application of AREVA analysis methodologies to EPU conditions. Although, as stated above, this request is for non EPU conditions, the responses to the prior EPU-related RAIs have been included in the Attachments to this submittal for completeness. Attachment 22 provides responses to technical items identified in Reference 2, along with information on BLEU material.

As a result of their review of References 11 and 12, the NRC in a letter dated December 23, 2009 (Reference 13), requested that supplemental information be provided by January 15, 2010. The response to this request was submitted to the NRC on January 15, 2010 (Reference 14). TVA subsequently withdrew the application submitted by the Reference 11 and 12 letters on February 2, 2010. The information provided in Reference 14, as modified to reflect the additional issues identified by the NRC in Reference 15, is included in Attachment 26 (Part 3).

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

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28	Affidavits

Attachments 6, 8, 10, and 12 provide the Mechanical Design Report, the Thermal-Hydraulic Design Report, the Fuel Cycle Design Report, and the Reload Safety Analysis Report, respectively. These reports were developed to support an AREVA fuel transition for BFN, Unit 1 occurring in Cycle 9. However, due to schedule considerations, the transition to AREVA fuel for BFN, Unit 1 is now planned for Cycle 10. Attachments 6, 8, 10, and 12 are considered to be representative information for a transition from GE14 to ATRIUM-10 fuel for BFN, Unit 1. For Cycle 10, the same methods and processes (reflected in TS 5.6.5.b) used for the development of the Mechanical Design Report, the Thermal-Hydraulic Design Report, the Fuel Cycle Design Report, and the Reload Safety Analysis Report for BFN, Unit 1 Cycle 9 will be used for Cycle 10. In accordance with normal reload design schedules and procedures, these activities for Cycle 10 will be performed after the anticipated approval of this amendment request, using 10 CFR 50.59, "Changes, tests and experiments."

A 10 CFR 21 issue was identified in Reference 8. 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 9 and 10. 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.

The LOCA analyses provided in Attachments 16, 17, 18, 19, 20, and 21 do not account for the recent GE Hitachi (GEH) potential 10 CFR 21 issue related to increased leakage in the core spray system. This issue was communicated to the industry by GEH in Safety Communication (SC) 10-05 dated March 15, 2010. TVA has evaluated this potential 10 CFR 21 issue for applicability to the BFN units and has determined that the issue only applies to BFN, Unit 3. The AREVA LOCA analysis (Attachments 16 and 17) applies to all three BFN units. The issuance of SC 10-05 occurred after the work supporting the AREVA analysis was completed. Therefore, the impact of this potential 10 CFR 21 issue on the BFN, Unit 3 peak clad temperature will be reported and tracked under the 10 CFR 50.46 process. The GEH LOCA analysis in Attachments 18 and 19 is indicated to be applicable to all three BFN units. However, the GEH LOCA analysis is only relied upon to support operation and fuel transition work for BFN, Unit 1. Since the potential 10 CFR 21 issue does not impact BFN, Unit 1, Attachments 18 and 19 are unaffected. The GEH reports in Attachments 20 and 21 are BFN, Unit 1 specific, and are unaffected by the issue.

In addition, during an NRC staff review of AREVA analytical methods in August 2008, the NRC raised concerns over the impact of a potential non-conservative bias in the void quality correlations. To address these concerns, as discussed in Part 2 of Attachment 26, AREVA applies conservative adders to the calculated peak vessel pressure analysis results. These adders have been applied to the vessel overpressure analysis results provided in Attachment 12.

3.0 TECHNICAL EVALUATION

The fuel design to be introduced into BFN, Unit 1 in 2012 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 Reference 5.

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 TS 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.

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 a representative transition cycle, are presented in Attachment 12.

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 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 7).

Regarding the LOCA analysis (UFSAR section 14.6.3), a baseline LOCA break spectrum analysis of ATRIUM-10 fuel has been performed covering all three BFN units at 105% OLTP power conditions; it is included as Attachment 16. Attachments 18 and 20 provide the LOCA analysis which is applicable for GE14 co-resident fuel in BFN, Unit 1. 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 for a representative transition cycle. 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).

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 full 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.

For BFN, Unit 1, the transition work employed the AREVA methods referenced in the proposed TS change (Attachment 2) to evaluate both the ATRIUM-10 fuel and the co-resident GNF GE14 fuel. The AREVA methodologies are applied in accordance with NRC approval for performance of design and licensing analyses for mixed cores. The thermal-hydraulic characteristics of the GE14 fuel design were explicitly accounted for and a detailed thermal-hydraulic analysis of the mixed core was performed as documented in Attachment 8.

The GE14 lattice and fuel bundle geometry, as well as the specific enrichment and burnable poison distributions, were explicitly modeled in the neutronic analyses provided in Attachment 10. The AREVA neutronic methods have been extensively benchmarked for cores containing GE14 fuel, including multiple BFN cycles containing that fuel type.

The reload safety analysis documented in Attachment 12 explicitly evaluates the transient and accident response of the mixed core. Since the GE14 fuel is explicitly modeled, the impact of the co-resident fuel on the core response to transient and accident events is accounted for. TVA provided data to AREVA to ensure the critical power thermal limits of the co-resident fuel are appropriately calculated and monitored. In addition, TVA provided AREVA with thermal-mechanical limits for the co-resident fuel for use in the design and safety analyses, to ensure the appropriate limits are met. The core monitoring system will explicitly monitor the co-resident fuel, and will apply the specific thermal limits determined for that fuel.

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. In addition, the AREVA methodologies are applied in accordance with NRC approval for performance of design and licensing analyses for mixed cores. The proposed changes to TS 5.6.5, 3.3.1.1, 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 and 3, the required changes to the UFSAR have already been completed.

4.0 REGULATORY EVALUATION

4.1 PRECEDENT

A search of NRC actions on TS changes revealed the NRC 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 BFN 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-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 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) and the following regulatory requirements:
    - 10 CFR 50, Appendix A, General Design Criteria (GDC) 10, "Reactor design," GDC 27, "Combined reactivity control system capability," and GDC 35, "Emergency core cooling;"
    - 10 CFR 100, "Reactor Site Criteria;"
    - 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors;" and
    - 10 CFR 50, Appendix K, "ECCS Evaluation Models."

- Analytical methodologies being added to the TS 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 concerns related to vessel overpressure margins. These two concerns have been addressed in the applicable Unit 1 transition analyses provided in Attachment 12.

- 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. 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.

9. 1008192, "BWRVIP-130: BWR Vessel and Internals Project, BWR Water Chemistry Guidelines - 2004," EPRI, October 2004.

10. 1016579, "BWRVIP-190: BWR Vessel and Internals Project, BWR Water Chemistry Guidelines - 2008," EPRI, October 2008.

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

12. Letter from TVA to NRC, "Technical Specifications Change Request TS-467S – Utilization of AREVA Fuel and Associated Analysis Methodologies - Non EPU Supplement," November 17, 2009.

13. Letter from NRC to TVA, "Browns Ferry Nuclear Plant Unit 1 - Nonacceptance of Utilization of AREVA Fuel and Associated Analysis Methodologies (TAC No. ME2451)(TS-467)," December 23, 2009.

14. Letter from TVA to NRC, "Response to NRC Request for Supplemental Information Regarding Technical Specification Change TS-467 - Utilization of AREVA Fuel and Associated Analysis Methodologies," January 15, 2010.

15. Letter from NRC to TVA, "Browns Ferry Nuclear Plant Unit 1 - Withdrawal of an Amendment Request to Utilize AREVA Fuel and Associated Methodologies (TAC No. ME2451)," February 25, 2010.

ATTACHMENT 1

Browns Ferry Nuclear Plant (BFN)
Unit 1

Technical Specifications (TS) Change 473

AREVA Fuel Transition

Regulatory Commitments

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1. ADS will be modified to provide a single failure proof automatic initiation capability of 4 ADS valves, regardless of which 250 VDC battery fails. This modification is expected to be made to Unit 3 during the Unit 3 outage in the Spring of 2012, to Unit 1 during the Unit 1 outage in the Fall of 2012, and to Unit 2 during the Unit 2 outage in the Spring of 2013.

2. These following revisions will be proposed for BFN Units 2 and 3 in future Technical Specification Change Request(s).

   a. Revisions to TS 3.3.1.1, Reactor Protection Systems Instrumentation, for the Oscillation Power Range Monitor (OPRM) Upscale Function, i.e., Function 2.f, to indicate that OPRM period based detection algorithm setpoint limits are included in Core Operating Limits Report (COLR).

   b. Corresponding revisions are to TS 5.6.5.a to include the OPRM setpoint as a COLR item.

   c. Revisions to TS 5.6.5.b to include the AREVA stability related Topical Reports which describe the analytical methods used for determining the OPRM period based detection algorithm setpoint limits.

ATTACHMENT 2

Browns Ferry Nuclear Plant (BFN)
Unit 1

Technical Specifications (TS) Change 473

AREVA Fuel Transition

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.

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 <u>AND</u> 24 hours thereafter

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Inop	1,2	3(b)	G	SR 3.3.1.1.16	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA
f. OPRM Upscale	1	3(b)	I	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16 SR 3.3.1.1.17	NA(e)
3. Reactor Vessel Steam Dome Pressure - High(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1090 psig
4. Reactor Vessel Water Level - Low, Level 3(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 528 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Each APRM channel provides inputs to both trip systems.

(d) During instrument calibrations, if the As Found channel setpoint is conservative with respect to the Allowable Value but outside its acceptable As Found band as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence that the channel can perform as required before returning the channel to service in accordance with the Surveillance. If the As Found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

Prior to returning a channel to service, the instrument channel setpoint shall be calibrated to a value that is within the acceptable As Left tolerance of the setpoint; otherwise, the channel shall be declared inoperable.

The nominal Trip Setpoint shall be specified on design output documentation which is incorporated by reference in the Updated Final Safety Analysis Report. The methodology used to determine the nominal Trip Setpoint, the predefined As Found Tolerance, and the As Left Tolerance band, and a listing of the setpoint design output documentation shall be specified in Chapter 7 of the Updated Final Safety Analysis Report.

(e) Refer to COLR for OPRM period based detection algorithm (PBDA) setpoint limits.

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.	72 hours
	OR		
	A.2	------------NOTE------------ Not applicable if inoperable channel is the result of an inoperable breaker. -------------------------------- Place channel in trip.	72 hours
B. One or more Functions with EOC-RPT trip capability not maintained. AND MCPR and LHGR limits for inoperable EOC-RPT not made applicable.	B.1	Restore EOC-RPT trip capability.	2 hours
	OR		
	B.2	Apply ~~the~~ 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 period based detection algorithm (PBDA) setpoint for Function 2.f, Oscillation Power Range Monitor (OPRM) Upscale, for Specification 3.3.1.1; and

(45) 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.

18. EMF-CC-074(P)(A) Volume 4, BWR Stability Analysis - Assessment of STAIF with input from MICROBURN-B2.

(continued)

5.6 Reporting Requirements (continued)

19. BAW-10255(P)(A), Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code.

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

Browns Ferry Nuclear Plant (BFN)
Unit 1

Technical Specifications (TS) Change 473

AREVA Fuel Transition

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.

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 <u>AND</u> 24 hours thereafter

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Inop	1,2	3(b)	G	SR 3.3.1.1.16	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA
f. OPRM Upscale	1	3(b)	I	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16 SR 3.3.1.1.17	NA(e)
3. Reactor Vessel Steam Dome Pressure - High(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1090 psig
4. Reactor Vessel Water Level - Low, Level 3(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 528 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Each APRM channel provides inputs to both trip systems.

(d) During instrument calibrations, if the As Found channel setpoint is conservative with respect to the Allowable Value but outside its acceptable As Found band as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence that the channel can perform as required before returning the channel to service in accordance with the Surveillance. If the As Found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

Prior to returning a channel to service, the instrument channel setpoint shall be calibrated to a value that is within the acceptable As Left tolerance of the setpoint; otherwise, the channel shall be declared inoperable.

The nominal Trip Setpoint shall be specified on design output documentation which is incorporated by reference in the Updated Final Safety Analysis Report. The methodology used to determine the nominal Trip Setpoint, the predefined As Found Tolerance, and the As Left Tolerance band, and a listing of the setpoint design output documentation shall be specified in Chapter 7 of the Updated Final Safety Analysis Report.

(e) Refer to COLR for OPRM period based detection algorithm (PBDA) setpoint limits.

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------------ Not applicable if inoperable channel is the result of an inoperable breaker. -------------------------------- Place channel in trip.	72 hours
B. One or more Functions with EOC-RPT trip capability not maintained. <u>AND</u> 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.

<u>OR</u>

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;

(4) The period based detection algorithm (PBDA) setpoint for Function 2.f, Oscillation Power Range Monitor (OPRM) Upscale, for Specification 3.3.1.1; and

(5) 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.

(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.

18. EMF-CC-074(P)(A) Volume 4, BWR Stability Analysis - Assessment of STAIF with input from MICROBURN-B2.

(continued)

5.6 Reporting Requirements (continued)

19. BAW-10255(P)(A), Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code.

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 4

Browns Ferry Nuclear Plant (BFN)
Unit 1

Technical Specifications (TS) Change 473

AREVA Fuel Transition

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.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

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

The fuel cladding integrity SL is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2 ~~for General Electric Company (GE) fuel~~. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

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

While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross, rather than

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity

~~GE critical power correlations are applicable for all critical power calculations at pressures ≥ 785 psig and core flows ≥ 10% of rated flow.~~ The SPCB critical power correlation is used for both AREVA and coresident 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 ATRIUM-10 and GE14 fuel types. 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 due only to elevation effects from liquid only in the channel, core bypass region, and annulus at zero power, zero flow is approximately 4.5 psi. At all operating conditions, this pressure differential is maintained by the bypass region of the core and the annulus region of the vessel. The elevation head provided by the annulus produces natural circulation flow conditions which have balancing pressure head and loss terms inside the core shroud. This natural circulation principle maintains a core plenum to plenum pressure drop of about 4.5 to 5 psid along the natural circulation flow line of the P/F operating map. In the range of power levels of interest, approaching 25% of rated power below which thermal margin monitoring is not required, the pressure drop and density head terms tradeoff for power changes such that natural circulation flow is nearly independent of reactor power. This characteristic is represented by the nearly vertical portion of the natural circulation line on the P/F operating map. Analysis has shown that the hot channel flow rate is >28,000 lb_m/hr (>0.23 x 10^6 $lb_m/hr\text{-}ft^2$) in the region of operation with power ~25% and core pressure drop of about 4.5 to 5 psid. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at 28,000

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

lb_m/hr is approximately 3 MW_t. With the design peaking factors, this corresponds to a core thermal power of more than 50%.

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 800 psia ~~(the limit of the range of applicability of GETAB/GEXL for GE fuel)~~. 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 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 10% of rated flow ~~(the limit of applicability of the GETAB/GEXL correlations for GE fuel)~~.

(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 ~~that~~ combin~~es~~ing 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 ~~General Electric Critical Power correlations. Details of the fuel cladding integrity SL calculation are given in Reference 2. Reference 2 also includes a tabulation of the uncertainties used in the determination of the MCPR SL and of the nominal values of the parameters used in the MCPR SL statistical analysis~~, AREVA critical power correlations. One specific uncertainty included in the SL is the uncertainty inherent in the 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. ~~3~~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. GE SIL No. 516, Supplement 2, January 19, 1996.~~

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. ANF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors," (*as identified in the COLR*).

~~3~~5. 10 CFR 50.67.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

SDM requirements are specified to ensure:

a. The reactor can be made subcritical from all operating conditions and transients and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

These requirements are satisfied by the control rods, as described in GDC 26 (Ref. 1), which can compensate for the reactivity effects of the fuel and water temperature changes experienced during all operating conditions.

APPLICABLE SAFETY ANALYSES

The control rod drop accident (CRDA) analysis (Refs. 2, ~~and~~ 3, 9, and 10) assumes the core is subcritical with the highest worth control rod withdrawn. Typically, the first control rod withdrawn has a very high reactivity worth and, should the core be critical during the withdrawal of the first control rod, the consequences of a CRDA could exceed the fuel damage limits for a CRDA (see Bases for LCO 3.1.6, "Rod Pattern Control"). Also, SDM is assumed as an initial condition for the control rod removal error during refueling (Ref. 4) and fuel assembly insertion error during refueling (Ref. 5) accidents. The analysis of these reactivity insertion events assumes the refueling interlocks are OPERABLE when the reactor is in the refueling mode of operation. These interlocks prevent the withdrawal of more

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR, Section 14.6.2.
3. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," Section S.2.2.3.1, August 1996.
4. FSAR, Section 14.5.3.3.
5. FSAR, Section 14.5.3.4.
6. FSAR, Section 3.6.5.2.
7. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1, August 1996.
8. NRC 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
9. 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*).
10. 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*).

BASES (continued)

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 2, 3, and 4. The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. The resulting negative scram reactivity forms the basis for the determination of plant thermal limits (e.g., the MCPR). Other distributions of scram times (e.g., several control rods scramming slower than the average time with several control rods scramming faster than the average time) can also provide sufficient scram reactivity. Surveillance of each individual control rod's scram time ensures the scram reactivity assumed in the DBA and transient analyses can be met.

The scram function of the CRD System protects the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)", and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), which ensure that no fuel damage will occur if these limits are not exceeded. Above 800 psig, the scram function is designed to insert negative reactivity at a rate fast enough to prevent the actual MCPR from becoming less than the MCPR SL, during the analyzed limiting power transient. Below 800 psig, the scram function is assumed to perform during the control rod drop accident (Refs. 5, 8, and 9) and, therefore, also provides protection against violating fuel damage limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control"). For the reactor vessel overpressure protection analysis, the scram function, along with the safety/relief valves, ensure that the peak vessel pressure is maintained within the applicable ASME Code limits.

Control rod scram times satisfy Criterion 3 of the NRC Policy Statement (Ref. 7).

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. FSAR, Section 3.4.6.
3. FSAR, Section 14.5.
4. FSAR, Section 14.6.
5. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1, August 1996.
6. Letter from R. F. Janecek (BWROG) to R. W. Starostecki (NRC), "BWR Owners Group Revised Reactivity Control System Technical Specifications," BWROG-8754, September 17, 1987.
7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
8. 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*).
9. 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*).

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND	Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to 10% RTP. The sequences limit the potential amount of reactivity addition that could occur in the event of a Control Rod Drop Accident (CRDA). This Specification assures that the control rod patterns are consistent with the assumptions of the CRDA analyses of References 1, ~~and~~ 2, 11, and 12.
APPLICABLE SAFETY ANALYSES	The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1, ~~and~~ 2, 11, and 12. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RWM (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated. Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage which could result in the undue release of radioactivity. Since the failure consequences for UO_2 have been shown to be insignificant below fuel energy

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

depositions of 300 cal/gm (Ref. 3), the fuel damage limit of 280 cal/gm provides a margin of safety from significant core damage which would result in release of radioactivity (Refs. 4 and 5). Generic evaluations (Refs. 1, ~~and~~ 6, and 11) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required ASME Code limits (Ref. 7) and the calculated offsite doses will be well within the required limits (Ref. 5).

Control rod patterns analyzed in References 1, 11, and 12, follow the banked position withdrawal sequence (BPWS). The BPWS is applicable from the condition of all control rods fully inserted to 10% RTP (Ref. 2). For the BPWS, the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are established to minimize the maximum incremental control rod worth without being overly restrictive during normal plant operation. ~~Generic analysis of the BPWS (Ref. 8) has~~ Analyses are performed using the Reference 11 methodology demonstrating~~ed that~~ the 280 cal/gm fuel damage limit will not be violated during a CRDA while following the BPWS mode of operation. The evaluation provided by the generic BPWS analysis (Ref. 8) allows a limited number (i.e., eight) and corresponding distribution of fully inserted, inoperable control rods, that are not in compliance with the sequence.

When performing a shutdown of the plant, an optional BPWS control rod sequence (Ref. 10) may be used provided that all withdrawn control rods have been confirmed to be coupled. The rods may be inserted without the need to stop at intermediate positions since the possibility of a CRDA is eliminated by the confirmation that withdrawn control rods are coupled. When using the Reference 10 control rod sequence for shutdown, the Rod Worth Minimizer may be reprogrammed

(continued)

BASES (continued)

REFERENCES

1. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," Section 2.2.3.1, August 1996.
2. Letter from T. Pickens (BWROG) to G. C. Lainas (NRC), Amendment 17 to General Electric Licensing Topical Report, NEDE-24011-P-A, August 15, 1986.
3. NUREG-0979, Section 4.2.1.3.2, April 1983.
4. NUREG-0800, Section 15.0.1.
5. 10 CFR 50.67.
6. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.
7. ASME, Boiler and Pressure Vessel Code.
8. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.
9. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
10. NEDO 33091-A, Revision 2, "Improved BPWS Control Rod Insertion Process," July, 2004.
11. 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*).
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*).

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, ~~and~~ 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, and 15 for AREVA fuel.

~~Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and other fuel design limits described in Reference 1 are not exceeded during abnormal operational transients for operation with LHGRs up to the operating limit LHGR. APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly.~~

~~APLHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during abnormal operational transients (Ref. 7). Flow dependent APLHGR limits are determined using the three dimensional BWR simulator code (Ref. 8) to analyze slow flow runout transients. The flow dependent multiplier,~~ ~~$MAPFAC_f$~~~~, is dependent on the maximum core flow runout capability. The maximum runout flow is~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System.~~

~~Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, $MAPFAC_p$, are also generated. Due to the sensitivity of the transient reponse to initial core flow levels at power levels below those at which turbine stop valve closures and turbine control valve fast closure scram trips are bypassed, both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 25% RTP and the previously mentioned bypass power level. The exposure dependent APLHGR limits are reduced by $MAPFAC_p$ and $MAPFAC_f$ at various operating conditions to ensure that all fuel design criteria are met for normal operation and abnormal operational transients. A complete discussion of the analysis code is provided in Reference 9.~~

GE Fuel

LOCA analyses are ~~then~~ performed to ensure ~~that the above determined~~ 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 exposure and, along with 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 (continued)

LCO

The APLHGR limits specified in the COLR are the result of the fuel design, DBA, and transient analyses. ~~For operation at other than 100% power and 100% recirculation flow conditions, the APLHGR operating limit is determined by multiplying the smaller of the $MAPFAC_p$ and $MAPFAC_f$ factors times the exposure dependent APLHGR limit.~~ With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent limit by an APLHGR correction factor (Ref. 5 and Ref. 10). Cycle specific APLHGR correction factors for single recirculation loop operation are documented in the COLR. APLHGR limits are selected such that no power or flow dependent corrections are required. Additional APLHGR operating limit adjustments may be provided in the COLR supporting other analyzed equipment out-of-service conditions.

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Design calculations (Ref. 4) and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ≤ 25% RTP, the reactor is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NEDE-24011-P-A-13 "General Electric Standard Application for Reactor Fuel," August 1996.
2. FSAR, Chapter 3.
3. FSAR, Chapter 14.
4. FSAR, Appendix N.
5. NEDC-32484P, "Browns Ferry Nuclear Plant Units 1, 2, and 3, SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 2, December 1997.
6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
7. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Units 1, 2, and 3," April 1995.
8. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.
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. EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model," (*as identified in the COLR*).

(continued)

BASES

REFERENCES (continued)

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*).

BASES (continued)

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the abnormal operational transients to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 8, ~~and~~ 10, 11, 12, 13, 14, and 15. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power ~~state (MCPR~~$\cancel{_f}$ ~~and MCPR~~$\cancel{_p}$~~, respectively)~~ to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Reference 8). Flow dependent MCPR ($MCPR_f$) limits are determined by steady state thermal hydraulic methods ~~with key physics response inputs benchmarked using the three dimensional BWR simulator code (Reference 6) to analyze slow flow runout transients~~ using the three dimensional BWR simulator code (Ref. 12) and the multichannel thermal hydraulics code (Ref. 13). The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

Power dependent MCPR limits ($MCPR_p$) are determined by the ~~one-dimensional transient code (Reference 9)~~ three-dimensional BWR simulator code (Ref. 12) and the one-dimensional transient codes (Refs. 14 and 15). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine control valve fast closure scrams are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 25% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of the NRC Policy Statement (Ref. 7).

(continued)

BASES (continued)

LCO	The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. Additional MCPR operating limits supporting analyzed equipment out-of-service conditions are provided in the COLR. The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.
APPLICABILITY	The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at 25% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to 25% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 25% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.
ACTIONS	<u>A.1</u> If any MCPR is outside the required limits, an assumption regarding an initial condition of the design basis transient analyses may not be met. Therefore, prompt action should be taken to restore the MCPR(s) to within the required limits such

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines the ~~value of τ, which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram times of LCO 3.1.4, "Control Rod Scram Times") and Option B (realistic scram times) analyses. The parameter τ must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1 and SR 3.1.4.2 because the effective scram speed distribution may change during the cycle. The 72 hour Completion Time is acceptable due to the relatively minor changes in τ expected during the fuel cycle~~ actual scram speed distribution and compares it with the assumed distribution. The MCPR operating limit is determined based either on the applicable limit associated with scram times of LCO 3.1.4, "Control Rod Scram Times," or the nominal scram times. The scram speed dependent MCPR limits are contained in the COLR. This determination must be performed within 72 hours after each set of control rod scram time tests required by SR 3.1.4.1 and SR 3.1.4.2 because the effective scram speed distribution may change during the cycle. The 72 hour Completion Time is acceptable due to the relatively minor changes in the actual control rod scram speed distribution expected during the fuel cycle.

REFERENCES

1. NUREG-0562, "Fuel Rod Failure As a Consequence of Departure from Nucleate Boiling or Dryout," June 1979.
2. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," August 1996.
3. FSAR, Chapter 3.

(continued)

BASES

REFERENCES (continued)

4. FSAR, Chapter 14.

5. FSAR, Appendix N.

6. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.

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. 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. 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)

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 3).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also includes allowances for short term transient operation above the operating limit to account for abnormal operational transients, plus an allowance for densification power spiking.

LHGR limits are multiplied by the smaller of either the flow-dependent LHGR factor ($LHGRFAC_f$) or the power-dependent LHGR factor ($LHGRFAC_p$) corresponding to the existing core flow and power state to ensure adherence to the fuel mechanical design bases during the limiting transient. $LHGRFAC_f$ is generated to protect the core from slow flow runout transients. A curve is provided based on the maximum credible flow runout transient. $LHGRFAC_p$ is generated to protect the core from plant transients other than core flow increases. LHGRFAC multipliers are provided in the COLR.

The LHGR satisfies Criterion 2 of the NRC Policy Statement (Ref. 4).

LCO

The LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR calculated to cause a 1% fuel cladding plastic strain. The operating limit to accomplish this objective is specified in the COLR.

Additional LHGR operating limits adjustments may be provided in the COLR to support analyzed equipment out-of-service operation.

(continued)

BASES (continued)

APPLICABILITY

The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < 25% RTP, the reactor is operating with a substantial margin to the LHGR limits and, therefore, the Specification is only required when the reactor is operating at ≥ 25% RTP.

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BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY	1. Rod Block Monitor

The RBM is designed to prevent violation of the MCPR SL and the cladding 1% plastic strain fuel design limit that may result from a single control rod withdrawal error (RWE) event. The analytical methods and assumptions used in evaluating the RWE event are summarized in References 3, 13, and 14. A statistical analysis of RWE events was performed to determine the RBM response for both channels for each event. From these responses, the fuel thermal performance as a function of RBM Allowable Value was determined. The Allowable Values are chosen as a function of power level. Based on the specified Allowable Values, operating limits are established.

The RBM Function satisfies Criterion 3 of the NRC Policy Statement (Ref. 10).

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value for the associated power range to ensure that no single instrument failure can preclude a rod block from this Function. The setpoints are calibrated consistent with applicable setpoint methodology (nominal trip setpoint).

Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)	

2. Rod Worth Minimizer

The RWM enforces the banked position withdrawal sequence (BPWS) to ensure that the initial conditions of the CRDA analysis are not violated. The analytical methods and assumptions used in evaluating the CRDA are summarized in References 4, 5, 6, 7 ~~and~~ 12, 13 and 14. The BPWS requires that control rods be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions. Requirements that the control rod sequence is in compliance with the BPWS are specified in LCO 3.1.6, "Rod Pattern Control."

When performing a shutdown of the plant, an optional BPWS control rod sequence (Ref. 12) may be used if the coupling of each withdrawn control rod has been confirmed. The rods may be inserted without the need to stop at intermediate positions. When using the Reference 12 control rod insertion sequence for shutdown, the Rod Worth Minimizer may be programmed to enforce the requirements of the improved BPWS control rod insertion process or bypassed if it is not programmed to reflect the optional BPWS shutdown sequence, as permitted by the Applicability Note for the Rod Worth Minimizer in Table 3.3.2-1.

The RWM Function satisfies Criterion 3 of the NRC Policy Statement (Ref. 10).

Since the RWM is designed to act as a backup to operator control of the rod sequences, only one channel of the RWM is available and required to be OPERABLE (Ref. 7). Special circumstances provided for in the Required Action of LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.6 may necessitate bypassing the RWM to allow continued operation with inoperable control rods, or to allow correction of a control rod pattern not in compliance with the BPWS. The RWM may be bypassed as required by these conditions, but then it must be considered inoperable and the Required Actions of this LCO followed.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 7.5.8.2.3.

2. FSAR, Section 7.16.5.3.1.k.

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

4. NEDE-24011-P-A-US, "General Electrical Standard Application for Reload Fuel," Supplement for United States, (revision specified in the COLR).

5. "Modifications to the Requirements for Control Rod Drop Accident Mitigating Systems," BWR Owners' Group, July 1986.

6. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

7. NRC SER, "Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A," "General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," December 27, 1987.

8. NEDC-30851-P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.

9. GENE-770-06-1, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.

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

11. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October 1995.

(continued)

BASES

REFERENCES (continued)

12. NEDO 33091-A, Revision 2, “Improved BPWS Control Rod Insertion Process,” July 2004.

13. 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*).

14. 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*).

B 3.3 INSTRUMENTATION

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

BASES

BACKGROUND

The EOC-RPT instrumentation initiates a recirculation pump trip (RPT) to reduce the peak reactor pressure and power resulting from turbine trip or generator load rejection transients to provide additional margin to core thermal MCPR Safety Limits (SLs), and LHGR limits.

The need for the additional negative reactivity in excess of that normally inserted on a scram reflects end of cycle reactivity considerations. Flux shapes at the end of cycle are such that the control rods may not be able to ensure that thermal limits are maintained by inserting sufficient negative reactivity during the first few feet of rod travel upon a scram caused by Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure - Low or Turbine Stop Valve (TSV) - Closure. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity at a faster rate than the control rods can add negative reactivity.

The EOC-RPT instrumentation, as shown in Reference 1, is composed of sensors that detect initiation of closure of the TSVs or fast closure of the TCVs, combined with relays, logic circuits, and fast acting circuit breakers that interrupt power from the recirculation pump variable frequency drives (VFD) to each of the recirculation pump motors. When the channels pre-established setpoint is exceeded, the channel output relay actuates, which then outputs an EOC-RPT signal to the trip logic. When the RPT breakers trip open, the recirculation pumps coast down under their own inertia. The EOC-RPT has two identical trip systems, either of which can actuate an RPT.

(continued)

BASES

BACKGROUND (continued)	Each EOC-RPT trip system is a two-out-of-two logic for each Function; thus, either two TSV - Closure or two TCV Fast Closure, Trip Oil Pressure - Low signals are required for a trip system to actuate. If either trip system actuates, both recirculation pumps will trip. There are two EOC-RPT breakers in series per recirculation pump. One trip system trips one of the two EOC-RPT breakers for each recirculation pump, and the second trip system trips the other EOC-RPT breaker for each recirculation pump.
APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY	The TSV - Closure and the TCV Fast Closure, Trip Oil Pressure - Low Functions are designed to trip the recirculation pumps in the event of a turbine trip or generator load rejection to mitigate the increase in neutron flux, heat flux, and reactor pressure, and to increase the margin to the MCPR SL, and LHGR limits. The analytical methods and assumptions used in evaluating the turbine trip and generator load rejection are summarized in References 2, 3, and 4. To mitigate pressurization transient effects, the EOC-RPT must trip the recirculation pumps after initiation of closure movement of either the TSVs or the TCVs. The combined effects of this trip and a scram reduce fuel bundle power more rapidly than a scram alone, resulting in an increased margin to the MCPR SL, and LHGR limits. Alternatively, MCPR limits for an inoperable EOC-RPT, as specified in the COLR, are sufficient to prevent violation of the MCPR Safety Limit, and fuel mechanical limits. The EOC-RPT function is automatically disabled when turbine first stage pressure is < 30% RTP. EOC-RPT instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The specific Applicable Safety Analysis, LCO, and Applicability discussions are listed below on a Function by Function basis.

Alternatively, since this instrumentation protects against a MCPR SL violation, with the instrumentation inoperable, modifications to the MCPR limits (LCO 3.2.2) may be applied to allow this LCO to be met. The MCPR and LHGR penalties~~y~~ for the EOC-RPT inoperable condition ~~is~~are specified in the COLR.

Turbine Stop Valve - Closure

Closure of the TSVs and a main turbine trip result in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, an RPT is initiated on TSV - Closure in anticipation of the transients that would result from closure of these valves. EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL and LHGR limits are ~~is~~ not exceeded during the worst case transient.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Turbine Stop Valve - Closure (continued)

Closure of the TSVs is determined by measuring the position of each valve. There are two separate position signals associated with each stop valve, the signal from each switch being assigned to a separate trip channel. The logic for the TSV - Closure Function is such that two or more TSVs must be closed to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function. To consider this function OPERABLE, bypass of the function must not occur when bypass valves are open. Four channels of TSV - Closure, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TSV - Closure Allowable Value is selected to detect imminent TSV closure.

This protection is required, consistent with the safety analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. Below 30% RTP, the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor (APRM) Fixed Neutron Flux - High Functions of the Reactor Protection System (RPS) are adequate to maintain the necessary margin to the MCPR S~~afety~~ L~~imit~~, and LHGR limits.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

<u>Turbine Control Valve Fast Closure, Trip Oil Pressure - Low</u> (PS-47-142, PS-47-144, PS-47-146, and PS-47-148)

Fast closure of the TCVs during a generator load rejection results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, an RPT is initiated on TCV Fast Closure, Trip Oil Pressure - Low in anticipation of the transients that would result from the closure of these valves. The EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL, and LHGR limits are ~~is~~ not exceeded during the worst case transient.

Fast closure of the TCVs is determined by measuring the electrohydraulic control fluid pressure at each control valve. There is one pressure switch associated with each control valve, and the signal from each switch is assigned to a separate trip channel. The logic for the TCV Fast Closure, Trip Oil Pressure - Low Function is such that two or more TCVs must be closed (pressure switch trips) to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function. To consider this function OPERABLE, bypass of the function must not occur when bypass valves are open. Four channels of TCV Fast Closure, Trip Oil Pressure - Low, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TCV Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure.

(continued)

BASES

ACTIONS (continued)

A.1

With one or more channels inoperable, but with EOC-RPT trip capability maintained (refer to Required Actions B.1 and B.2 Bases), the EOC-RPT System is capable of performing the intended function. However, the reliability and redundancy of the EOC-RPT instrumentation is reduced such that a single failure in the remaining trip system could result in the inability of the EOC-RPT System to perform the intended function. Therefore, only a limited time is allowed to restore compliance with the LCO. Because of the diversity of sensors available to provide trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of an EOC-RPT, 72 hours is provided to restore the inoperable channels (Required Action A.1) or apply the EOC-RPT inoperable MCPR and LHGR limits. Alternately, the inoperable channels may be placed in trip (Required Action A.2) since this would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. As noted, placing the channel in trip with no further restrictions is not allowed if the inoperable channel is the result of an inoperable breaker, since this may not adequately compensate for the inoperable breaker (e.g., the breaker may be inoperable such that it will not open). If it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an RPT, or if the inoperable channel is the result of an inoperable breaker), Condition C must be entered and its Required Actions taken.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

Required Actions B.1 and B.2 are intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in the Function not maintaining EOC-RPT trip capability. A Function is considered to be maintaining EOC-RPT trip capability when sufficient channels are OPERABLE or in trip, such that the EOC-RPT System will generate a trip signal from the given Function on a valid signal and both recirculation pumps can be tripped. Alternately, Required Action B.2 requires the MCPR and LHGR limits for inoperable EOC-RPT, as specified in the COLR, to be applied. This also restores the margin to MCPR and LHGR limits assumed in the safety analysis.

The 2 hour Completion Time is sufficient time for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of the EOC-RPT instrumentation during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR or LHGR violation.

C.1

With any Required Action and associated Completion Time not met, THERMAL POWER must be reduced to < 30% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reduce THERMAL POWER to < 30% RTP from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES	The Main Turbine Bypass System is assumed to function during abnormal operational transients (e.g., the feedwater controller failure-maximum demand event), as discussed in the FSAR, Section 14.5.1.1 (Ref. 2). Opening the bypass valves during the event mitigates the increase in reactor vessel pressure, which affects the MCPR during the event. An inoperable Main Turbine Bypass System may result in APLHGR, ~~and~~ MCPR, and LHGR penalties. The Main Turbine Bypass System satisfies Criterion 3 of the NRC Policy Statement (Ref. 3).
LCO	The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the ~~Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)") and the MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") may be applied to allow this LCO to be met. The APLHGR and MCPR~~ APLHGR limits, MCPR Safety Limit, and LHGR limits are not exceeded. With the Main Turbine Bypass System inoperable, modifications to the APLHGR limits (LCO 3.2.1, “AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)”), the MCPR limits (LCO 3.2.2, “MINIMUM CRITICAL POWER RATIO (MCPR)”), and LHGR limits (LCO 3.2.3, “LINEAR HEAT GENERATION RATE (LHGR)”) may be applied to allow this LCO to be met. The APLHGR, MCPR, and LHGR limits for the inoperable Main Turbine Bypass System are specified in the COLR. An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analysis (Ref. 2).

(continued)

BASES (continued)

APPLICABILITY	The Main Turbine Bypass System is required to be OPERABLE at ≥ 25% RTP to ensure that the fuel cladding integrity Safety Limit is not violated during abnormal operational transients. As discussed in the Bases for LCO 3.2.1 and LCO 3.2.2, sufficient margin to these limits exists at < 25% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), or the APLHGR,~~ and~~ MCPR, and LHGR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the APLHGR,~~ and~~ MCPR, and LHGR limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the APLHGR,~~ and~~ MCPR, and LHGR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 25% RTP. As discussed in the Applicability section, operation at < 25% RTP results in sufficient margin to the required limits, and the Main

(continued)

B 3.10 SPECIAL OPERATIONS

B 3.10.7 Control Rod Testing - Operating

BASES

BACKGROUND	The purpose of this Special Operations LCO is to permit control rod testing, while in MODES 1 and 2, by imposing certain administrative controls. Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), such that only the specified control rod sequences and relative positions required by LCO 3.1.6, "Rod Pattern Control," are allowed over the operating range from all control rods inserted to the low power setpoint (LPSP) of the RWM. The sequences effectively limit the potential amount and rate of reactivity increase that could occur during a control rod drop accident (CRDA). During these conditions, control rod testing is sometimes required that may result in control rod patterns not in compliance with the prescribed sequences of LCO 3.1.6. These tests include SDM demonstrations, control rod scram time testing, and control rod friction testing. This Special Operations LCO provides the necessary exemption to the requirements of LCO 3.1.6 and provides additional administrative controls to allow the deviations in such tests from the prescribed sequences in LCO 3.1.6.
APPLICABLE SAFETY ANALYSES	The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1, ~~and~~ 2, 3, and 4. CRDA analyses assume the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analyses. The RWM provides backup to operator control of the withdrawal sequences to ensure the initial conditions of the CRDA analyses are not

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

violated. For special sequences developed for control rod testing, the initial control rod patterns assumed in the safety analysis of References 1, ~~and~~ 2, 3, and 4 may not be preserved. Therefore, special CRDA analyses may be required to demonstrate that these special sequences will not result in unacceptable consequences, should a CRDA occur during the testing. These analyses, performed in accordance with an NRC approved methodology, are dependent on the specific test being performed.

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of the NRC Policy Statement apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. Control rod testing may be performed in compliance with the prescribed sequences of LCO 3.1.6, and during these tests, no exceptions to the requirements of LCO 3.1.6 are necessary. For testing performed with a sequence not in compliance with LCO 3.1.6, the requirements of LCO 3.1.6 may be suspended, provided additional administrative controls are placed on the test to ensure that the assumptions of the special safety analysis for the test sequence are satisfied. Assurances that the test sequence is followed can be provided by either programming the test sequence into the RWM, with conformance verified as specified in SR 3.3.2.1.8 and allowing the RWM to monitor

(continued)

BASES

LCO (continued)

control rod withdrawal and provide appropriate control rod blocks if necessary, or by verifying conformance to the approved test sequence by a second licensed operator or other qualified member of the technical staff (i.e., personnel trained in accordance with an approved training program for this test). These controls are consistent with those normally applied to operation in the startup range as defined in the SRs and ACTIONS of LCO 3.3.2.1, "Control Rod Block Instrumentation."

APPLICABILITY

Control rod testing, while in MODES 1 and 2, with THERMAL POWER greater than 10% RTP, is adequately controlled by the existing LCOs on power distribution limits and control rod block instrumentation. Control rod movement during these conditions is not restricted to prescribed sequences and can be performed within the constraints of LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," and LCO 3.3.2.1. With THERMAL POWER less than or equal to 10% RTP, the provisions of this Special Operations LCO are necessary to perform special tests that are not in conformance with the prescribed sequences of LCO 3.1.6.

While in MODES 3 and 4, control rod withdrawal is only allowed if performed in accordance with Special Operations LCO 3.10.3, "Single Control Rod Withdrawal - Hot Shutdown," or Special Operations LCO 3.10.4, "Single Control Rod Withdrawal - Cold Shutdown," which provide adequate controls to ensure that the assumptions of the safety analyses of References 1, ~~and~~ 2, 3, and 4 are

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.10.7.2

When the RWM provides conformance to the special test sequence, the test sequence must be verified to be correctly loaded into the RWM prior to control rod movement. This Surveillance demonstrates compliance with SR 3.3.2.1.8, thereby demonstrating that the RWM is OPERABLE. A Note has been added to indicate that this Surveillance does not need to be performed if SR 3.10.7.1 is satisfied.

REFERENCES

1. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," August 1996.
2. Letter from T. Pickens (BWROG) to G. C. Lainas (NRC) "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," August 15, 1986.
3. 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*).
4. 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*).

BASES (continued)

APPLICABLE SAFETY ANALYSES

Prevention and mitigation of unacceptable reactivity excursions during control rod withdrawal, with the reactor mode switch in the startup/hot standby position while in MODE 5, is provided by the intermediate range monitor (IRM) neutron flux scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), and control rod block instrumentation (LCO 3.3.2.1, "Control Rod Block Instrumentation"). The limiting reactivity excursion during startup conditions while in MODE 5 is the control rod drop accident (CRDA).

CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. For SDM tests performed within these defined sequences, the analyses of References 1, ~~and~~ 2, 3, and 4 are applicable. However, for some sequences developed for the SDM testing, the control rod patterns assumed in the safety analyses of References 1, ~~and~~ 2, 3, and 4 may not be met. Therefore, special CRDA analyses, performed in accordance with an NRC approved methodology, may be required to demonstrate the SDM test sequence will not result in unacceptable consequences should a CRDA occur during the testing. For the purpose of this test, the protection provided by the normally required MODE 5 applicable LCOs, in addition to the requirements of this LCO, will maintain normal test operations as well as postulated accidents within the bounds of the appropriate safety analyses (Refs. 1, ~~and~~ 2, 3, and 4). In addition to the added requirements for the RWM, APRM, and control rod coupling, the notch out mode is specified for out of sequence withdrawals. Requiring the notch out mode limits withdrawal steps to a single notch, which limits inserted reactivity, and allows adequate monitoring of changes in neutron flux, which may occur during the test.

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SURVEILLANCE REQUIREMENTS (continued)

SR 3.10.8.5

Coupling verification is performed to ensure the control rod is connected to the control rod drive mechanism and will perform its intended function when necessary. The verification is required to be performed any time a control rod is withdrawn to the "full out" notch position, or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved as well as operating experience related to uncoupling events.

SR 3.10.8.6

CRD charging water header pressure verification is performed to ensure the motive force is available to scram the control rods in the event of a scram signal. Since the reactor is depressurized in MODE 5, there is insufficient reactor pressure to scram the control rods. Verification of charging water pressure ensures that if a scram is required, capability for rapid control rod insertion would exist. The minimum pressure of 940 psig, which is well below the expected pressure of approximately 1100 psig, ensures sufficient pressure for rapid control rod insertion. The 7 day Frequency has been shown to be acceptable through operating experience and takes into account indications available in the control room.

REFERENCES

1. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," August 1996.
2. Letter from T. Pickens (BWROG) to G. C. Lainas, NRC, "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," August 15, 1986.

(continued)

BASES

REFERENCES (continued)

3. 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*).

4. 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*).

ATTACHMENT 5

Browns Ferry Nuclear Plant (BFN)
Unit 1

Technical Specifications (TS) Change 473

AREVA Fuel Transition

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.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

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

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

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

While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross, rather than

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity

The SPCB critical power correlation is used for both AREVA and coresident 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 ATRIUM-10 and GE14 fuel types. 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 due only to elevation effects from liquid only in the channel, core bypass region, and annulus at zero power, zero flow is approximately 4.5 psi. At all operating conditions, this pressure differential is maintained by the bypass region of the core and the annulus region of the vessel. The elevation head provided by the annulus produces natural circulation flow conditions which have balancing pressure head and loss terms inside the core shroud. This natural circulation principle maintains a core plenum to plenum pressure drop of about 4.5 to 5 psid along the natural circulation flow line of the P/F operating map. In the range of power levels of interest, approaching 25% of rated power below which thermal margin monitoring is not required, the pressure drop and density head terms tradeoff for power changes such that natural circulation flow is nearly independent of reactor power. This characteristic is represented by the nearly vertical portion of the natural circulation line on the P/F operating map. Analysis has shown that the hot channel flow rate is >28,000 lb_m/hr (>0.23 x 10^6 lb_m/hr-ft^2) in the region of operation with power ~25% and core pressure drop of about 4.5 to 5 psid. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at 28,000 lb_m/hr is approximately 3 MW_t. With the design peaking factors, this corresponds to a core thermal power of more than 50%.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

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 800 psia. 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 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 10% of rated flow.

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

SDM requirements are specified to ensure:

a. The reactor can be made subcritical from all operating conditions and transients and Design Basis Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

These requirements are satisfied by the control rods, as described in GDC 26 (Ref. 1), which can compensate for the reactivity effects of the fuel and water temperature changes experienced during all operating conditions.

APPLICABLE SAFETY ANALYSES

The control rod drop accident (CRDA) analysis (Refs. 2, 3, 9, and 10) assumes the core is subcritical with the highest worth control rod withdrawn. Typically, the first control rod withdrawn has a very high reactivity worth and, should the core be critical during the withdrawal of the first control rod, the consequences of a CRDA could exceed the fuel damage limits for a CRDA (see Bases for LCO 3.1.6, "Rod Pattern Control"). Also, SDM is assumed as an initial condition for the control rod removal error during refueling (Ref. 4) and fuel assembly insertion error during refueling (Ref. 5) accidents. The analysis of these reactivity insertion events assumes the refueling interlocks are OPERABLE when the reactor is in the refueling mode of operation. These interlocks prevent the withdrawal of more

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR, Section 14.6.2.
3. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," Section S.2.2.3.1, August 1996.
4. FSAR, Section 14.5.3.3.
5. FSAR, Section 14.5.3.4.
6. FSAR, Section 3.6.5.2.
7. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1, August 1996.
8. NRC 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
9. 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*).
10. 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*).

BASES (continued)

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the control rod scram function are presented in References 2, 3, and 4. The Design Basis Accident (DBA) and transient analyses assume that all of the control rods scram at a specified insertion rate. The resulting negative scram reactivity forms the basis for the determination of plant thermal limits (e.g., the MCPR). Other distributions of scram times (e.g., several control rods scramming slower than the average time with several control rods scramming faster than the average time) can also provide sufficient scram reactivity. Surveillance of each individual control rod's scram time ensures the scram reactivity assumed in the DBA and transient analyses can be met.

The scram function of the CRD System protects the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)", and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), which ensure that no fuel damage will occur if these limits are not exceeded. Above 800 psig, the scram function is designed to insert negative reactivity at a rate fast enough to prevent the actual MCPR from becoming less than the MCPR SL, during the analyzed limiting power transient. Below 800 psig, the scram function is assumed to perform during the control rod drop accident (Refs. 5, 8, and 9) and, therefore, also provides protection against violating fuel damage limits during reactivity insertion accidents (see Bases for LCO 3.1.6, "Rod Pattern Control"). For the reactor vessel overpressure protection analysis, the scram function, along with the safety/relief valves, ensure that the peak vessel pressure is maintained within the applicable ASME Code limits.

Control rod scram times satisfy Criterion 3 of the NRC Policy Statement (Ref. 7).

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. FSAR, Section 3.4.6.
3. FSAR, Section 14.5.
4. FSAR, Section 14.6.
5. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1, August 1996.
6. Letter from R. F. Janecek (BWROG) to R. W. Starostecki (NRC), "BWR Owners Group Revised Reactivity Control System Technical Specifications," BWROG-8754, September 17, 1987.
7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
8. 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*).
9. 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*).

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND	Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to 10% RTP. The sequences limit the potential amount of reactivity addition that could occur in the event of a Control Rod Drop Accident (CRDA). This Specification assures that the control rod patterns are consistent with the assumptions of the CRDA analyses of References 1, 2, 11, and 12.
APPLICABLE SAFETY ANALYSES	The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1, 2, 11, and 12. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RWM (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated. Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage which could result in the undue release of radioactivity. Since the failure consequences for UO_2 have been shown to be insignificant below fuel energy

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

depositions of 300 cal/gm (Ref. 3), the fuel damage limit of 280 cal/gm provides a margin of safety from significant core damage which would result in release of radioactivity (Refs. 4 and 5). Generic evaluations (Refs. 1, 6, and 11) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required ASME Code limits (Ref. 7) and the calculated offsite doses will be well within the required limits (Ref. 5).

Control rod patterns analyzed in References 1, 11, and 12, follow the banked position withdrawal sequence (BPWS). The BPWS is applicable from the condition of all control rods fully inserted to 10% RTP (Ref. 2). For the BPWS, the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are established to minimize the maximum incremental control rod worth without being overly restrictive during normal plant operation. Analyses are performed using the Reference 11 methodology demonstrating the 280 cal/gm fuel damage limit will not be violated during a CRDA while following the BPWS mode of operation. The evaluation provided by the generic BPWS analysis (Ref. 8) allows a limited number (i.e., eight) and corresponding distribution of fully inserted, inoperable control rods, that are not in compliance with the sequence.

When performing a shutdown of the plant, an optional BPWS control rod sequence (Ref. 10) may be used provided that all withdrawn control rods have been confirmed to be coupled. The rods may be inserted without the need to stop at intermediate positions since the possibility of a CRDA is eliminated by the confirmation that withdrawn control rods are coupled. When using the Reference 10 control rod sequence for shutdown, the Rod Worth Minimizer may be reprogrammed

(continued)

BASES (continued)

REFERENCES

1. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," Section 2.2.3.1, August 1996.

2. Letter from T. Pickens (BWROG) to G. C. Lainas (NRC), Amendment 17 to General Electric Licensing Topical Report, NEDE-24011-P-A, August 15, 1986.

3. NUREG-0979, Section 4.2.1.3.2, April 1983.

4. NUREG-0800, Section 15.0.1.

5. 10 CFR 50.67.

6. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.

7. ASME, Boiler and Pressure Vessel Code.

8. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

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

10. NEDO 33091-A, Revision 2, "Improved BPWS Control Rod Insertion Process," July, 2004.

11. 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*).

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*).

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, and 15 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.

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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

AREVA Fuel

For AREVA fuel, the APLHGR limits are developed as a function of exposure and, along with 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 (Reference 11). The multiplier is documented in the COLR.

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

(continued)

BASES (continued)

LCO

The APLHGR limits specified in the COLR are the result of the fuel design, DBA, and transient analyses. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent limit by an APLHGR correction factor (Ref. 5 and Ref. 10). Cycle specific APLHGR correction factors for single recirculation loop operation are documented in the COLR. APLHGR limits are selected such that no power or flow dependent corrections are required. Additional APLHGR operating limit adjustments may be provided in the COLR supporting other analyzed equipment out-of-service conditions.

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Design calculations (Ref. 4) and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ≤ 25% RTP, the reactor is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

operation. The 12 hour allowance after THERMAL POWER $\geq$ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NEDE-24011-P-A-13 "General Electric Standard Application for Reactor Fuel," August 1996.
2. FSAR, Chapter 3.
3. FSAR, Chapter 14.
4. FSAR, Appendix N.
5. NEDC-32484P, "Browns Ferry Nuclear Plant Units 1, 2, and 3, SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 2, December 1997.
6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
7. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Units 1, 2, and 3," April 1995.
8. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.
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. EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model," (*as identified in the COLR*).

(continued)

BASES

REFERENCES (continued)

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*).

BASES (continued)

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the abnormal operational transients to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 8, 10, 11, 12, 13, 14, and 15. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Reference 8). Flow dependent MCPR ($MCPR_f$) limits are determined by steady state thermal hydraulic methods using the three dimensional BWR simulator code (Ref. 12) and the multichannel thermal hydraulics code (Ref. 13). The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

Power dependent MCPR limits ($MCPR_p$) are determined by the three-dimensional BWR simulator code (Ref. 12) and the one-dimensional transient codes (Refs. 14 and 15). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine control valve fast closure scrams are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 25% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of the NRC Policy Statement (Ref. 7).

(continued)

BASES (continued)

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. Additional MCPR operating limits supporting analyzed equipment out-of-service conditions are provided in the COLR. The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at 25% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to 25% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 25% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

ACTIONS

A.1

If any MCPR is outside the required limits, an assumption regarding an initial condition of the design basis transient analyses may not be met. Therefore, prompt action should be taken to restore the MCPR(s) to within the required limits such

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines the actual scram speed distribution and compares it with the assumed distribution. The MCPR operating limit is determined based either on the applicable limit associated with scram times of LCO 3.1.4, "Control Rod Scram Times," or the nominal scram times. The scram speed dependent MCPR limits are contained in the COLR. This determination must be performed within 72 hours after each set of control rod scram time tests required by SR 3.1.4.1 and SR 3.1.4.2 because the effective scram speed distribution may change during the cycle. The 72 hour Completion Time is acceptable due to the relatively minor changes in the actual control rod scram speed distribution expected during the fuel cycle.

REFERENCES

1. NUREG-0562, "Fuel Rod Failure As a Consequence of Departure from Nucleate Boiling or Dryout," June 1979.
2. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," August 1996.
3. FSAR, Chapter 3.
4. FSAR, Chapter 14.
5. FSAR, Appendix N.
6. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.
7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

(continued)

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. 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)	A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 3). Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also includes allowances for short term transient operation above the operating limit to account for abnormal operational transients, plus an allowance for densification power spiking. LHGR limits are multiplied by the smaller of either the flow-dependent LHGR factor ($LHGRFAC_f$) or the power-dependent LHGR factor ($LHGRFAC_p$) corresponding to the existing core flow and power state to ensure adherence to the fuel mechanical design bases during the limiting transient. $LHGRFAC_f$ is generated to protect the core from slow flow runout transients. A curve is provided based on the maximum credible flow runout transient. $LHGRFAC_p$ is generated to protect the core from plant transients other than core flow increases. LHGRFAC multipliers are provided in the COLR. The LHGR satisfies Criterion 2 of the NRC Policy Statement (Ref. 4).
LCO	The LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR calculated to cause a 1% fuel cladding plastic strain. The operating limit to accomplish this objective is specified in the COLR. Additional LHGR operating limits adjustments may be provided in the COLR to support analyzed equipment out-of-service operation.

(continued)

BASES (continued)

APPLICABILITY

The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < 25% RTP, the reactor is operating with a substantial margin to the LHGR limits and, therefore, the Specification is only required when the reactor is operating at ≥ 25% RTP.

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor

The RBM is designed to prevent violation of the MCPR SL and the cladding 1% plastic strain fuel design limit that may result from a single control rod withdrawal error (RWE) event. The analytical methods and assumptions used in evaluating the RWE event are summarized in References 3, 13, and 14. A statistical analysis of RWE events was performed to determine the RBM response for both channels for each event. From these responses, the fuel thermal performance as a function of RBM Allowable Value was determined. The Allowable Values are chosen as a function of power level. Based on the specified Allowable Values, operating limits are established.

The RBM Function satisfies Criterion 3 of the NRC Policy Statement (Ref. 10).

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value for the associated power range to ensure that no single instrument failure can preclude a rod block from this Function. The setpoints are calibrated consistent with applicable setpoint methodology (nominal trip setpoint).

Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2. Rod Worth Minimizer

The RWM enforces the banked position withdrawal sequence (BPWS) to ensure that the initial conditions of the CRD analysis are not violated. The analytical methods and assumptions used in evaluating the CRDA are summarized in References 4, 5, 6, 7, 12, 13 and 14. The BPWS requires that control rods be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions. Requirements that the control rod sequence is in compliance with the BPWS are specified in LCO 3.1.6, "Rod Pattern Control."

When performing a shutdown of the plant, an optional BPWS control rod sequence (Ref. 12) may be used if the coupling of each withdrawn control rod has been confirmed. The rods may be inserted without the need to stop at intermediate positions. When using the Reference 12 control rod insertion sequence for shutdown, the Rod Worth Minimizer may be programmed to enforce the requirements of the improved BPWS control rod insertion process or bypassed if it is not programmed to reflect the optional BPWS shutdown sequence, as permitted by the Applicability Note for the Rod Worth Minimizer in Table 3.3.2-1.

The RWM Function satisfies Criterion 3 of the NRC Policy Statement (Ref. 10).

Since the RWM is designed to act as a backup to operator control of the rod sequences, only one channel of the RWM is available and required to be OPERABLE (Ref. 7). Special circumstances provided for in the Required Action of LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.6 may necessitate bypassing the RWM to allow continued operation with inoperable control rods, or to allow correction of a control rod pattern not in compliance with the BPWS. The RWM may be bypassed as required by these conditions, but then it must be considered inoperable and the Required Actions of this LCO followed.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 7.5.8.2.3.
2. FSAR, Section 7.16.5.3.1.k.
3. NEDC-32433P, "Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Unit 1, 2 and 3," April 1995.
4. NEDE-24011-P-A-US, "General Electrical Standard Application for Reload Fuel," Supplement for United States, (revision specified in the COLR).
5. "Modifications to the Requirements for Control Rod Drop Accident Mitigating Systems," BWR Owners' Group, July 1986.
6. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.
7. NRC SER, "Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A," "General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," December 27, 1987.
8. NEDC-30851-P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.
9. GENE-770-06-1, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.
10. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
11. NEDC-32410P-A, “Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function,” October 1995.

(continued)

BASES

REFERENCES (continued)

12. NEDO 33091-A, Revision 2, "Improved BPWS Control Rod Insertion Process," July 2004.

13. 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*).

14. 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*).

B 3.3 INSTRUMENTATION

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

BASES

BACKGROUND

The EOC-RPT instrumentation initiates a recirculation pump trip (RPT) to reduce the peak reactor pressure and power resulting from turbine trip or generator load rejection transients to provide additional margin to core thermal MCPR Safety Limits (SLs), and LHGR limits.

The need for the additional negative reactivity in excess of that normally inserted on a scram reflects end of cycle reactivity considerations. Flux shapes at the end of cycle are such that the control rods may not be able to ensure that thermal limits are maintained by inserting sufficient negative reactivity during the first few feet of rod travel upon a scram caused by Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure - Low or Turbine Stop Valve (TSV) - Closure. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity at a faster rate than the control rods can add negative reactivity.

The EOC-RPT instrumentation, as shown in Reference 1, is composed of sensors that detect initiation of closure of the TSVs or fast closure of the TCVs, combined with relays, logic circuits, and fast acting circuit breakers that interrupt power from the recirculation pump variable frequency drives (VFD) to each of the recirculation pump motors. When the channels pre-established setpoint is exceeded, the channel output relay actuates, which then outputs an EOC-RPT signal to the trip logic. When the RPT breakers trip open, the recirculation pumps coast down under their own inertia. The EOC-RPT has two identical trip systems, either of which can actuate an RPT.

(continued)

BASES

BACKGROUND (continued)	Each EOC-RPT trip system is a two-out-of-two logic for each Function; thus, either two TSV - Closure or two TCV Fast Closure, Trip Oil Pressure - Low signals are required for a trip system to actuate. If either trip system actuates, both recirculation pumps will trip. There are two EOC-RPT breakers in series per recirculation pump. One trip system trips one of the two EOC-RPT breakers for each recirculation pump, and the second trip system trips the other EOC-RPT breaker for each recirculation pump.
APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY	The TSV - Closure and the TCV Fast Closure, Trip Oil Pressure - Low Functions are designed to trip the recirculation pumps in the event of a turbine trip or generator load rejection to mitigate the increase in neutron flux, heat flux, and reactor pressure, and to increase the margin to the MCPR SL, and LHGR limits. The analytical methods and assumptions used in evaluating the turbine trip and generator load rejection are summarized in References 2, 3, and 4. To mitigate pressurization transient effects, the EOC-RPT must trip the recirculation pumps after initiation of closure movement of either the TSVs or the TCVs. The combined effects of this trip and a scram reduce fuel bundle power more rapidly than a scram alone, resulting in an increased margin to the MCPR SL, and LHGR limits. Alternatively, MCPR limits for an inoperable EOC-RPT, as specified in the COLR, are sufficient to prevent violation of the MCPR Safety Limit, and fuel mechanical limits. The EOC-RPT function is automatically disabled when turbine first stage pressure is < 30% RTP. EOC-RPT instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The specific Applicable Safety Analysis, LCO, and Applicability discussions are listed below on a Function by Function basis.

Alternatively, since this instrumentation protects against a MCPR SL violation, with the instrumentation inoperable, modifications to the MCPR limits (LCO 3.2.2) may be applied to allow this LCO to be met. The MCPR and LHGR penalties for the EOC-RPT inoperable condition are specified in the COLR.

Turbine Stop Valve - Closure

Closure of the TSVs and a main turbine trip result in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, an RPT is initiated on TSV - Closure in anticipation of the transients that would result from closure of these valves. EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL and LHGR limits are not exceeded during the worst case transient.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Turbine Stop Valve - Closure (continued)

Closure of the TSVs is determined by measuring the position of each valve. There are two separate position signals associated with each stop valve, the signal from each switch being assigned to a separate trip channel. The logic for the TSV - Closure Function is such that two or more TSVs must be closed to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function. To consider this function OPERABLE, bypass of the function must not occur when bypass valves are open. Four channels of TSV - Closure, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TSV - Closure Allowable Value is selected to detect imminent TSV closure.

This protection is required, consistent with the safety analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. Below 30% RTP, the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor (APRM) Fixed Neutron Flux - High Functions of the Reactor Protection System (RPS) are adequate to maintain the necessary margin to the MCPR SL, and LHGR limits.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)	<u>Turbine Control Valve Fast Closure, Trip Oil Pressure - Low</u> (PS-47-142, PS-47-144, PS-47-146, and PS-47-148)

Fast closure of the TCVs during a generator load rejection results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, an RPT is initiated on TCV Fast Closure, Trip Oil Pressure - Low in anticipation of the transients that would result from the closure of these valves. The EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL, and LHGR limits are not exceeded during the worst case transient.

Fast closure of the TCVs is determined by measuring the electrohydraulic control fluid pressure at each control valve. There is one pressure switch associated with each control valve, and the signal from each switch is assigned to a separate trip channel. The logic for the TCV Fast Closure, Trip Oil Pressure - Low Function is such that two or more TCVs must be closed (pressure switch trips) to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function. To consider this function OPERABLE, bypass of the function must not occur when bypass valves are open. Four channels of TCV Fast Closure, Trip Oil Pressure - Low, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TCV Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure.

(continued)

BASES

ACTIONS (continued)

A.1

With one or more channels inoperable, but with EOC-RPT trip capability maintained (refer to Required Actions B.1 and B.2 Bases), the EOC-RPT System is capable of performing the intended function. However, the reliability and redundancy of the EOC-RPT instrumentation is reduced such that a single failure in the remaining trip system could result in the inability of the EOC-RPT System to perform the intended function. Therefore, only a limited time is allowed to restore compliance with the LCO. Because of the diversity of sensors available to provide trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of an EOC-RPT, 72 hours is provided to restore the inoperable channels (Required Action A.1) or apply the EOC-RPT inoperable MCPR and LHGR limits. Alternately, the inoperable channels may be placed in trip (Required Action A.2) since this would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. As noted, placing the channel in trip with no further restrictions is not allowed if the inoperable channel is the result of an inoperable breaker, since this may not adequately compensate for the inoperable breaker (e.g., the breaker may be inoperable such that it will not open). If it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an RPT, or if the inoperable channel is the result of an inoperable breaker), Condition C must be entered and its Required Actions taken.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

Required Actions B.1 and B.2 are intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in the Function not maintaining EOC-RPT trip capability. A Function is considered to be maintaining EOC-RPT trip capability when sufficient channels are OPERABLE or in trip, such that the EOC-RPT System will generate a trip signal from the given Function on a valid signal and both recirculation pumps can be tripped. Alternately, Required Action B.2 requires the MCPR and LHGR limits for inoperable EOC-RPT, as specified in the COLR, to be applied. This also restores the margin to MCPR and LHGR limits assumed in the safety analysis.

The 2 hour Completion Time is sufficient time for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of the EOC-RPT instrumentation during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR or LHGR violation.

C.1

With any Required Action and associated Completion Time not met, THERMAL POWER must be reduced to < 30% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reduce THERMAL POWER to < 30% RTP from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System is assumed to function during abnormal operational transients (e.g., the feedwater controller failure-maximum demand event), as discussed in the FSAR, Section 14.5.1.1 (Ref. 2). Opening the bypass valves during the event mitigates the increase in reactor vessel pressure, which affects the MCPR during the event. An inoperable Main Turbine Bypass System may result in APLHGR, MCPR, and LHGR penalties.

The Main Turbine Bypass System satisfies Criterion 3 of the NRC Policy Statement (Ref. 3).

LCO

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the APLHGR limits, MCPR Safety Limit, and LHGR limits are not exceeded. With the Main Turbine Bypass System inoperable, modifications to the APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), the MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), and LHGR limits (LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)") may be applied to allow this LCO to be met. The APLHGR, MCPR, and LHGR limits for the inoperable Main Turbine Bypass System are specified in the COLR. An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analysis (Ref. 2).

BASES (continued)

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ 25% RTP to ensure that the fuel cladding integrity Safety Limit is not violated during abnormal operational transients. As discussed in the Bases for LCO 3.2.1 and LCO 3.2.2, sufficient margin to these limits exists at < 25% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), or the APLHGR, MCPR, and LHGR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the APLHGR, MCPR, and LHGR limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the APLHGR, MCPR, and LHGR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 25% RTP. As discussed in the Applicability section, operation at < 25% RTP results in sufficient margin to the required limits, and the Main

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B 3.10 SPECIAL OPERATIONS

B 3.10.7 Control Rod Testing - Operating

BASES

BACKGROUND	The purpose of this Special Operations LCO is to permit control rod testing, while in MODES 1 and 2, by imposing certain administrative controls. Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), such that only the specified control rod sequences and relative positions required by LCO 3.1.6, "Rod Pattern Control," are allowed over the operating range from all control rods inserted to the low power setpoint (LPSP) of the RWM. The sequences effectively limit the potential amount and rate of reactivity increase that could occur during a control rod drop accident (CRDA). During these conditions, control rod testing is sometimes required that may result in control rod patterns not in compliance with the prescribed sequences of LCO 3.1.6. These tests include SDM demonstrations, control rod scram time testing, and control rod friction testing. This Special Operations LCO provides the necessary exemption to the requirements of LCO 3.1.6 and provides additional administrative controls to allow the deviations in such tests from the prescribed sequences in LCO 3.1.6.
APPLICABLE SAFETY ANALYSES	The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1, 2, 3, and 4. CRDA analyses assume the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analyses. The RWM provides backup to operator control of the withdrawal sequences to ensure the initial conditions of the CRDA analyses are not

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	violated. For special sequences developed for control rod testing, the initial control rod patterns assumed in the safety analysis of References 1, 2, 3, and 4 may not be preserved. Therefore, special CRDA analyses may be required to demonstrate that these special sequences will not result in unacceptable consequences, should a CRDA occur during the testing. These analyses, performed in accordance with an NRC approved methodology, are dependent on the specific test being performed. As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of the NRC Policy Statement apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.
LCO	As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. Control rod testing may be performed in compliance with the prescribed sequences of LCO 3.1.6, and during these tests, no exceptions to the requirements of LCO 3.1.6 are necessary. For testing performed with a sequence not in compliance with LCO 3.1.6, the requirements of LCO 3.1.6 may be suspended, provided additional administrative controls are placed on the test to ensure that the assumptions of the special safety analysis for the test sequence are satisfied. Assurances that the test sequence is followed can be provided by either programming the test sequence into the RWM, with conformance verified as specified in SR 3.3.2.1.8 and allowing the RWM to monitor

(continued)

BASES

LCO (continued)	control rod withdrawal and provide appropriate control rod blocks if necessary, or by verifying conformance to the approved test sequence by a second licensed operator or other qualified member of the technical staff (i.e., personnel trained in accordance with an approved training program for this test). These controls are consistent with those normally applied to operation in the startup range as defined in the SRs and ACTIONS of LCO 3.3.2.1, "Control Rod Block Instrumentation."
APPLICABILITY	Control rod testing, while in MODES 1 and 2, with THERMAL POWER greater than 10% RTP, is adequately controlled by the existing LCOs on power distribution limits and control rod block instrumentation. Control rod movement during these conditions is not restricted to prescribed sequences and can be performed within the constraints of LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," and LCO 3.3.2.1. With THERMAL POWER less than or equal to 10% RTP, the provisions of this Special Operations LCO are necessary to perform special tests that are not in conformance with the prescribed sequences of LCO 3.1.6. While in MODES 3 and 4, control rod withdrawal is only allowed if performed in accordance with Special Operations LCO 3.10.3, "Single Control Rod Withdrawal - Hot Shutdown," or Special Operations LCO 3.10.4, "Single Control Rod Withdrawal - Cold Shutdown," which provide adequate controls to ensure that the assumptions of the safety analyses of References 1, 2, 3, and 4 are

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.10.7.2

When the RWM provides conformance to the special test sequence, the test sequence must be verified to be correctly loaded into the RWM prior to control rod movement. This Surveillance demonstrates compliance with SR 3.3.2.1.8, thereby demonstrating that the RWM is OPERABLE. A Note has been added to indicate that this Surveillance does not need to be performed if SR 3.10.7.1 is satisfied.

REFERENCES

1. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," August 1996.

2. Letter from T. Pickens (BWROG) to G. C. Lainas (NRC) "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," August 15, 1986.

3. 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*).

4. 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*).

BASES (continued)

APPLICABLE SAFETY ANALYSES

Prevention and mitigation of unacceptable reactivity excursions during control rod withdrawal, with the reactor mode switch in the startup/hot standby position while in MODE 5, is provided by the intermediate range monitor (IRM) neutron flux scram (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), and control rod block instrumentation (LCO 3.3.2.1, "Control Rod Block Instrumentation"). The limiting reactivity excursion during startup conditions while in MODE 5 is the control rod drop accident (CRDA).

CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. For SDM tests performed within these defined sequences, the analyses of References 1, 2, 3, and 4 are applicable. However, for some sequences developed for the SDM testing, the control rod patterns assumed in the safety analyses of References 1, 2, 3, and 4 may not be met. Therefore, special CRDA analyses, performed in accordance with an NRC approved methodology, may be required to demonstrate the SDM test sequence will not result in unacceptable consequences should a CRDA occur during the testing. For the purpose of this test, the protection provided by the normally required MODE 5 applicable LCOs, in addition to the requirements of this LCO, will maintain normal test operations as well as postulated accidents within the bounds of the appropriate safety analyses (Refs. 1, 2, 3, and 4). In addition to the added requirements for the RWM, APRM, and control rod coupling, the notch out mode is specified for out of sequence withdrawals. Requiring the notch out mode limits withdrawal steps to a single notch, which limits inserted reactivity, and allows adequate monitoring of changes in neutron flux, which may occur during the test.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.10.8.5

Coupling verification is performed to ensure the control rod is connected to the control rod drive mechanism and will perform its intended function when necessary. The verification is required to be performed any time a control rod is withdrawn to the "full out" notch position, or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved as well as operating experience related to uncoupling events.

SR 3.10.8.6

CRD charging water header pressure verification is performed to ensure the motive force is available to scram the control rods in the event of a scram signal. Since the reactor is depressurized in MODE 5, there is insufficient reactor pressure to scram the control rods. Verification of charging water pressure ensures that if a scram is required, capability for rapid control rod insertion would exist. The minimum pressure of 940 psig, which is well below the expected pressure of approximately 1100 psig, ensures sufficient pressure for rapid control rod insertion. The 7 day Frequency has been shown to be acceptable through operating experience and takes into account indications available in the control room.

REFERENCES

1. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," August 1996.

2. Letter from T. Pickens (BWROG) to G. C. Lainas, NRC, "Amendment 17 to General Electric Licensing Topical Report NEDE-24011-P-A," August 15, 1986.

(continued)

BASES

REFERENCES (continued)

3. 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*).

4. 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*).