Tennessee Valley Authority, Post Office Box 2000, Decatur, Alabama 35609-2000

February 20, 2018

10 CFR 50.4
10 CFR 50.59(d)(2)
10 CFR 50.71(e)

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

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

Subject: **Summary Report for 10 CFR 50.59 Evaluations, Technical Specifications Bases Changes, Technical Requirements Manual Changes, and NRC Commitment Revisions - Updated**

Reference: TVA letter to NRC, "Summary Report for 10 CFR 50.59 Evaluations, Technical Specifications Bases Changes, Technical Requirements Manual Changes, and NRC Commitment Revisions," dated November 27, 2017 (ML17362A016)

The purpose of this letter is to provide the Nuclear Regulatory Commission (NRC) with an update to the referenced, required periodic submittal.

Enclosures 1, 2, and 3 of the referenced letter remain unchanged. Enclosure 4 of the referenced letter contains summaries of revised NRC commitments. The Tennessee Valley Authority (TVA) revised these commitments in accordance with the Nuclear Energy Institute's, "Guidelines for Managing NRC Commitment Changes," as endorsed in the NRC Regulatory Issue Summary 2000-17. An update to the referenced letter is necessary to correct mislabeled pages in Enclosure 4, which will permit the NRC to make the entire submittal available to the public. This submittal replaces the referenced submittal in its entirety. The mislabeling issue was entered into TVA's corrective action program.

There are no new regulatory commitments contained in this letter. If you have any questions regarding this report, please contact J. L. Paul, Nuclear Site Licensing Manager, at (256) 729-2636.

A001
NRR

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 20th day of February 2018.

Respectfully,

D. L. Hughes
Vice President

Enclosures:

Enclosure 1 - 10 CFR 50.59 Summary Report
Enclosure 2 - Technical Specifications (TS) Bases Changes and Additions
Enclosure 3 - Technical Requirements Manual (TRM) Changes and Additions
Enclosure 4 - Summary of Revised Commitments

cc (w/Enclosure):

NRC Regional Administrator - Region II
NRC Senior Resident Inspector - Browns Ferry Nuclear Plant

ENCLOSURE 1

Tennessee Valley Authority
Browns Ferry Nuclear Plant
Units 1, 2, and 3

10 CFR 50.59(d)(2) Summary Report

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1. DCN 72226A, Adjust Setpoints for 1/2/3-PS-32-70, Revision 0
2. DCN 51138 BFNP U1 Recovery Turb Bldg. I & C Lead DCN - SYS 003, Revision 1
3. DCN 51688A, Upgrade U3 HP Turbine And Auxiliaries for EPU, Revision 1
4. DCN 51689A, Upgrade U2 HP Turbine and Auxiliaries for EPU, Revision 1
5. DCN 67325A, Modify U2 FWCS Software for EPU Impact, Revision 1
6. DCN 71436A, Replace Existing U123 Liquid Radwaste Processing System With New Dual Train System, Revision 0
7. DCN 71988A, Replace U2 HPCI Vacuum Breaker Valves, Revision 1
8. DCN 71389A, Modify Hardened Containment Vent System to Comply with NRC Order EA-13-109, U1, Revision 0
9. DCN 67324A, Modify U3 FWCS Software for EPU Impact, Revision 2
10. DCN 71220A, Replace Existing Obsolete U1 50KVA SOLA Type I&C Bus Voltage Regulators, Revision 0
11. DCN 71731A, Tritium Mitigation for U123 Condensate Storage & Supply Steam, Revision 0
12. DCN 51216A, U1 Recovery - Electrical Lead DCN - System 57-4 (Reactor Bldg) Revision 0
13. DCN 51052A, Replace U3 Condensate Booster Pumps & Motors A, B, and C, Revision 1
14. DCN 71865B, Replace U3 HPCI Vacuum Breaker Valves, Revision 0
15. DCN 51057A, Replace U3 Reactor Feedwater Pumps A, B, and C, Revision 0
16. SNUBBERS U1/2/3 TRM and BASES, Revision 0
17. 0-AOI-57-1A R71 TN75 & FSAR 8.10.2, Revision 0
18. DCN 71212A, Replace Existing Obsolete U2 50KV SOLA Type I&C Bus Voltage Regulators, Revision 0
19. ECP 71666A, U2 Refuel Bridge Upgrades, Revision 0
20. DCN 71634A, Replace U3 Time Delay Relays for Loops 3-P-3-204& 3-L-3-208, Revision A
21. EOI Program Manual 0-TOC R85, Revision 0
22. DCN 51079A, U1 Recovery Control Bay-I&C Lead DCN-System 092, Revision 2
23. DCN 71390A, Modify Hardened Containment Vent System to Comply with NRC ORDER EA-13-109 U2, Revision 0
24. TVA-COLR-BF2C19, Revision 5
25. TVA-COLR-BF3C18, Revision 1 (BFE-4020, Revision 0)
26. TVA-COLR-BF1C12, Revision 0
27. TVA-COLR-BF2C20, Revision 0
28. TVA-COLR-BF3C17, Revision 2 (BFE-3731, Revision 1)
29. BFE-4076-R0 UFSAR Update, Revision 0

ENCLOSURE 1

Tennessee Valley Authority
Browns Ferry Nuclear Plant
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DCN 72226A, Adjust Setpoints for 1/2/3-PS-32-70, Revision 0

Design Change Notice (DCN) 72226 evaluates the ability of the inboard main steam line isolation valves (MSIVs) (1/2/3-FCV-1-14, 1/2/3-FCV-1-26, 1/2/3-FCV-1-37, and 1/2/3-FCV-1-51) to perform their design basis closure function during a loss of coolant accident (LOCA) inside containment. This evaluation is contained in MSIV design basis review calculation MDQ0000012016000566, "Main Steam Isolation Valve (MSIV) Component Level Review." Calculation MDQ0000012016000566 determines that the MSIVs can perform their design basis closure function within 2 minutes following a LOCA inside primary containment. Closure times for scenarios other than a LOCA inside primary containment are not impacted by this activity, and closure times for the outboard MSIVs are also not impacted. Core and containment pressures corresponding to 2 minutes following a LOCA are credited rather than peak core and containment pressures. DCN 72226 also changes the low pressure alarm setpoint (1/2/3-PS-32-70) of each unit's set of drywell control air receiver tanks to ensure adequate drywell control air pressure to close the inboard MSIVs.

Summary of Evaluation:

Changing the required MSIV closure time for a LOCA from the FSAR-described 10 seconds to 2 minutes and changing the MSIV closure method from air or springs to air and springs does not result in an increase in consequences, and is therefore acceptable. The credible failure mode of this activity is failure of the inboard MSIV to close. As discussed below, failure of an MSIV to close is already postulated and not impacted by this activity. Utilizing the extended power uprate containment pressure analysis results from 001N8894-R0 rather than the stretch power uprate results in GE-NE-B13-01866-4 as inputs into calculation MDQ0000012016000566 is acceptable because the methods used in 001N8894-R4 have been previously accepted by the NRC per the Peach Bottom extended power uprate license amendment (ML14133A046). Therefore, this activity does not require prior NRC approval.

DCN 51138 BFNP U1 Recovery Turb Bldg. I & C Lead DCN - SYS 003, Revision 1

The proposed modification upgrades the Unit 1 Reactor Feedwater Control System (RFWCS) from the existing obsolete discrete analog components to a fault tolerant distributed control system. Additionally, this modification upgrades the Unit 1 Reactor Feedpump Turbine (RFPT) control system and vibration monitoring equipment.

The Unit 1 Extended Power Uprate (EPU) modifications have a direct impact on various parameters utilized by the Unit 1 RFWCS such as steam flow, feed flow and RFPT speed. This DCN spans the Unit 1 RFWCS and RFPT control system software setpoints and scaling parameters to support operation at Extended Power Uprate conditions (120% of original licensed thermal power - OLTP); however, Unit 1 will operate at 105% OLTP and changed to support 120% operation after the approval of EPU.

The Reactor Vessel Water Level 2 setpoint could be reached for various reactor SCRAM events at EPU conditions with the End of Cycle Recirculation Pump Trip (EOC-RPT) disabled. Recirculation pump trip, initiation of the alternate rod injection logic, and the initiation of HPCI

and RCIC initiation occur at the Reactor Vessel Water Level 2 setpoint. This DCN adds a 75% Recirculation Runback on a reactor SCRAM signal to mitigate the magnitude of the reactor water level transients and minimize the probability of the level transients reaching the level 2 setpoint. The corrective action for PER 106204 to enable the EOC-RTP is being performed outside the scope of this DCN and is not addressed by this 10 CFR 50.59 evaluation.

The scope of this 10 CFR 50.59 evaluation is limited to the implementation of the SCRAM signal initiation of the 75% Recirculation Runback. The remaining changes in this DCN did not screen in for further review, and are addressed in the screening review associated with this 10 CFR 50.59 evaluation.

Summary of Evaluation:

Plant transient data indicates that if a SCRAM were to occur during operation at current licensed conditions (105% of original licensed thermal power or 3458 MWt) the existing recirculation 75% runback would be initiated on a Low Feed Pump Flow signal with a coincident Low Vessel Level signal.

Therefore, the modification to initiate a recirculation 75% runback directly from a SCRAM signal only initiates the runback a few seconds sooner to attempt to mitigate the magnitude of the level transients. There is no impact on the consequences of an accident, malfunction of equipment, or fission product barriers. Therefore, this DCN may be implemented per plant procedures without obtaining a License Amendment.

DCN 51688A, Upgrade U3 HP Turbine And Auxiliaries for EPU, Revision 1

This activity upgrades the High Pressure Turbine (HPT) and Auxiliaries to accommodate the higher steam flow requirements of EPU. This will involve the incorporation of GE's Advanced Design Steam Path (ADSP), which is designed for the increased flow associated with the EPU. The modified High Pressure Turbine has buckets and stationary steam path diaphragms designed to maximize efficiency at EPU operating conditions. The remaining changes in this DCN (CARV setpoint changes, Steam Seal changes and Instrumentation changes) did not screen in for further review, and are addressed in the screening review associated with this 10CFR50.59 evaluation.

Summary of Evaluation:

The modified High Pressure Turbine has buckets and stationary steam path diaphragms designed to maximize efficiency at EPU operating conditions. Correspondingly, the steam pressure and temperature conditions will change and are documented in the change package. The system piping has been analyzed and found to meet the new design conditions. The turbine-generator controls will continue to work in conjunction with the Nuclear Steam Supply System controls to maintain essentially constant reactor pressure and to limit transients during load variations. The entrapped energy increases slightly with EPU conditions. GE FTR T0700 demonstrates that even at EPU conditions, the probability of an external missile remains below the value of 1E-4 as described in the UFSAR. Implementation of the proposed HPT modifications will not affect frequency or consequences of an accident or malfunction of an SSC important to safety.

The components replaced by this activity operate in the same manner as the existing components and meet all the design requirements of the system, so no new failure modes are introduced. The HPT modifications will not create the possibility of a new accident or malfunction with a different result. The HPT modifications do not create a departure from a method of evaluation.

The TCV nominal position and characteristic curve are inputs to Generator Load Reject (TCV Fast Closure) with Turbine Bypass Valve Failure (LRNBP) and Turbine Bypass Valve Failure Following Turbine Trip, High Power (TTNBP) which are limiting transients for core-wide rapid pressurization. The nominal TCV position and the TCV characteristic curve (i.e. valve position versus steam flow) are affected by the HPT modification. However, the plant transient inputs are re-validated for each operating cycle with respect to the upcoming cycle and planned plant configuration. The limiting AOTs are re-analyzed for each fuel load and subsequent operating cycle to determine which is the most limiting and to demonstrate that the Design Basis Limit for a Fission Product Barrier (DBLFPB) affected by the AOTs are not exceeded. Thus, this process ensures the HPT modifications will not result in a design basis limit for a fission product barrier being exceeded or altered. Therefore, this activity can be implemented without obtaining a License Amendment.

DCN 51689A, Upgrade U2 HP Turbine and Auxiliaries for EPU, Revision 1

This activity upgrades the High Pressure Turbine (HPT) and Auxiliaries to accommodate the higher steam flow requirements of EPU. This will involve the incorporation of GE's Advanced Design Steam Path (ADSP), which is designed for the increased flow associated with the EPU. The modified High Pressure Turbine has buckets and stationary steam path diaphragms designed to maximize efficiency at EPU operating conditions. The remaining changes in this DCN (CARV setpoint changes, Steam Seal changes and Instrumentation changes) did not screen in for further review, and are addressed in the screening review associated with this 10 CFR 50.59 evaluation.

Summary of Evaluation

The modified High Pressure Turbine has buckets and stationary steam path diaphragms designed to maximize efficiency at EPU operating conditions. Correspondingly, the steam pressure and temperature conditions will change and are documented in the change package. The system piping has been analyzed and found to meet the new design conditions. The turbine-generator controls will continue to work in conjunction with the Nuclear Steam Supply System controls to maintain essentially constant reactor pressure and to limit transients during load variations. The entrapped energy increases slightly with EPU conditions. GE FTR T0700 demonstrates that even at EPU conditions, the probability of an external missile remains below the value of 1E-4 as described in the UFSAR. Implementation of the proposed HPT modifications will not affect frequency or consequences of an accident or malfunction of an SSC important to safety.

The components replaced by this activity operate in the same manner as the existing components and meet all the design requirements of the system, so no new failure modes are

introduced. The HPT modifications will not create the possibility of a new accident or malfunction with a different result. The HPT modifications do not create a departure from a method of evaluation.

The TCV nominal position and characteristic curve are inputs to Generator Load Reject (TCV Fast Closure) with Turbine Bypass Valve Failure (LRNBP) and Turbine Bypass Valve Failure Following Turbine Trip, High Power (TTNBP) which are limiting transients for core wide rapid pressurization. The nominal TCV position and the TCV characteristic curve (i.e. valve position versus steam flow) are affected by the HPT modification. However, the plant transient inputs are re-validated for each operating cycle with respect to the upcoming cycle and planned plant configuration. The limiting AOTs are re-analyzed for each fuel load and subsequent operating cycle to determine which is the most limiting and to demonstrate that the Design Basis Limit for a Fission Product Barrier (DBLFPB) affected by the AOTs are not exceeded. Thus, this process ensures the HPT modifications will not result in a design basis limit for a fission product barrier being exceeded or altered. Therefore, this activity can be implemented without obtaining a License Amendment.

DCN 67325A, Modify U2 FWCS Software for EPU Impact, Revision 1

The Reactor Feedwater Control System (RFWCS) automatically controls the flow of feedwater into the reactor vessel to maintain vessel water level within predetermined levels during all modes of plant operation and provides monitoring and alarming. Vessel water level control is accomplished by controlling feedwater flow into the vessel through the adjustment of the speed of the Reactor Feed Pump Turbines (RFPT). Each RFPT is controlled by its associated Woodward Governor that is automatically controlled by the Foxboro I/A RFWCS.

Unit 2 Extended Power Uprate (EPU) modifications have a direct impact on various parameters utilized by the Unit 2 RFWCS such as steam flow, feed flow and RFPT speed. This DCN updates the Unit 2 Foxboro I/A RFWCS software setpoint and scaling parameters and the Woodward Governor parameters required to support the Unit 2 EPU effort modifications. DCN 51056 installed higher capacity feedwater pumps to support operation at EPU rated flows. The increased flow capacity for the Condensate Feedwater System results in a higher runout flow rate for the feedwater controller failure maximum demand transient. The feedwater controller failure maximum demand is a bounding transient that is re-analyzed for each fuel reload and subsequent operating cycle. The higher runout is a revised input for the reload analysis. Results of the reload analyses are used to establish the Technical Specifications/COLR limits and ensure design and licensing criteria are met for each operating cycle. Therefore, the increase in the runout flow rate does not adversely affect the UFSAR transient and accident results. This modification does not involve a change to an SSC that adversely affects an UFSAR described design function. In addition, this DCN updates the Human Machine Interface (HMI) displays on the workstation on Panel 2-9-18. These displays provide graphical display data, system health monitoring, and communications functions for the Foxboro I/A system.

The RFWCS provides an input to automatically enable the rod worth minimizer (RWM) at low power operations. During low power operations, control rod blocks from the RWM enforce specific control rod sequences designed to mitigate the consequences of the control rod drop accident. PER 111293 was initiated to document that the current Unit 2/3 RWM bypass setpoints of 16% rated flow are based on a non-conservative accuracy calculation methodology. This DCN corrects the RWM setpoint calculation methodology for the Unit 2 RWM setpoints and updates the RWM setpoints to support 105% power operation with the flow instruments spanned for 120% EPU flow rates.

The steam and feedwater flow transmitters currently output a square rooted signal (which is proportional to flow) to the RFWCS. These instruments do not provide an accurate square rooted output signal at low flow rates. This inaccurate output signal is insignificant at the current flow ranges required for operation at 105% of original licensed power (3458 MWt). However, operation at EPU conditions (120% of original licensed power 3952 MWt) will require an expanded flow range which magnifies the impact of this inaccurate output signal at low flow conditions. DCN 67325 configures the flow transmitters to output a more accurate linear Signal, modifies the RFWCS to accept the linear Signal and to perform the square root function for use in the reactor vessel level control and re-spans the steam flow and feedwater flow parameters. The steam and feedwater flow transmitters will be re-spanned by other DCNs as EPU modifications are implemented.

The Reactor Vessel Pressure logic performs validation checks utilizing various parameters for the pressure loop inputs so that an invalid input would be bypassed and not adversely impact the logic control functions. With the existing pressure validation logic, if one channel was already bypassed the deviation between the average and median wide range pressure signal alarm validation logic would be inhibited for all three pressure channels. This DCN modifies the pressure validation logic so that only the required loops deviation logic is inhibited when pressure loop are bypassed. The remaining valid input channels would remain active in the control logic. This change makes the pressure validation logic function as currently described in the existing RFWCS software description.

PER 106204 was written to document that a reactor scram from EPU conditions with the End of Cycle Recirculation Pump Trip (EOC-RPT) disabled, could cause reactor water level transients that could reach the Level 2 setpoint resulting in a recirculation pump trip, initiation of the alternate rod injection logic, and the initiation of HPCI and RCIC. This DCN adds a 75% of rated flow Recirculation Runback on a reactor SCRAM signal (Full SCRAM) in an attempt to mitigate the magnitude of the reactor water level transients and minimize the probability of the level transients reaching the level 2 setpoint. However, simulator testing indicates that for more severe Reactor SCRAM events such as turbine trip or load reject at EPU power levels with the EOC-RPT disabled, the Reactor Level 2 setpoint could still be reached.

The scope of this 10 CFR 50.59 evaluation is limited to the implementation of the SCRAM signal initiation of the 75% Recirculation Runback. The remaining changes in this DCN did not screen in for further review, and are addressed in the screening review associated with this 10 CFR 50.59 evaluation.

Summary of Evaluation:

Actual plant transient data indicates that if a SCRAM were to occur during operation at current licensed conditions (105% of original licensed power or 3458 MWt) the existing recirculation 75% runback would be initiated on a Low Feed Pump Flow signal with a coincident Low Vessel Level signal. Therefore, the modification to initiate a recirculation 75% runback directly from a SCRAM signal only initiates the runback a few seconds sooner in an attempt to mitigate the magnitude of the level transients. There is no impact on the consequences of an accident, malfunction of equipment, or fission product barriers.

Therefore, this DCN may be implemented per plant procedures without obtaining a License Amendment.

DCN 71436A, Replace Existing U123 Liquid Radwaste Processing System with New Dual Train System, Revision 0

The Liquid Radwaste System (LRW) includes a single train "Thermex" processing skid, which is obsolete and has experienced continued operational challenges.

The existing, single train "Thermex" processing skid will be replaced by two (2) 100% redundant, higher capacity processing skids from AVANTech, Inc."Ultrex" LRW skids with a flowrate of 100 gpm per train.

The DCN scope of work includes installation of the vendor supplied LRW system and making necessary mechanical, electrical, and instrumentation interface connections to the existing plant systems. The Floor Drain Sample Tank (FDST) A-2 will be demolished to create space for installation of the new LRW skids and will allow continued "Thermex" processing during design installation.

The existing "Thermex" LRW skids have process connections from the Floor Drain Filter / Waste Filter, from the Backwash Condensate, and from the Laundry Drain Filter. The new "Ultrex" system will also receive waste flow from the same plant equipment and from the same locations, except for the Laundry Drain Filter. Since the "Ultrex" system has a minimum required inlet flow rate of 90 gpm, the Laundry Drain Filter water will need to be sent to the floor drain sump in order to be processed by "Ultrex". The existing hose connection from the Laundry Drain Pumps to "Thermex" will be spared and capped.

The new "Ultrex" skids include air compressors and a dryer to furnish all service/control air requirements. The existing ancillary connections that are not required for the "Ultrex" system will be cut back to the nearest available pipe/tube support, spared, and abandoned/spared in place. Isolation and abandonment of the existing ancillary system connections is also included in the design.

The process connections on the discharge of the new LRW skids will include a connection line to the remaining Floor Drain Sample Tank A-1 and a connection line to each of the two Waste Sample Tanks. Routing of the Reverse Osmosis (RO) waste water, filter backwash, and Granular Activated Carbon (GAC) backwash to the RW Building floor drains is provided. The

system design includes sluicing of solid waste from the GAC filters to High Integrity Containers (HICs) or their equivalent.

Floor Drain Sample Tank A-2 will be demolished to create space for installation of the new LRW skids and their control and power panels. An engineering evaluation has been prepared that provided justification for removing FDST A-2 and not replacing the lost capacity.

Removal of FDST A-2 requires that FDST A-1 and FDST A-2 are isolated from each other. Isolation will be accomplished by cutting and capping the equalizing, vent, recycle, overflow and pump suction lines for FDST A-2. Panel 925-190 will be removed from FDST A-2 and relocated to FDST A-1.

The new "Ultrex" LRW processing system supplied by AVANTech will have variety of non-safety related instruments and controls; this addition of instrument will not impact the existing instrument loop of the existing LRW circuit. Instrument sense lines will be field routed, which is acceptable per N1E-003 Section 3.2.1 for sense lines routed within a local panel. Additionally, the existing sense lines to which the new instruments will connect are also field routed.

Additionally, the level transmitter and sensing lines from FDST A-2 will be moved to FDST A-1. Isolation will be accomplished by draining both FDST A-1 and FDST A-2, removing the flanged equalization line and installing a blind flange in place of that line on FDST A-1. Isolation will also include cutting and capping the vent, recycle, overflow and pump suction lines for FDST A-2.

Level transmitter LT-77-55, located on panel 925-190 will be relocated from FDST A-2 to FDST A-1. Panel 925-74 will be removed entirely to provide a pathway for the new "Ultrex" equipment, and the existing pressure indicating instruments from panel 925-74 will be relocated onto Panel 925-207.

Hand switches for the floor drain sample pumps will also be relocated on a new Junction Box 12666 on Panel 925-207.

AVANTech scope of supply and installation includes new power panels, two control panels and all power between their skids, power panels and control panels. The design includes providing power to the two AVANTech power panels, installing a new breaker for the "Ultrex" Train 'A' on the 480V Service Building Main Board and providing power to the backup battery for the new CPU in the Radwaste Control Room. The existing "Thermex" breaker on the 480V Service Building Main Board will be re-used for the "Ultrex" Train 'B' system. New loads will be added to 480V Service Building Main Board by the addition of the new LRW processing system. Calculations have been performed to support the added load.

Replacement of the existing "Thermex" system with the new "Ultrex" system will increase the heat load in the RW Building EL 578' and 580'. The RW Building Ventilation System (System 030) cooling capability has not been modified by this change.

Contact(s) from the new "Ultrex" system will interface with an existing annunciator circuit to provide an alarm for trouble conditions of the "Ultrex" system. This modification will impact operators and the simulator due to the replacement of the LRW skids. A full Human Factors Evaluation (HFE) review has been completed for tis design.

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Some of the non-safety related instruments and controls are digital devices which are being added via the new "Ultrex" system, therefore, a new SQAP Calculation for LRW Software will be provided per NPG-SPP-12.7 and SS-E.18.15.01. A new non-safety related cable will connect the vendor provided LRW Networking Equipment with the vendor provided new CPU in the Radwaste Control Room Operator Desk.

This design also installs 3 cameras that will be used to remotely view "Ultrex" local instrumentation.

Screening Review Items that Screened in for further evaluation:

The impacts to the Radwaste System do not change the design functions of liquid radwaste. It does however impact the systems design functions because of the following and requires additional evaluations:

- Reduction in Floor Drain Sample Tank Capacity
- Digital Upgrades to the Liquid Radwaste System
- System conversion from a vendor skid to a plant configured new system
- Reduction in design margin due to ASME Section VIII revision used in design of vessel

Summary of Evaluation:

The proposed activity replaces an existing system with a new system for the processing of liquid radwaste. There are modernization and digital controls added as part of the new system, but these improvements are limited to the processing of liquid radwaste and do not create an accident scenario or interfacing system scenario that has not been previously evaluated. It does not create new malfunctions beyond those previously evaluated for the system.

The reduction in floor drain sample tank capacity has been evaluated and the reduced capacity is acceptable to meet the requirements for the system.

The digital upgrade to the system has been evaluated and has no impact on the design functions for the system. New failures of components have been reviewed and do not impact the systems previously evaluated failures. The new system adds additional redundancy in the design.

System conversion from a vendor skid to a permanent plant configured system provides the plant with a configuration controlled and plant operated permanent system. It does not impact the design functions of the system.

Reductions in design margin due to ASME Section VIII revision used in design of vessel was evaluate and determined that although the requirement in the TVA Specification allowed for a reduced safety factor, the actual calculations completed proved the safety factor to exceed the older requirements. Thus the design safety factors were not reduced by this modification.

Based on this evaluation, the proposed activity may be implemented without a License Amendment being obtained from the NRC.

ENCLOSURE 1

Tennessee Valley Authority
Browns Ferry Nuclear Plant
Units 1, 2, and 3

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DCN 71988A, Replace U2 HPCI Vacuum Breaker Valves, Revision 1

Browns Ferry Nuclear (BFN) has requested a Design Change Notice (DCN) to replace the High Pressure Coolant Injection (HPCI) turbine exhaust vacuum breaker 2 inch lift check valves 2- CKV-073-0633, 2-CKV-073-0634, 2-CKV-073-0635, and 2-CKV-073-0636 with 3 inch, vertically mounted, downward flow nozzle check valves. The existing 2 inch shutoff valve (2-SHV-073- 0534) and a short portion of the existing 2 inch pipe run will be retained. The majority of the existing 2 inch piping in the HPCI vacuum breaker system will be increased from 2 inch nominal pipe size to 3 inch nominal pipe size. Stop check valve 2-ISV-073-0023 and lift check valve 2-CKV-073-0603 and the associated HPCI turbine exhaust piping will be elevated to maximize the elevation difference between the highest point in the HPCI turbine exhaust header and the maximum suppression pool level. A 3/4 inch test line, with two 3/4 inch test valves, will be installed between the vacuum breaker valve assembly and manual shutoff valve 2-SHV-073- 0534 to provide real time pressure versus time data during HPCI surveillance testing.

Summary of Evaluation:

The proposed modification does not require NRC approval prior to implementation. The primary containment system is adversely affected by the addition of the new HPCI 3/4" instrument line and test valves 2-TV-073-1745 and 2-TV-073-1746 which introduce a new flow path outside of Primary Containment. However, the new instrument line and test valves 2-TV-073-1745 and 2-TV-073-1746 do not result in a more than minimal increase in:

- The frequency of occurrence of an accident;
- Likelihood of an SSC failure;
- The consequences of an accident; or,
- The consequences of a malfunction.

Nor does the addition of the new instrument line and test valves 2-TV-073-1745 and 2-TV-073-1746 create or affect:

- A possibility of a different type of accident not previously evaluated;
- A possibility of a malfunction of an SSC important to safety with a different result;
- A design basis limit for a fission product barrier being exceeded or altered; or,
- A method of evaluation described in the UFSAR used to establish the design bases or safety analysis.

The new test valves 2-TV-073-1745 and 2-TV-073-1746 and associated piping are not the initiator of any postulated design basis accidents, are passive components that will have standard post-modification testing in accordance with industry standards and codes to ensure pressure boundary integrity, and will be normally closed valves that are not required to function during or following a design basis event. Additionally, the consequences of failure of the new test valves 2-TV-073-1745 and 2-TV-073-1746 and associated piping are bounded by those previously evaluated for a Category D accident. A Category D accident results in radioactive release directly to secondary containment with primary containment not intact. The Category D design basis accident is a refueling accident (fuel assembly drops on spent fuel during refueling)

(Ref. UFSAR, Section 14.6.1). Based upon the results of this evaluation: Implement the activity per plant procedures without obtaining a License Amendment.

DCN 71389A, Modify Hardened Containment Vent System To Comply with NRC ORDER EA-13-109, U1, Revision 0

Following the March 11, 2011 Fukushima Dai-ichi accident in Japan, which resulted from a beyond design basis tsunami, producing severe damage to three (3) of the site's six (6) boiling water reactors (BWRs), NRC Order EA-12-050 was issued on March 12, 2012 to U.S. licensed BWRs requiring licensees to implement requirements for reliable hardened containment vents at their facilities. NRC Order EA-12-050 notes that hardened containment vents have been in place in the U.S. Mark I and Mark II BWR containments for many years as the result of previous NRC regulatory action contained in Generic Letter 89-16, while recognizing that variances exist among U.S. BWRs regarding the capability to vent containment during a broad spectrum of events. The conclusion of NRC Order EA-12-050 is that on the basis of the need to enhance the health and safety of the public, a reliable Hardened Containment Venting System (HCVS) for Mark I and Mark II containments is mandated for all BWR license holders. Specifically, NRC Order EA-12-050 directed BWR licensees to add qualification features to the hardened vent installations that were originally performed in response to NRC Generic Letter 89-16. For Browns Ferry (BFN) Units 1, 2, and 3, a system similar to the HCVS described in NRC Order EA-12-050, termed the Hardened Wetwell Vent (HWWV), was installed in response to NRC Generic Letter 89-16.

On June 6, 2013, the NRC issued Order EA-13-109 to rescind and replace the previous NRC Order EA-12-050. Phase I of NRC Order EA-13-109 implementation requires installation of an HCVS that will remain functional during severe accident conditions (i.e., when core damage has occurred). The primary design objective of the HCVS is to provide sufficient venting capacity to prevent a long-term overpressure failure of the containment by restoration and maintenance of containment pressure below the Primary Containment design pressure and the Primary Containment Pressure Limit. On June 30, 2014, the Licensee (TVA) communicated its planned commitment for installing the HCVS at BFN by submitting an Overall Integrated Plan (OIP) in its response to NRC Order EA-13-109 utilizing nuclear industry guidance provided by NEI 13-02 that is acceptable to the NRC for licensees to use in meeting the requirements of NRC Order EA-13-109. TVA submitted to the NRC a 6-month update to the OIP on December 19, 2014, and on February 11, 2015, the NRC issued an Interim Staff Evaluation relating to the design and ongoing implementation of Order EA-13-109 at BFN. The second, third, and fourth 6-month updates to the OIP were submitted on June 29, 2015, December 29, 2015, and June 30, 2016, respectively.

Using the TVA commitments in the OIP and subsequent communications as a basis, DCN 71389 documents the design to implement a reliable, severe accident capable, HCVS for Unit 1 at BFN. Separate DCNs will be prepared to implement the similar HCVS for BFN Units 2 and 3. When implemented the Unit 1 HCVS will comply with Phase I requirements of NRC Order EA-13-109 along with the appropriate site guidance.

Portions of the existing HWWV that were installed in response to NRC Generic Letter 89-16 will be used in the overall HCVS that is implemented by DCN 71389. The HVWWV provides a direct vent path from the wetwell (also referred to as the torus) to an exhaust point inside the concrete portion of the existing plant stack above elevation 666.5'. The original flow path utilized a 14 inch line tapping off of the 20 inch torus vacuum breaker piping at Reactor Building elevation 565'. Two 14 inch pneumatically operated Primary Containment Isolation Valves (PCIVs) (1-FCV-064-0221 and 1-FCV-064-0222) are located within the section of HWWV piping inside the Reactor Building. The existing HVWWV penetrates the Reactor Building wall along column line U at a location west of column line R4. After exiting the Reactor Building, the current pipe enters a valve pit where 1-SHV-064-0737 is located and then exits the valve pit tying into a common header for all three Units which routes to the existing plant stack. The existing HWWV in its original configuration is not adequate for meeting the newest regulatory guidance due to new requirements for Unit separation, effluent release point, the need to address flammable gasses, and requirements to function during a severe accident when core damage has occurred. To meet the latest regulatory requirements a new vent path will be established specifically for each unit, however, DCN 71389 only addresses Unit 1. Portions of the Unit 1 HWWV interior to the Reactor Building are adequate for reuse.

The existing HWWV piping will be rerouted underground, in a valve pit, just south of Reactor Building column line U. Following permanent removal of 1-SHV-064-0737, new HCVS piping will elbow up at the flange location previously upstream of 1-SHV-064-0737 traveling vertically up the exterior wall of the Reactor Building. The piping will continue up the wall of the Reactor Building until it reaches the approximate elevation of 665', where it will turn and penetrate the siding of the Reactor Building superstructure. Once inside the superstructure the piping will be routed vertically adjacent to an existing steel column until it penetrates the roof of the superstructure, ultimately terminating at the approximate elevation of 741'-6". The portion of piping previously downstream of 1-SHV-064-0737 (which ties to the common HWWV header) will be cut and capped as close as possible to the South wall of the valve pit penetration such that remaining HWWV common header piping will remain functional for Units 2 and 3.

Altering the existing HWWV piping configuration as described for implementation of HCVS changes the original release point from inside the 600' tall plant stack (at approximate elevation 666.5') to a location above the Unit 1 Reactor Building at approximate elevation 741'-6". The change in release point not only impacts beyond design basis venting scenarios which utilize HCVS but also alters the release point for a design basis Loss of Coolant Accident (LOCA) as described in Section 14.6.3.5 of the UFSAR. As described in 14.6.3.5.e of the UFSAR, releases occur as a result of leakage past existing HWWV PCIVs (1- FCV-064-0221 and 1-FCV-064-0222). By altering the original HWVVV release point for HCVS to a location near the Unit 1 Reactor Building Exhaust Fans the UFSAR described design function of the Primary Containment System is adversely impacted. Specifically the Primary Containment System is required to limit leakage during and following a postulated LOCA to values which are substantially less than leakage rates that would result in offsite doses approaching 10 CFR 50.67 limits. This change has the potential to increase the consequences of a design basis LOCA.

Summary of Evaluation:

After evaluation, the changes required to implement HCVS do not:

- increase in the frequency of occurrence of an accident previously evaluated in the UFSAR,
- increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR,
- result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR,
- create a possibility for an accident of a different type than any previously evaluated in the UFSAR,
- create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR,
- result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered,
- result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

Implementation of HCVS does increase the consequences of an accident previously evaluated in the UFSAR, however, when evaluated against the criteria established by Section 4.3.3 of NEI 96-07 the increase is not more than minimal. Based upon the results of this evaluation: Implement the activity per plant procedures without obtaining a License Amendment.

DCN 67324A, Modify U3 FWCS Software for EPU Impact, Revision 2

The Reactor Feedwater Control System (RFWCS) automatically controls the flow of feedwater into the reactor vessel to maintain vessel water level within predetermined levels during all modes of plant operation and provides monitoring and alarming. Vessel water level control is accomplished by controlling feedwater flow into the vessel through the adjustment of the speed of the Reactor Feed Pump Turbines (RFPT). Each RFPT is controlled by its associated Woodward Governor that is automatically controlled by the Foxboro I/A RFWCS.

Unit 3 Extended Power Uprate (EPU) modifications performed by other DCNs have a direct impact on various parameters utilized by the Unit 3 RFWCS such as steam flow, feed flow and RFPT speed.

This DCN updates the Unit 3 Foxboro I/A RFWCS software setpoint and scaling parameters and the Woodward Governor parameters required to support the Unit 3 EPU effort modifications.

Co-requisite DCN 51057 will install higher capacity feedwater pumps to support operation at EPU rated flows. The feedwater controller failure maximum demand transient results in a higher flow rate due to the increased flow capacity of the Condensate Feedwater System. The feedwater controller failure maximum demand is a bounding transient that is re-analyzed for each fuel reload and subsequent operating cycle. The higher runout is a revised input for the reload analysis. Results of the reload analyses are used to establish the Technical Specifications/COLR limits and ensure design and licensing criteria are met for each operating

cycle. Therefore, the increase in the runout flow rate does not adversely affect the UFSAR transient and accident results.

Additionally a UFSAR change request is issued to remove the runout flow for the Feedwater Controller Failure (FWCF) - maximum demand - abnormal operational transient. The percentage documented in section 14.5.8.1.2 was correct for the old feedwater pumps but is incorrect for the new pumps. Since section 14.5.8.1.2 already states that the value used in the core re-load analysis may be adjusted to account for updated equipment performance, the specific historical percentages previously used should account for updated equipment performance, the specific historical percentages previously used should be removed from the FSAR to prevent misinterpretation.

PER 106204 was written to document that from simulator testing, it is anticipated that the Reactor Level 2 setpoint could be reached for some Reactor SCRAM events at EPU power levels with the End of Cycle-Recirculation Pump Trip (EOC-RPT) disabled. This would result in a recirculation pump trip, initiation of the alternate rod injection logic, and the initiation of HPCI and RCIC. This DCN adds a 75% of rated flow Recirculation Runback on a reactor SCRAM signal (Full SCRAM) in an attempt to mitigate the magnitude of the reactor water level transients and minimize the probability of the level transients reaching the level 2 setpoint. However, simulator testing indicates that for more severe Reactor SCRAM events such as turbine trip or load reject at EPU power levels with the EOC-RPT disabled, the Reactor Level 2 setpoint could still be reached.

A UFSAR Change Request issued in DCN 67324 documents the new SCRAM signal initiation of the 75% Recirculation Runback. A second UFSAR Change Request removes the actual feedwater low flow setpoint value from the UFSAR description and generically discusses that the 28% Recirculation Runback is initiated at the low flow setpoint.

The scope of this 10 CFR 50.59 evaluation is limited to the implementation of the SCRAM signal initiation of the 75% Recirculation Runback at EPU operation. The remaining changes in this DCN did not screen in for further review, and are addressed in the screening review associated with this 10 CFR 50.59 evaluation.

Summary of Evaluation

Actual plant transient data indicates that if a SCRAM were to occur during operation at current licensed conditions (105% of original licensed power or 3458 MWt) the existing recirculation 75% runback would be initiated on a Low Feed Pump Flow signal with a coincident Low Vessel Level signal. Therefore, the modification to initiate a recirculation 75% runback directly from a SCRAM signal only initiates the runback a few seconds sooner in an attempt to mitigate the magnitude of the level transients. There is no impact on the consequences of an accident, malfunction of equipment, or fission product barriers.

Based upon the results of this evaluation: Implement the activity per plant procedures without obtaining a License Amendment.

10 CFR 50.59(d)(2) Summary Report

DCN 71220A, Replace Existing Obsolete U1 50KVA SOLA Type I&C Bus Voltage Regulators, Revision 0

The purpose of this DCN is to replace existing obsolete I & C Bus Regulating Transformers for Unit 1 (1-XFA-253- 0001A2 and 1-XFA-253-0001B2). The I & C Bus Regulating Transformers are safety related components that provide power to 120/208V AC Instrument and Control Power System, System 253. The function of this system is to provide regulated power to vital instruments and controls during normal and design basis events.

As discussed in the Screening Review, the only aspect of this change which is considered "adverse" is the additional time needed for the new transformers to supply regulated output voltage after being energized. There is a short delay before the transformer output is fully regulated (about 22 seconds). The time for the existing transformers to achieve regulated output voltage upon being energized is not known; however, the time is believed to be closer to a few seconds.

This activity is necessary due to BFN U2 experiencing difficulties adjusting the voltage on one of the existing transformers. The transformers are obsolete and spare parts are not readily available. Replacement of the transformers would reduce operational risk due to failure of the transformers. Also, the new transformers would allow for having a spare parts inventory in the event that unforeseen repairs are needed.

Summary of Evaluation:

DCN 71220 does not require prior NRC approval as the design functions of the I & C Bus Regulating Transformers remain the same. The change replaces obsolete equipment with equipment performing the same design function. The replacement transformers do take additional time to provide a regulated output voltage upon being energized. However, the UFSAR does not discuss any specific time requirements with respect to availability of the 120/208V AC Instrument and Control Power System. The events applicable to this change are those associated with a loss of normal auxiliary power. With respect to this change, the bounding malfunction for a loss of normal auxiliary power would be an initial loss of the 120/208V AC Instrument and Control Power System. This would be true for the existing transformers as well. The delay to provide regulated output voltage does not prevent any downstream load from performing its design function. This has been evaluated in calculation EDQ125320020060. Additionally, no other equipment is affected by this change. Therefore, the additional time required to achieve regulated output voltage does not impact the likelihood of malfunctions of the 120/208V AC Instrument and Control Power System or malfunctions of the transformers themselves.

The 120/208V AC Instrument and Control Power System is designed as an interruptible power source. The system is designed to account for momentary outages or voltage transients. Therefore, the additional time required to achieve regulated output voltage does not alter the system performance requirements of the 120/208V AC Instrument and Control Power System. The new regulating transformers will allow the 120/208V AC Instrument and Control Power System to perform its design function to support mitigating design basis events, particularly those associated with a loss of normal auxiliary power ("Loss of Coolant Accident", "Events

Resulting in a Reactor Vessel Coolant Inventory Decrease" and "Station Blackout Event"). No new failure modes are created by this modification. This modification replaces obsolete equipment with a qualified reliable alternative.

Therefore, a UFSAR described design function is not impacted by this change and the change does not require NRC approval.

DCN 71731A, Tritium Mitigation for U123 Condensate Storage & Supply Steam, Revision 0

This proposed activity implements two design changes to support tritium mitigation strategies developed for the condensate storage and supply system at Browns Ferry Nuclear Plant (BFN):

1. Installing a new level control system for the condensate head tank and removing the current system; and
2. Locking the condensate head tank drain valve 0-DRV-002-0753 closed and adding a radioactive liquids caution tag.

Historic monitoring of groundwater wells at BFN has indicated plumes of tritium exist in the vicinity of the condensate storage tanks (CSTs) and condensate transfer tunnel. Inadvertent releases from the condensate storage and supply system over time may have been the cause of this elevated tritium concentration. A report produced by Bartlett Nuclear, Inc. in 2011 identified the CSTs and supply lines at BFN as being high-risk channels for contaminating the groundwater with licensed radioactive materials.

This 2-stage DCN evaluates and documents the modifications described above. The stage groupings and their scopes are:

Stage 1: Condensate Head Tank Level Control System Replacement and Drain Valve Lock

This stage installs a new level transmitter in the existing level control cabinet for the condensate head tank, located on the Unit 1 reactor building roof. The existing level switches will be removed. A new PLC-based control system will be installed for the condensate transfer pump automatic level controls. The four analog level switches are being replaced by a transmitter-and-PLC combination. This stage also installs a field-determined locking device to the condensate head tank drain valve 0-DRV-002-0753. For additional controls, a radioactive liquids caution tag will be hung from the drain valve.

Stage 2: Main Control Room Work to Support New Level Indication for Condensate Head Tank

This stage installs a new indicator in the control room and consolidates two existing alarms (HIGH/LOW) for the condensate head tank into a single trouble alarm.

Summary of Evaluation

Based on Question 1 of the 10 CFR 50.59 Screen, the proposed activity was found to potentially adversely affect an UFSAR-described design function because previously separate functions (mechanical, analog level switch control) will be combined into a combination transmitter-and-PLC digital device. NRC approval is not required to implement the proposed

activity because the documented evaluation demonstrates that none of the eight criteria in 10 CFR 50.59(c)(2) are met.

NRC Information Notice (IN) 2010-10 discusses a digital modification to the LaSalle non-safety related control rod drive system. The NRC criticized LaSalle's application of 10CFR50.59 for not answering all the questions in Appendix A of NEI 01.01 in the associated 50.59 Evaluation. In the criteria evaluation below, the questions in Appendix A of NEI 01.01 are provided in italics and answered for this modification.

Following the NEI 01.01 questions the criterion question is answered for the analyzed activity.

Based upon the results of this evaluation: Implement the activity per plant procedures without obtaining a License Amendment.

DCN 51216A, U1 Recovery - Electrical Lead DCN - System 57-4 (Reactor Bldg) Revision 0

This Evaluation is prepared for portions of modification issued in DCN 51216. Specifically the Removal of the Unit 1, LPCI M-G Sets that feed 480V RMOV Boards 1D and 1E.

The change deletes the automatic bus transfer scheme between 480V Bus 1D and 1E which power valves required tor LPCI injection mode of RHR. Technical Specifications SR 3.5.1.12, LCO 3.7.8 and technical requirements manual (TRM) Section 3.8.7.C.1 are removed/revised by a separate technical specification submittal number 427.

The MG Sets were installed to be used as electrical isolators between the 480V Shutdown Boards and the 480V RMOV Boards 1D / 1E (Reference ECN L1845 installed in 1977). The MG sets were required to be used as Isolators to assure the Integrity of the redundant divisional power supplies, since an automatic transfer scheme is used. This configuration provides automatic access to both divisions of power for each 480V RMOV Board. The automatic transfer scheme thus limits the significance of postulated DG or 480V Shutdown board failures upon the RHR-LPCI injection valves. However, postulated failures which render the 480V RMOV board itself inoperable can not be limited by the transfer scheme.

The equipment directly affected by this DCN includes:

MG Sets 1DN, 1DA, 1EN, 1EA
480V RMOV Boards: 1D / 1E

Summary of Evaluation:

480V RMOV Board 1D and 1E provide power to the respective Division I and II RHR valves which provide RHR-LPCI injection to the core. The board's power supply is designed to automatically transfer to the Emergency source, which is supplied from the opposite division 480V Shutdown Board. The original design basis of the automate transfer capability between the RMOV Boards was to provide a design feature to ensure that LPCI Injection occurred from both divisions with at least one pump in each loop. If one Loop's injection valve lost power it would automaticity transfer to the opposite division's power supply to ensure operation of the valve. It was recognized that this would not provide LPCI injection when a single failure was taken on a RMOV board or the valve Itself, however, adequate redundant equipment would be

available with the loss of the RMOV Board as the single failure. Subsequently, it was determined that the automatic transfer feature presented a potential concern for propagating a fault to both divisions of power supply. As a result, MG sets were added (Reference ECN L1845) to both the normal and emergency supplied to provide isolation between the associated 480V Shutdown Board and the RMOV Boards 1D/1E.

In 1996, Brown Ferry adopted the General Electric SAFER/GESTR model for evaluation of its LOCA analysis (Reference Calculation ND-Q0-000-970008). Based on the minimum ECCS requirement documented in the new analysis and UFSAR Table 6.5-3, the RHR system is no longer required to have at least one RHR pump running in each loop for a design basis recirculation line break, or one pump in one loop for a discharge line break. The RHR system is now only required to have two pumps running in one loop for a suction line break and no RHR pumps are required for a discharge line break with a concurrent loss of a diesel generator. Thus it is no longer necessary to have both loops provide LPCI Injection and the automatic transfer capability between the RMOV Boards is not required.

With the deletion of the automatic transfer capability between the RMOV Boards, the MG sets are no longer required to provide an isolation function and are deleted. Deletion of the automatic transfer components and the MG sets will removes potential failure modes and improves the reliability of the power supply to the downstream valves. This has been documented in a FME analysis (Reference Calculation ED-Q1999-2003-0051).

Removal of the automatic transfer feature will render 480V RMOV Boards 1D and 1E to preform the same safety related function as any RMOV Board located in Browns Ferry. These boards are physically located in the Unit 1Reactor Building which is a harsh environment. The loads are relocated to a mild environment. Furthermore, Appendix R fire zone analyses conclude that the boards must be relocated to satisfy Appendix R physical location requirements or the power supplies must be relocated to another source. There are only 3 loads on each board. Therefore, the loads from Division I board 1D will be relocated to Division I Board 1A. Division II loads from board 1E will be relocated to Division II board 1B. The 480V RMOV Boards 1A and 1B are physically located in a mild environment and satisfy Appendix R safe shutdown requirements.

The feeder breaker on the 480V Shutdown Board and the supply breaker on the 480V RMOV Boards 1A and 1B in series provides sufficient protection to ensure the integrity of the redundant division of power should transfer lo the emergency source be desired. System Voltage Drop and Loading are evaluated in electrical calculation ED-Q0-057-2002-0022 and are determined to be adequate to move an of the loads to RMOV Boards 1A and 1B.

DCN 51216 abandons the existing 480V RMOV Boards 1D and 1E. The loads are relocated in DCN 51218 for system 68 loads; DCN 51222 for System 074 Loads. The specific elements of the load relocation are discussed in the 50.59 reviews for those packages.

However, technical specifications SR 3.5.1.12 and LCO 3.7.8 and technical requirements manual (TRM) Section 3.8.7.C.1 are required to be revised as a result of this DCN. In accordance with SPP 9.4, a license amendment is required.

DCN 51052A, Replace U3 Condensate Booster Pumps & Motors A, B, and C, Revision 1

The activity evaluated is the procedure changes required to implement Operational load limitations. In order to ensure the Normal Auxiliary Power System is not adversely impacted when fed from the 161kV offsite source. Specifically, auto transfer of the 4kV Unit Board 3C to the 161kV System will be blocked whenever 480V Shutdown Board 3B is fed from its alternate supply. This limitation applies when Unit 3 is In Modes 1, 2, or 3. This change is made to alleviate any adverse voltage drop Impacts when 4kV Unit Boards are fed from the 161kV System. The effect of this load limitation would be that non-safety related 4kV Unit Board 3C would be de-energized upon a loss of the normal 500kV offsite circuit. The activity does not Impact plant operations with normal electrical plant alignments and does not adversely impact any electrical power feed to safety related Emergency Core Colling System (ECCS) loads. The activity does not introduce any new failure modes.

Summary of Evaluation

Operational load limitations will be implemented as part of the DCN to ensure the Normal Auxiliary Power System is not adversely impacted when fed from the 161kV offsite source. Specifically, auto transfer of 4kV Unit Board 3C to the 161kV System will be blocked whenever 480V Shutdown Board 3B is fed from its alternate supply. Blocking the auto transfer of 4KV Unit Board 3C protects the alternate off-site power source for safety related loads by temporarily defeating the auto transfer of the alternate power source for some non-safety related loads during the off normal alignment of the 480V Shutdown Board 3B. This limitation applies when Unit 3 is In Mode 1, 2, or 3. 4KV Unit Board 3C provides power to non-safety related equipment. Loss of power to 4KV Unit Board 3C due to blocking of the automatic transfer will not affect the frequency or consequences of an accident or malfunction of an SSC, or create the possibility of a new accident or malfunction with a different result. This activity will not affect the design basis limits for fission product barriers. Therefore, this activity can be implemented without obtaining a Licensing Amendment.

DCN 71865B, Replace U3 HPCI Vacuum Breaker Valves, Revision 0

BFN has experienced backflow of suppression pool water into the Unit 3 HPCI exhaust line following a HPCI turbine trip as described in multiple Condition Reports (CRs) including a Level 2 Evaluation from June 2015 (CR 1038747). This condition is non-conforming with respect to Design Criteria BFN-50-7073, High Pressure Coolant Injection System (Ref. Section 3.7.2, Item 29); however, there is no UFSAR licensing commitment associated with this condition. TVAEBFN042-REPT-001, HPCI Turbine Exhaust Study (Ref. RIMS #: RO6 151208 194), was issued in November 2015 to evaluate the cause of the backflow. The cause of the backflow was determined to be a vacuum created from the condensing steam after the HPCI turbine is secured or tripped. The amount of backflow is amplified by the minimal elevation difference between the suppression pool water level and the centerline of the exhaust line, as well as the differential pressure needed to open the existing HPCI exhaust line vacuum breaker check valves (3-CKV-073-0633, -0634, -0635, and -0636).

ENCLOSURE 1

Tennessee Valley Authority
Browns Ferry Nuclear Plant
Units 1, 2, and 3

10 CFR 50.59(d)(2) Summary Report

The proposed modification implements the following changes to the HPCI turbine exhaust line and HPCI vacuum breaker subsystem:

- Replace the existing vacuum breaker lift check valves (3-CKV-073-0633, -0634, -0635, and -0636) with faster acting nozzle check valves. The new nozzle check vacuum breaker valves will be 3 inch nominal diameter, whereas the current lift check valves are 2 inch nominal diameter.
- Increase the vacuum breaker line size from 2 inch nominal diameter piping to 3 inch nominal diameter piping.
- Increase the elevation difference between the torus penetration and the centerline of the HPCI turbine exhaust piping to be consistent with the recommendations provided by the BWR (Boiling Water Reactor) Owners' Group Technical Report of 5 feet (Ref. GEH 0000-0095-9231-R0, BWR Owners' Group Technical Report, Water Hammer Concerns in HPCI / RCIC Turbine Exhaust Lines Due to Vacuum Breaker Design).
- Install a new test line for temporary instrumentation between the vacuum breaker assembly and the existing stop check valve (3-ISV-073-0023) to provide real time pressure versus time data during HPCI surveillance testing. Two new shutoff valves will be installed on the instrumentation line (3-TV-073-0745 and 3-TV-073- 0746). The instrumentation line test valves will be normally closed and the instrument line will only be in service during HPCI surveillance testing.

Due to lead times associated with the new nozzle check valves, DCN 71865 has two stages to allow implementation and return to operations (RTO) of as much work as possible during the Unit 3 outage. Flanged connections are designed on each side of the new and existing vacuum breaker assemblies to support installation of these components at a future date.

Stage 1 modifications will include:

- Increase the HPCI turbine exhaust piping elevation to the maximum extent possible. Move the existing containment isolation valves, 3-ISV-073-0023 and 3-CKV-073-0603, closer to the HPCI turbine to allow the valves to be the high points in the exhaust line to ensure proper drainage. Install a 20 inch diameter pipe spool piece where existing valves are currently located. Optimize the piping design to minimize impact to existing structures, piping, and components to the maximum extent possible.
- Install new vacuum breaker piping with larger piping diameter to increase vacuum relieving capacity. The existing 2 inch vacuum breaker piping penetration (X-220) will not be affected. All piping downstream of 3- FCV-073-0064, near the 2x3 inch reducer, will be replaced. The piping upstream of 3-FCV-073-0064 until approximately 6 inches from X-220 will also be replaced with 3 inch pipe.

The new vacuum breaker piping will be provided with flanges on each side of the existing vacuum breaker assembly to support future removal and replacement of the existing vacuum breaker assembly with a pipe spool (Stage 2). Flanged connections for the new nozzle check valve vacuum breaker assembly will be installed to support installation in Stage 2. The new vacuum breaker assembly will have a new vertical-downward flow configuration and will be located as close as possible to the torus penetration to reduce system full flow pressure.

- Replace the existing 2 inch isolation valve 3-SHV-073-0534 with a new, larger diameter valve (3 inch).
- Install the new 3/4 inch instrumentation piping that will contain two ASME Section III, Code Class 2 equivalent test valves. A threaded cap will be installed at the end of the instrumentation piping to support future connection of a temporary pressure gauge.

Stage 2 modifications will include:

- The existing vacuum breaker assembly will be removed and replaced with a flanged piping spool piece. The new 3 inch vacuum breaker assembly utilizing nozzle check valves will be installed between the flanged connections previously installed in Stage 1. Stage 2 is expected to be implemented during a 14-day HPCI system outage (Limiting Condition for Operation (LCO) 3.5.1.C.2), but will not require a Unit outage.

This modification does not substitute the current manual action of test valves and check valves with automatic actions. This modification also does not add or modify existing operator actions associated with the HPCI system that are credited in the safety analysis.

This modification does not install any permanent plant digital assets; therefore, the requirements of NEI 01-01 are not applicable.

The HPCI System (System 073), Primary Containment Isolation System (System 064), Secondary Containment System (Reactor Building) (System 064C), and Fire Protection & Raw Service Water System (System 026) are SSCs affected by the proposed modification. Impacts to these systems as a result of this proposed modification are evaluated in the 50.59 Screening Review and determined to not be adverse to their UFSAR described design functions.

The Primary Containment (System 064A) is also affected by the proposed modification. Impacts to the primary containment system were determined to be adverse in the 50.59 Screening Review. The proposed modification will install a new HPCI test instrument line and associated test valves 3-TV-073-0745 and 3-TV- 073-0746 (Ref. DCA-71865-101). Test valves 3-TV-073-0745 and 3-TV-073-0746 will not be containment isolation valves, but will be included in the Appendix J Primary Reactor Containment Leakage boundary. The addition of a new flow path from primary containment is adverse to the design function of primary containment which is to retain integrity as a radioactive material barrier during and following accidents that release radioactive material into the primary containment volume (Ref. UFSAR, Section 1.5.1.6).

Therefore, this 50.59 evaluation addresses the adverse change to the design function of the primary containment system.

The proposed modification is necessary to mitigate backflow of suppression pool water into the Unit 3 HPCI exhaust line following a HPCI turbine trip. Specifically, installation of the new HPCI 3/4" instrument line and test valves 3-TV-073-0745 and 3-TV-073-0746 which introduce a new flow path outside of Primary Containment is necessary to support future surveillance monitoring of the HPCI exhaust system.

<u>Summary of Evaluation:</u>

The proposed modification does not require NRC approval prior to implementation. The primary

containment system is adversely affected by the addition of the new HPCI 3/4" instrument line and test valves 3-TV-073- 0745 and 3-TV-073-0746 which introduce a new flow path outside of Primary Containment. However, the new instrument line and test valves 3-TV-073-0745 and 3-TV-073-0746 do not result in a more than minimal increase in:

- The frequency of occurrence of an accident:
- Likelihood of an SSC failure;
- The consequences of an accident; or,
- The consequences of a malfunction.

Nor does the addition of the new instrument line and test valves 3-TV-073-0745 and 3-TV-073-0746 create or affect:

- A possibility of a different type of accident not previously evaluated;
- A possibility of a malfunction of an SSC important to safety with a different result;
- A design basis limit for a fission product barrier being exceeded or altered; or,
- A method of evaluation described in the UFSAR used to establish the design bases or safety analysis.

The new test valves 3-TV-073-0745 and 3-TV-073-0746 and associated piping are not the initiator of any postulated design basis accidents; are passive components that will have standard post-modification testing in accordance with industry standards and codes to ensure pressure boundary integrity; and, will be normally closed valves that are not required to function during or following a design basis event. Additionally, the consequences of failure of the new test valves 3-TV-073-0745 and 3-TV-073-0746 and associated piping are bounded by those previously evaluated for a Category D accident. A Category D accident results in radioactive release directly to secondary containment with primary containment not intact. The Category D design basis accident is a refueling accident (fuel assembly drops on spent fuel during refueling) (Ref. UFSAR. Section 14.6.1). Based upon the results of this evaluation: Implement the activity per plant procedures without obtaining a License Amendment.

DCN 51057A, Replace U3 Reactor Feedwater Pumps A, B, and C, Revision 0

The activity is the replacement of the feedwater (FW) pumps with pumps of a larger capacity and the upgrade of support systems for the FW pumps. These changes are made to support the higher feedwater flows required to implement Extended Power Uprate (EPU). Although this activity will result in the capacity to produce the higher FW flow rates, the FW flow required during normal operation depends on the power level, which is not changed by this activity.

The upgraded support systems consist of the FW pump instrumentation and the lube oil and seal injection systems. Instrument changes include replacement of thermocouples and vibration monitors, and instrument recalibration, or replacement where necessary, to accommodate the new flow, pressure and temperature conditions. Upgrade to the seal injection system includes two new seal injection pumps and motors to accommodate higher injection flow rates required by the new pumps, and the replacement of piping, valves and the seal water differential pressure controller. Only minor piping changes are made to the lube oil system. The support

systems are not adversely affected by this activity so the evaluation is limited to the effect of the increase in FW pump capacity.

The hydraulic model for the pump and the capability to provide the required bypass flow are documented in calculations. Additional calculations assess the piping associated with the new FW and seal injection pumps and the loads on the foundation of the FW pump.

Summary of Evaluation:

This activity will not affect the frequency or consequences of an accident or malfunction of an SSC, or create the possibility of a new accident or malfunction with a different result. This activity affects the design basis limits for fission product barriers. The analysis of the Feedwater Controller Failure - Maximum Demand (FWCF) abnormal operational transient (AOT) is impacted by this activity because the activity increases the feedwater runout flow. The FWCF is one of the bounding AOTs that are re-evaluated for each core reload and operating cycle to demonstrate that the reactor pressure safety limit and the minimum critical power ratio safety limit (SLMCPR) will not be exceeded. The MCPR operating limits (OLMCPRs) for each reload are obtained from the Supplemental Reload Licensing. Report (SRLR) and assure that the SLMCPR will not be exceeded. The feedwater runout flow is adjusted as needed in each SRLR to reflect updated equipment performance information. Therefore, the process that is required by the UFSAR to be used to establish the operating limits for each reload will assure that this activity will not cause a design basis limit for a fission product barrier to be exceeded.

Therefore, this activity can be implemented without obtaining a license amendment.

SNUBBERS U1/2/3 TRM and BASES, Revision 0

Revise section 3.7.4 of the Browns Ferry Nuclear (BFN) Unit 1, 2, and 3 Technical Requirements Manual (TRM) Requirements and Bases to communicate BFN Snubber Program governance changes from Nuclear Regulatory Commission (NRC) approved relief request from American Society of Mechanical Engineers (ASME) Section XI to the ASME Operating and Maintenance Code, Subsection ISTA and ISTD (Inservice Testing). The surveillance requirements of the TRM that discuss functional testing and visual examinations are therefore updated to align with and reference the new code requirements for snubbers. The majority of other snubber program documents are being revised as well to reflect this change.

Currently, BFN snubber program has an approved relief request from the NRC to perform In-service Inspection and Testing of snubbers in accordance with the station Technical Requirements Manual (TRM) in lieu of Section XI of the ASME B&PV Code.

However, 10 CFR 50.55a rulemaking now states that nuclear station ASME Code updates for In-service examination and testing programs include snubber programs under the ASME Operation and Maintenance Code 2004 edition, with 2005 and 2006 Addenda for all snubber examination and testing, not ASME Section XI.

The current BFN Technical Requirements Manual has two primary functions with respect to snubbers and is intended for use by the BFN Operations organization to: 1.) Provide controlled

information and guidance on snubber operability based on various plant conditions, including actions and limiting condition of operation (LCO) tracking and 2.) Describe Snubber visual examination and functional testing requirements and frequency needed in order to demonstrate snubber operability. The changes to the TRM will only reflect updates to the visual examination / functional testing / surveillance requirements. The ASME OM Code does not contain requirements for snubber or plant system operability or Limiting Condition for Operation (LCO) tracking.

Other than adding definitions, nomenclature, and providing verbiage clarifications of existing passages, which all result in better alignment with the ISTA and ISTD code, some of the more significant changes between the currently NRC approved ASME Section XI relief request and the requirements documented in the ASME OM Code Subsection ISTA and ISTD are related to the following:

Additional Functional Test Sampling: ASME OM Code Subsection ISTD requires that additional functional test samples, those as a result of a functional test failure, shall be at least half of the initial sample of the Design Test Plan Group (DTPG, i.e. Subgroup). TRM 3.7.4 currently requires subsequent test samples to be 10% of the remaining subgroup population. The ASME OM Code Subsection ISTD requirement of a functional test scope expansion sample being at least half of the initial sample of the DTPG will be adopted in the TRM.

Visual Examination Interval Frequency: Table ISTD-4252-1 "Visual Examination Table" content replaces content for TRM Table 3.7.4-1 "Snubber Visual Inspection Interval". The data in the table are equal, however there are some differences in the verbiage of the notes section of the table.

Visual Examination: Visual Inspection criteria attributes will now specifically include the verification of the condition of the overfill vent on certain hydraulic snubbers.

Inservice Examination Failure Evaluations: Snubbers that do not meet visual examination acceptance criteria are required to be classified as unacceptable. This is not a change between requirements of ISTA/ISTD code and the approved relief requests. However, the current TRM requirements state that a snubber reclassified as acceptable, after meeting certain conditions, doesn't get counted for establishing the next visual examination interval. The ISTD code requires initially classified unacceptable snubbers be counted in visual exam interval evaluation, regardless if it was reclassified as acceptable.

The plant transients and design basis accidents previously analyzed that are applicable and potentially affected by this activity are:

Main Steam Isolation Valve (MSIV) Closure, Main Steam Relief Valve (MSRV) inadvertent opening, Shutdown Cooling Malfunction, Loss of Feedwater, Recirculation Pump Trip, Design Basis Earthquake, and Loss of Coolant Accident (LOCA).

Snubber credible failure modes:

Mechanical snubber - Locked in place, high drag, does not activate, damage to snubber hardware

Hydraulic snubber - No lockup, high lockup, low lockup, high bleed, no bleed, miscellaneous

Summary of Evaluation:

The 50.59 screening review indicated the need for this evaluation based on a reduced control of the reliable performance of snubber design functions. Specifically, the transition from ASME Section XI NRC approved relief request to the ASME OM Code, subsection ISTA and ISTD, whereby the BFN Units 1, 2, 3 Technical Requirements and Bases section 3.7.4 for Snubbers is being revised, as well as other BFN Snubber Program procedures and documents, includes adoption of the ASME OM Code provision for selecting 5% of the snubbers in the subject Design Test Plan Group (DTPG) for additional samples as a result of a functional failure, as opposed to the current requirement of selecting 10% of the remaining subgroup population, where it appears that less snubbers would be tested when selecting the 5% for scope expansion. However, it is discussed herein that the method of selecting 1/2 of the initial sample size has been a provision/requirement in the applicable codes and standards since regulation for snubbers was initiated. Based upon the results of this evaluation: Implement the activity per plant procedures without obtaining a License Amendment.

0-AOI-57-1A R71 TN75 & FSAR 8.10.2, Revision 0

This evaluation examines the manual use of MSRVs for controlled depressurization cited by FSAR Section 8.10.2 and the following changes to Revision 71 of 0-AOI-57-1A:

1. Revising the cooldown method of HPCI in pressure control at a rate of <100 deg F/hr starting at 16 minutes to a cooldown with MSRV's at a rate of <90 deg F/hr starting at 1 hour.
2. Modification of the PSP and HCTL Curves for Station Blackout

As discussed in Question 1 of the screening review, the predicted drywell and wetwell pressures and drywell air temperature following an SBO are less than the allowable values, but are lower than previously analyzed. Per NEI 96-07, these increases of margin to the design limits are viewed as un-conservative and thus adversely affect the FSAR described function of Primary Containment.

As discussed in Question 2 of the screening review, this change results in the following adverse effects:

- decease the reliability of an SSC design function with regards to MSRVs
- increase operator burden by substituting the manually initiated operation of HPCI for pressure control mode (essentially auto afterwards) to the manual only operation of SRVs
- increases attention required by the Operator for RPV level and pressure control when using MSRVs
- increases operator burden to manually align CAD to replace depleting compressed gas supply for MSRVs
- involved a procedure not described in the UFSAR which potentially changed how actions related to system operation are to be performed and controls over the performance of

design functions are maintained

Per NEI 96-07, these effects are considered to be an adverse change to how a described function is procedurally controlled.

Summary of Evaluation:

This evaluation concludes that this change does not have more than a minimal adverse impact on the probability or consequences of an accident or malfunction of equipment important to safety. In addition, this activity does not create a new accident or new malfunction of equipment important to safety. Finally, this activity does not result in a design basis limit for a fission product barrier being exceeded or altered and therefore may be performed without prior NRC approval.

DCN 71212A, Replace Existing Obsolete U2 50KV SOLA Type I&C Bus Voltage Regulators, Revision 0

The purpose of this DCN Is to replace existing obsolete I & C Bus Regulating Transformers for Unit 2 (2-XFA-253-0002A2 and 2-XFA-253-0002B2). The I & C Bus Regulating Transformers are safety related components that provide power to 120/208V AC Instrument and Control Power System, System 253. The function of this system is to provide regulated power to vital instruments and controls during normal and design basis events.

As discussed in the Screening Review, the only aspect of this change which is considered "adverse" is the additional time needed for the new transformers to supply regulated output voltage after being energized. There is a short delay before the transformer output is fully regulated (about 22 seconds). The time for the existing transformers to achieve regulated output voltage upon being energized is not known; however, the time is believed to be closer to a few seconds.

This activity is necessary due to BFN U2 experiencing difficulties adjusting the voltage on one of the existing transformers. The transformers are obsolete and spare parts are not readily available. Replacement of the transformers would reduce operational risk due to failure of the transformers. Also, the new transformers would allow for having a spare parts inventory in the event that unforeseen repairs are needed.

Summary of Evaluation:

DCN 71212 does not require prior NRC approval as the design functions of the I &C Bus Regulating Transformers remain the same. The change replaces obsolete equipment with equipment performing the same design function. The replacement transformers do take additional time to provide a regulated output voltage upon being energized. However, the UFSAR does not discuss any specific time requirements with respect to availability of the 120/208V AC Instrument and Control Power System. The events applicable to this change are those associated with a loss of normal auxiliary power. With respect to this change, the bounding malfunction for a loss of normal auxiliary power would be an Initial loss of the 120/208V AC Instrument and Control Power System. This would be true for the existing

transformers as well. The delay to provide regulated output voltage does not prevent any downstream load from performing its design function. This has been evaluated in Appendix 1 of calculation EDQ2000870028. Therefore, the additional time required to achieve regulated output voltage does not impact the likelihood of malfunctions of the 120/208V AC Instrument and Control Power System or malfunctions of the transformers themselves.

The 120/208V AC Instrument and Control Power System is designed as an interruptible power source. The system is designed to account for momentary outages or voltage transients. Therefore, the additional time required to achieve regulated output voltage does not alter the system performance requirements of the 120/208V AC Instrument and Control Power System. The new regulating transformers will allow the 120/208V AC Instrument and Control Power System to perform its design function to support mitigating design basis events, particularly those associated with a loss of normal auxiliary power ("Loss of Coolant Accident", "Events Resulting in a Reactor Vessel Coolant Inventory Decrease" and "Station Blackout Event"). No new failure modes are created by this modification. This modification replaces obsolete equipment with a qualified reliable alternative. Therefore, a UFSAR described design function is not impacted by this change and the change does not require NRC approval.

ECP (DCN) 71666A, U2 Refuel Bridge Upgrades, Revision 0

Engineering Change Package (ECP) or DCN 71666A Refueling Platform Upgrades includes the replacement of two auxiliary hoists on the Refueling Platform. The auxiliary hoists are primarily used to service reactor vessel internals; however, they are also capable of moving new and spent fuel. The two auxiliary hoists consist of a "Monorail Hoist" and a "Frame Mounted Hoist." The existing auxiliary hoist control systems are analog systems that are being replaced with a control system that utilizes digital controls in the form of Variable Frequency Drives (VFDs) due to reliability and obsolescence issues with the current hoists. The use of VFDs in this application increases complexity and potentially creates new malfunctions (NEI 01-01 (Reference G), Section 4.3.3); therefore, the use of VFDs to control the auxiliary hoists is potentially adverse and is the subject of this evaluation. Other changes accomplished by ECP 71666A are the subject of the associated 10 CFR 50.59 screening review.

Summary of Evaluation:

This modification resolves reliability and obsolescence problems with the Refueling Platform Main Hoist VFD and the Refueling Platform auxiliary hoists which consist of the Monorail Hoist and the Frame Mounted Hoist. The auxiliary hoist upgrades included the use of VFDs which constituted analog-to-digital controls system upgrades. Refueling Platform operation, as described in the UFSAR, remains unchanged. The design functions, interlocks and other safety features of the Refueling Platform are retained and are not adversely impacted by the proposed activity. The modification incorporates improvements in the man-machine interface of the auxiliary hoist operating pendants which result in a reduction in human errors and consequently an improvement in safety.

In order to ensure that the 50.59 evaluation sufficiently addresses issues related to digital upgrades, the questions in Appendix A of NEI 01.01 (Reference G) were listed and answered

with the corresponding 50.59 evaluation question.

Digital device quality assurance measures included the performance of a critical digital review and a failure modes and effects analysis (FMEA) for the replacement controls (i.e., VFDs) and concluded that the use of digital controls in the auxiliary hoist VFD upgrades does not increase the probability or the consequences of any Refueling Accident described in the UFSAR, nor will it introduce the possibility of a new type of accident not previously considered. The assumptions and conclusions of the existing Refueling Accident analyses remain valid. No reductions in the existing margins of safety are created by implementing this modification. The existing shutdown margins during refueling operations are maintained. The existing submergence limits and radiation shielding margins are retained. No changes to the Technical Specifications or their Bases are required. Based upon the results of this evaluation: Implement the activity per plant procedures without obtaining a License Amendment.

DCN 71634A, Replace U3 Time Delay Relays for Loops 3-P-3-204 & 3-L-3-208, Revision A

CR 881051 identified that during a functional test of Unit 3 level instrumentation, the Anticipated Transient Without a Scram Reactor Protection Trip (ATWS-RPT) unexpectedly actuated resulting in a dual Reactor Recirculation Pump trip from 100% power and insertion of all control rods, challenging nuclear safety and safety systems. The initiation occurred as 3-LS-3-58A1 (REACTOR WATER LEVEL A INTERLOK) was being calibrated by simulating a lowering of reactor water level. Contrary to the test design, once the setpoint was reached ATWS initiation occurred. As a result, the scram air header was depressurized through the ATWS/ARI valves, causing all rods to insert into the core. The ATWS/ARI signal also simultaneously opened the Recirculation Pump Trip (RPT) breakers, tripping both pumps. The loss of both Recirculation pumps along with reduced core flow caused a reactor water level transient that lowered level below the RPS trip setpoint (+ 2 inches), resulting in a full reactor scram signal.

Corrective actions were developed as a result of Root Cause Analysis 881051 to install time delay relays for loops P-3-204 and L-3-208 on Units 1, 2, and 3 to prevent recurrence of a spurious, actuation of the ATWS initiation relay and allow short-lived transients to bypass without ATWS initiation. DCN 71634 will replace the instantaneous relays on Units 1, 2, and 3 loop P-3-204 (REACTOR PRESSURE ATWS) with time delay relays to prevent unexpected ATWS initiation due to a spurious voltage transient from any source during testing or under normal conditions.

Summary of Evaluation:

The proposed modification provides for a time delay of 0.20 sec for Units 1/2/3 loop P-3-204.

The specified duration of this time delay will differ due to Unit differences in sense lines. A comparative analysis of the loop response times for the specific units are listed below:

Unit 1: (P-3-204)

Maximum loop response time = 0.75 sec

Existing loop response time (Unit 1) = Transmitter (200ms) + Pressure switch (10ms) + Sensing lines (29ms) + Instrument relay response (30ms)

Existing loop response time (U1) = 0.269 sec

New Loop response time (Unit 1) = Transmitter (200ms) + Pressure switch (10ms) + Sensing lines (29ms) + Maximum relay uncertainty (34ms)+ time delay relay setting (200ms)

New loop response time (U1) = 0.473 sec

New Margin (based on new loop response time) = 0.750 sec - 0.473 sec = 0.277 sec

Unit 2 and Unit 3: (P-3-204)

Maximum loop response time = 0.75 sec

Existing loop response time (Unit 2 and 3) = Transmitter (200ms) + Pressure switch (10ms) + Sensing lines (60ms) + Instrument relay response (30ms)

Existing loop response time = 0.300 sec

New Loop response time (Unit 2 and 3) = Transmitter (200ms) + Pressure switch (10ms) + Sensing lines (60ms) + Maximum relay uncertainty (34ms) + time delay relay setting (200ms)

New loop response time (Unit 2 and 3) = 0.504sec

New Margin (based on new loop response time) = 0.750 sec - 0.504 sec = 0.246 sec

Currently, GE's OPL-3 Design Guide, provides for a maximum loop response time of 0.5 seconds as a typical value for the pressure sensor response time for the ATWS-RPT (high pressure). However, Reactor Engineering and Fuels have updated the transient analysis inputs for the upcoming Unit 3 Cycle 18 to reflect a maximum of 0.75 second delay for

loop (P-3-204) response time. The transient analysis for Units 1 and 2 will be performed during the next reload cycles. The proposed new configuration (time delays), based on the above evaluation, is safe in BFN's Power Uprate configuration. The only limitation is that this modification can only be implemented during a refueling outage.

Likewise, in regards to Extended Power Uprate (EPU), Task Report T0902 was revised and the EPU LAR was submitted to the NRC to reflect a maximum 0.75 second delay for loop P-3-204. In addition, AREVA has provided an analysis which demonstrates that the new configuration is safe in an Extended Rower Uprate configuration and the 0.75 second delay will not exceed vessel overpressure of 1500 psig.

NEI 96-07 was reviewed and used as guidance for this evaluation. Additionally, NEI 01-01 was reviewed; however, the proposed modification does not involve any digital components; therefore, questions from NEI 01-01 are not applicable in this review.

Based upon the results of this evaluation: Implement the activity per plant procedures without obtaining a License Amendment.

EOI Program Manual 0-TOC R85, Revision 0

This 50.59 evaluation addresses changes being made to BFN Emergency Operating Instructions (EOIs) developed from BWROG EPG/SAG Rev. 3. The changes are as follows:

Change 14 authorizes disablement of the RCIC EG-M during an extended SBO (ELAP) event. In accordance with BWROG RCIC Operation in Prolonged Station Blackout-Feasibility Study, DRF 0000-0143-0389, the RCIC EG-M is subject to failure at temperatures above 150 deg. F. This change removes the fuses to the EG-M, preventing an unexpected low output failure of the device, and allows RCIC turbine speed to be controlled by the operator manually by throttling of the Trip/Throttle valve. UFSAR section 4.7.5 states "The turbine control system is positioned by the demand signal from a flow controller, and satisfies a twofold purpose: a) Limit the turbine pump speed to its maximum value, and b) Position the turbine governor valve as required to maintain a constant pump discharge flow over the pressure range of operation." The change permits disabling of this control. However, during an extended SBO when RCIC may be the only injection source, it is desirable to keep RCIC in service and not challenge it by potential EG-M failure. This disablement would not be needed during a design basis SBO event.

Change 15 authorizes operation of HPCI without use of the Steam Packing Exhauster and in "batch" mode of operation where HPCI is manually tripped on high RPV water level, and then locked out until level lowers to -45 in. where it is then manually started. These actions are only performed during an extended SBO with RCIC unavailable, and are intended to extend the life of the batteries. UFSAR section 6.4.1 states "The HPCI starts when the water level reaches a pre-selected height above the core, or if high pressure exists in the primary containment (drywell). The HPCI automatically stops when a high water level in the reactor vessel is signaled. The HPCI automatically resets and will restart if vessel water level again reaches a pre-selected height above the core." Also, it further states "Noncondensable gases from the gland seal condenser are pumped to the Standby Gas Treatment System." During an extended SBO event, the Standby Gas Treatment System is unavailable, and operation of the Steam Packing Exhauster serves no purpose. During an extended SBO event, when RCIC is unavailable and HPCI is the only injection source, it is desirable to keep the batteries available in order to maintain HPCI in service. These steps would not be needed during a design basis SBO event, nor in a beyond design basis event if RCIC were available.

Summary of Evaluation:

The changes above to the EOIs are evaluated against design basis accidents and events and against UFSAR and specific SERs (EPG Rev. 4 SER) and demonstrate that the proposed changes are within the bases. The changes are shown to be appropriate for the beyond design basis conditions which they are addressing (extended loss of AC power and loss of ultimate heat sink). Finally it is shown that they do not require prior approval by NRC to implement since the changes are for beyond design basis events and, as applicable, meet the requirements of the Rev 4 SER. Based upon the results of this evaluation: Implement the activity per plant procedures without obtaining a License Amendment.

ENCLOSURE 1

Tennessee Valley Authority
Browns Ferry Nuclear Plant
Units 1, 2, and 3

10 CFR 50.59(d)(2) Summary Report

DCN 51079A, U1 Recovery Control Bay-I&C Lead DCN-System 092, Revision 2

DCN 51079 will replace the Average Power Range Monitor (APRM), Rod Block Monitor (RBM) and Local Power Range Monitor (LPRM) electronic components in panel 9-14 in the Unit 1 control room with digital Nuclear Measurement, Analysis, and Control (NUMAC) PRNM modules. The new system adds the capability for an oscillation power range monitor (OPRM) automatic trip function to detect and suppress possible thermal hydraulic instabilities. DCN 51079 is staged so that the PRNM setpoints are initially established to support operation at 105% of original licensed power (3458 MWt). Upon approval of the Unit 1 EPU Technical Specifications change, the PRNM setpoints will be changed to their EPU values to support operation at 120% of original licensed power (3952 MWt). This DCN also implements the APRM and RBM Technical Specification (ARTS) improvements and operation in an expanded core power/flow domain, the Maximum Extended Load Line Limit (MELLL) region. RBM modifications and APRM setpoint changes required to implement ARTS/MELLL, and 24 month cycle are included in the NUMAC PRNM design. The Browns Ferry Unit 1Technical Requirements Manual and Bases are revised by DCN 51079 to incorporate these changes to the RBM instrumentation.

The scope of this 10 CFR 50.59 Evaluation is limited to the aspects of the modification that has the potential to adversely affect an UFSAR described design function as identified in the preceding 10CFR50.59 Screening Review. Specifically the scope of this 10 CFR 50.59 Evaluation is the PRNM system function changes along with the incorporation of digital software and the addition of the PCIS seal-in relay and reset hand switch which adds new components in the TIP PCIS logic and adds a new operator action to reset the TIP isolation signal.

Summary of Evaluation

The changes do not impact the initiation of any accidents. The effects have been analyzed and determined not to exceed PCT limits and therefore dose consequences of a LOCA are not affected. No new failure modes are created and therefore, no possibility of a new accident or malfunction with a different result is created. No new or changed analysis methods were used in determining the acceptability of the modifications. Following the NRC approval of the technical specification changed requested by References 18, 19, 20, and 21 this change may be implemented without additional NRC approval.

DCN 71390A, Modify Hardened Containment Vent System To Comply with NRC ORDER EA-13-109 U2, Revision 0

Following the March 11, 2011, Fukushima Dai-ichi accident in Japan, which resulted from a beyond design basis tsunami, producing severe damage to three (3) of the site's six (6) boiling water reactors (BWRs), NRC Order EA-12-050 was issued on March 12, 2012, to U.S. licensed BWRs requiring licensees to implement requirements for reliable hardened containment vents at their facilities. NRC Order EA-12-050 notes that hardened containment vents have been in place in the U.S. Mark I and Mark II BWR containments for many years as the result of previous NRC regulatory action contained in Generic Letter 89-16, while recognizing that variances exist among U.S. BWRs regarding the capability to vent containment during a broad spectrum of

events. The conclusion of NRC Order EA-12-050 is that on the basis of the need to enhance the health and safety of the public, a reliable Hardened Containment Venting System (HCVS) for Mark I and Mark II containments is mandated for all BWR license holders. Specifically, NRC Order EA-12-050 directed BWR licensees to add qualification features to the hardened vent installations that were originally performed in response to NRC Generic Letter 89-16. For Browns Ferry (BFN) Units 1, 2, and 3, a system similar to the HCVS described in NRC Order EA-12-050, termed the Hardened Wetwell Vent (HWWV), was installed in response to NRC Generic Letter 89-16.

On June 6, 2013, the NRC issued Order EA-13-109 to rescind and replace the previous NRC Order EA-12-050. Phase I of NRC Order EA-13-109 implementation requires installation of an HCVS that will remain functional during severe accident conditions (i.e., when core damage has occurred). The primary design objective of the HCVS is to provide sufficient venting capacity to prevent a long-term overpressure failure of the containment by restoration and maintenance of containment pressure below the Primary Containment design pressure and the Primary Containment Pressure Limit. On June 30, 2014, the Licensee (TVA) communicated its planned commitment for installing the HCVS at BFN by submitting an Overall Integrated Plan (OIP) in its response to NRC Order EA-13-109 utilizing nuclear industry guidance provided by NEI 13-02 that is acceptable to the NRC for licensees to use in meeting the requirements of NRC Order EA-13-109. TVA submitted to the NRC a 6-month update to the OIP on December 19, 2014, and on February 11, 2015, the NRC issued an Interim Staff Evaluation relating to the design and ongoing implementation of Order EA-13-109 at BFN. The second, third, and fourth 6-month updates to the OIP were submitted on June 29, 2015, December 29, 2015, and June 30, 2016, respectively.

Using the TVA commitments in the OIP and subsequent communications as a basis, DCN 71390 documents the design to implement a reliable, severe accident capable, HCVS for Unit 2 at BFN. Separate DCNs will be prepared to implement the similar HCVS for BFN Units 1 and 3. When implemented, the Unit 2 HCVS will comply with Phase I requirements of NRC Order EA-13-109 along with the appropriate site guidance.

Portions of the existing HWWV that were installed in response to NRC Generic Letter 89-16 will be used in the overall HCVS that is implemented by DCN 71390. The HWWV provides a direct vent path from the wetwell (also referred to as the torus) to an exhaust point inside the concrete portion of the existing plant stack above elevation 666.5'. The original flow path utilized a 14 inch line tapping off of the 20 inch torus vacuum breaker piping at Reactor Building elevation 565'. Two 14 inch pneumatically operated PCIVs (2- FCV-064-0221 and 2-FCV-064-0222) are located within the section of HWWV piping inside the Reactor Building. The existing HWWV penetrates the Reactor Building wall along column line U at a location approximately 6' east of column line R10. After exiting the Reactor Building, the current pipe enters a valve pit where 2-SHV-064-0737 is located and then exits the valve pit tying into a common header for all three Units which routes to the existing plant stack. The existing HWWV in its original configuration is not adequate for meeting the newest regulatory guidance due to new requirements for Unit separation, effluent release point, the need to address flammable gasses, and requirements to function during a severe accident when core damage has

occurred. To meet the latest regulatory requirements a new vent path will be established specifically for each unit, however, DCN 71390 only addresses Unit 2. Portions of the Unit 2 HWWV interior to the Reactor Building are adequate for reuse.

The existing HWWV piping will be rerouted underground, in a valve pit, just south of Reactor Building column line U. Following permanent removal of 2-SHV-064-0737, new HCVS piping will elbow up at the flange location previously upstream of 2-SHV-064-0737 traveling vertically up the exterior wall of the Reactor Building. The piping will continue up the wall of the Reactor Building until it reaches the approximate elevation of 665', where it will turn and penetrate the siding of the Reactor Building superstructure. Once inside the superstructure the piping will be routed vertically adjacent to an existing steel column until it penetrates the roof of the superstructure, ultimately terminating at the approximate elevation of 741'-6". The portion of piping previously downstream of 2-SHV-064-0737 (which ties to the common HWWV header) will be cut and capped as close as possible to the South wall of the valve pit penetration such that remaining HWWV common header piping will remain functional for Unit 3.

Altering the existing HWWV piping configuration as described for implementation of HCVS changes the original release point from inside the 600' tall plant stack (at approximate elevation 666.5') to a location above the Unit 2 Reactor Building at approximate elevation 741'-6". The change in release point not only impacts beyond design basis venting scenarios which utilize HCVS but also alters the release point for a design basis Loss of Coolant Accident (LOCA) as described in Section 14.6.3.5 of the UFSAR. As described in 14.6.3.5.e of the UFSAR, releases occur as a result of leakage past existing HWWV Primary Containment Isolation Valves (PCIVs) (2-FCV-064-0221 and 2-FCV-064-0222). By altering the original HWWV release point for HCVS to a location near the Unit 2 Reactor Building Exhaust Fans the UFSAR described design function of the Primary Containment System is adversely impacted. Specifically the Primary Containment System is required to limit leakage during and following a postulated LOCA to values which are substantially less than leakage rates that would result in offsite doses approaching 10 CFR 50.67 limits. This change has the potential to increase the consequences of a design basis LOCA.

Summary of Evaluation:

After evaluation, the changes required to implement HCVS do not:

- increase in the frequency of occurrence of an accident previously evaluated in the UFSAR,
- increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR,
- result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR,
- create a possibility for an accident of a different type than any previously evaluated in the UFSAR,
- create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in UFSAR,

- result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered,
- result in a departure from a method of evaluation described in the UFSAR used in establishing the design bases or in the safety analyses.

Implementation of HCVS does increase the consequences of an accident previously evaluated in the UFSAR, however, when evaluated against the criteria established by Section 4.3.3 of NEI 96-07 the increase is not more than minimal. Based upon the results of this evaluation: Implement the activity per plant procedures without obtaining a License Amendment.

TVA-COLR-BF2C19, Revision 5

A core operating limits report is generated each cycle as a result of updated safety analyses. Cycle specific reload licensing safety analyses results are summarized in Reference 13; the report is added to FSAR, Appendix N. Cycle specific operating limits for the core design are identified in the new COLR, per Reference 1.

The new core is similar to the previous cycle: it is designed for approximately 24 months of calendar duration, at 105% OLTP (3458 MWt). The core is licensed for operation over the MELLLA flow range of 81% to 105% (ICF) at rated power conditions. Lower flows, at reduced powers, are available along the MELLLA rod line. Core flow above 105% rated flow is analytically allowed by following the constant pump speed line during coastdown from the 100%P/105%F point.

The Unit 2 Cycle 19 will consist of ATRIUM-10XM, 8 lead use ATRIUM-11, and legacy ATRIUM-10 fuel assemblies. The ATRIUM-10 was generically evaluated per References 3 & 4, ATRIUM-10XM per Reference 6, and ATRIUM-11 per Reference 7. Fresh ATRIUM-10XM fuel will utilize BLEU material; however, ATRIUM-11 will only utilize CGU material. BLEU material was evaluated per References 3, 5, & 6.

The core is licensed for RPTOOS, TBVOOS, FHOOS, PLUOOS, and Dual or Single Recirculation Loop operation, with appropriate thermal limits provided. Out-Of-Service options may be combined, as long as appropriate limit sets are implemented. FFTR and ICF may be used in any order for cycle extension. ICF and FHOOS are available at any point in the cycle.

CPR limits are provided for NEOC, EOC, and FFTR/Coastdown exposure windows. NEOC limits may be used until a core average exposure of 30,910.3 MWd/MTU is reached. EOC limits may be used at any point in the cycle up to a maximum core average exposure of 33,210.3 MWd/MTU. FFTR/Coastdown limits may be used at any point in the cycle up to a maximum core average exposure of 34,624.4 MWd/MTU.

Pressurization transients use a minimum power for operability for the Power Load Unbalance (PLU) feature of the Digital EHC system. The PLU will generate a TCV fast closure for a load rejection at or above 40% generator power. Reload analyses assume PLU operation above 50% core thermal power to conservatively account for efficiency loss when operating at low power conditions with FHOOS.

The OPRM system will be armed with cycle specific setpoints provided in Reference 13. Setpoints are based upon cycle specific DIVOM analyses. Backup stability protection regions have been provided for periods when the primary OPRM system is determined to be inoperable. The regions have been confirmed to be bounding for the cycle, or expanded as necessary.

The Westinghouse CR82M-1 blade design will be used for any replacements. The blade type is currently in use at all three BFN units. The NA-300 LPRM design will be used for any planned replacements. The LPRM type is currently in use at all three BFN units.

Summary of Evaluation:

The core design uses ATRIUM-10XM, 8 lead use ATRIUM-11, and legacy ATRIUM-10 fuel. This is the first reload of ATRIUM-10XM and ATRIUM-11 LUAs for Unit 2. ATRIUM-10XM and legacy ATRIUM-10 fuel will utilize BLEU material. BLEU material is currently used in Units 1, 2, & 3. ATRIUM-11 lead use assemblies will utilize CGU only. Control Blade replacements use the previously approved Westinghouse CR82M-1 design. LPRM replacements will be made using the previously approved NA-300 design. All potential changes to the core and core components used in this reload have been evaluated and approved for use.

Cycle specific reload evaluations have been performed as required to establish the cycle specific core operating limits supplied in the COLR. Operation of the core within required limits will ensure the fuel performs without fuel failures expected for normal operation and releases within 10 CFR 20 guidelines for AOTs, and within 10 CFR 50.67 guidelines for DBAs. All reload analyses were determined to have been performed in accordance with approved methodologies. The use of ATRIUM-11 LUAs requires meeting the Technical Specification limits of section 4.2.1. Lead use assemblies are allowed in limited number and in non-limiting locations. The use of eight ATRIUM-11 LUAs is approximately 1% of the total core load which meets the intent of limited number of assemblies. The U2C19 core design shows that all of the ATRIUM-11 LUAs are loaded into non limiting locations. Based upon the results of this evaluation: Implement the activity per plant procedures without obtaining a License Amendment.

TVA-COLR-BF3C18, Revision 1 (BFE-4020, Revision 0)

A core operating limits report is generated each cycle as a result of updated safety analyses. Cycle specific reload licensing safety analyses results are summarized in Reference 11; the report is added to FSAR, Appendix N. Cycle specific operating limits for the core design are identified in the new COLR, per Reference 1.

The new core is similar to the previous cycle: it is designed for approximately 24 months of calendar duration, at 105% OLTP (3458 MWt). The core is licensed for operation over the MELLLA flow range of 81% to 105% (ICF) at rated power conditions. Lower flows, at reduced powers, are available along the MELLLA rod line. Core flow above 105% rated flow is analytically allowed by following the constant pump speed line during coastdown from the 100%P/105%F point.

ENCLOSURE 1

Tennessee Valley Authority
Browns Ferry Nuclear Plant
Units 1, 2, and 3

10 CFR 50.59(d)(2) Summary Report

The Unit 3 Cycle 18 will consist of ATRIUM-10XM and legacy ATRIUM-10 fuel assemblies. The ATRIUM-10 was generically evaluated per References 3 & 4, ATRIUM-10XM per Reference 6. Fresh ATRIUM-10XM fuel will utilize BLEU material. BLEU material was evaluated per References 3, 5, & 6.

The core is licensed for RPTOOS, TBVOOS, FHOOS, PLUOOS, and Dual or Single Recirculation Loop operation, with appropriate thermal limits provided in Reference 1. Out-Of-Service options may be combined, as long as appropriate limit sets are implemented. FFTR and ICF may be used in any order for cycle extension. ICF and FHOOS are available at any point in the cycle.

CPR limits are provided to the NEOC exposure window. NEOC limits may be used until a core average exposure of 28,825.7 MWd/MTU is reached. (Limits in the subject COLR are only valid for cycle operation to NEOC. A future revision to the COLR will be necessary to take advantage of the full energy capability of the core design).

Pressurization transients use a minimum power for operability for the Power Load Unbalance (PLU) feature of the Digital EHC system. The PLU will generate a TCV fast closure for a load rejection at or above 40% generator power. Reload analyses assume PLU operation above 50% core thermal power to conservatively account for efficiency loss when operating at low power conditions with FHOOS.

The OPRM system will be armed with cycle specific setpoints provided in Reference 11. Setpoints are based upon cycle specific DIVOM analyses. Backup stability protection regions have been provided for periods when the primary OPRM system is determined to be inoperable. The regions have been confirmed to be bounding for the cycle, or expanded as necessary.

Control Blade replacements, if any, use the previously approved Westinghouse CR82M-1 and GEH Ultra designs. This is the first application of the new GEH Ultra blade types under the latest procurement contract. The NA-300 LPRM design will be used for any planned replacements. The LPRM type is currently in use at all three BFN units.

Summary of Evaluation:

The core design uses ATRIUM-10XM and legacy ATRIUM-10 fuel. This is the first reload of ATRIUM-10XM for Unit 3. ATRIUM-10XM and legacy ATRIUM-10 fuel will utilize BLEU material. BLEU material is currently used in Units 1, 2, & 3. Control Blade replacements use the previously approved Westinghouse CR82M-1 and GEH Ultra designs. LPRM replacements will be made using the previously approved NA-300 design. All potential changes to the core and core components used in this reload have been evaluated and approved for use.

Cycle specific reload evaluations have been performed as required to establish the cycle specific core operating limits supplied in the COLR. Operation of the core within required limits will ensure the fuel performs without fuel failures expected for normal operation and releases within 10 CFR 20 guidelines for AOTs, and within 10 CFR 50.67 guidelines for DBAs. All reload analyses were determined to have been performed in accordance with approved methodologies. Based upon the results of this evaluation: Implement the activity per plant procedures without obtaining a License Amendment.

Tennessee Valley Authority
Browns Ferry Nuclear Plant
Units 1, 2, and 3

10 CFR 50.59(d)(2) Summary Report

TVA-COLR-BF1C12, Revision 0

A core operating limits report is generated each cycle as a result of updated safety analyses. Cycle specific reload licensing safety analyses results are summarized in Reference 11; the report is added to FSAR, Appendix N. Cycle specific operating limits for the core design are identified in the new COLR, per Reference 1.

The new core is similar to the previous cycle: it is designed for approximately 24 months of calendar duration, at 105% OLTP (3458 MWt). The core is licensed for operation over the MELLLA flow range of 81% to 105% (ICF) at rated power conditions. Lower flows, at reduced powers, are available along the MELLLA rod line. Core flow above 105% rated flow is analytically allowed by following the constant pump speed line during coastdown from the 100%P/105%F point.

The Unit 1 Cycle 12 will consist of ATRIUM-10XM and legacy ATRIUM-10 fuel assemblies. The ATRIUM-10 was generically evaluated per References 3 & 4, ATRIUM-10XM per Reference 6. Fresh ATRIUM-10XM fuel will utilize mix of BLEU & CGU material. BLEU material was evaluated per References 3, 5, & 6.

The core is licensed for RPTOOS, TBVOOS, FHOOS, PLUOOS, and Dual or Single Recirculation Loop operation, with appropriate thermal limits provided in Reference 1. Out-Of-Service options may be combined, as long as appropriate limit sets are implemented. FFTR and ICF may be used in any order for cycle extension. ICF and FHOOS are available at any point in the cycle.

CPR limits are provided for NEOC, EOC, and FFTR/Coastdown exposure windows. NEOC limits may be used until a core average exposure of 29,949.1 MWd/MTU is reached. EOC limits may be used at any point in the cycle up to a maximum core average exposure of 33,699.9 MWd/MTU. FFTR/Coastdown limits may be used at any point in the cycle up to a maximum core average exposure of 35,231.8 MWd/MTU.

Pressurization transients use a minimum power for operability for the Power Load Unbalance (PLU) feature of the Digital EHC system. The PLU will generate a TCV fast closure for a load rejection at or above 40% generator power. Reload analyses assume PLU operation above 50% core thermal power to conservatively account for efficiency loss when operating at low power conditions with FHOOS.

The OPRM system will be armed with cycle specific setpoints provided in Reference 11. Setpoints are based upon cycle specific DIVOM analyses. Backup stability protection regions have been provided for periods when the primary OPRM system is determined to be inoperable. The regions have been confirmed to be bounding for the cycle, or expanded as necessary.

Control Blade replacements, if any, use the previously approved Westinghouse CR82M-1 and GEH Ultra designs. This is the first application of the new GEH Ultra blade types, for Unit 1, under the latest procurement contract. The NA-300 LPRM design will be used for any planned replacements. The LPRM type is currently in use at all three BFN units.

ENCLOSURE 1

Tennessee Valley Authority
Browns Ferry Nuclear Plant
Units 1, 2, and 3

10 CFR 50.59(d)(2) Summary Report

Summary of Evaluation:

The core design uses ATRIUM-10XM and legacy ATRIUM-10 fuel. This is the first reload of ATRIUM-10XM for Unit 1. ATRIUM-10XM and legacy ATRIUM-10 fuel will utilize a mix of BLEU & CGU material. BLEU material is currently used in Units 1, 2, & 3. Control Blade replacements use the previously approved Westinghouse CR82M-1 and GEH Ultra designs. LPRM replacements will be made using the previously approved NA-300 design. All potential changes to the core and core components used in this reload have been evaluated and approved for use.

Cycle specific reload evaluations have been performed as required to establish the cycle specific core operating limits supplied in the COLR. Operation of the core within required limits will ensure the fuel performs without fuel failures expected for normal operation and releases within 10 CFR 20 guidelines for AOTs, and within 10 CFR 50.67 guidelines for DBAs. All reload analyses were determined to have been performed in accordance with approved methodologies. Based upon the results of this evaluation: Implement the activity per plant procedures without obtaining a License Amendment.

TVA-COLR-BF2C20, Revision 0

A core operating limits report is generated each cycle as a result of updated safety analyses. Cycle specific reload licensing safety analyses results are summarized in Reference 14; the report is added to FSAR, Appendix N. Cycle specific operating limits for the core design are identified in the new COLR, per Reference 1.

The new core is similar to the previous cycle: it is designed for approximately 24 months of calendar duration, at 105% OLTP (3458 MWt). The core is licensed for operation over the MELLLA flow range of 81% to 105% (ICF) at rated power conditions. Lower flows, at reduced powers, are available along the MELLLA rod line. Core flow above 105% rated flow is analytically allowed by following the constant pump speed line during coastdown from the 100%P/105%F point.

The Unit 2 Cycle 20 will consist of ATRIUM-10XM, lead use ATRIUM-11, and legacy ATRIUM-10 fuel assemblies. The ATRIUM-10 was generically evaluated per References 3 & 4, ATRIUM-10XM per Reference 6, and ATRIUM-11 per Reference 7. Fresh ATRIUM-10XM fuel will utilize BLEU & CGU material; legacy ATRIUM-10 fuel is BLEU; ATRIUM-11 only utilizes CGU material. BLEU material was evaluated per References 3, 5, & 6.

The core is licensed for RPTOOS, TBVOOS, FHOOS, PLUOOS, and Dual or Single Recirculation Loop operation, with appropriate thermal limits provided in Reference 1. Out-Of-Service options may be combined, as long as appropriate limit sets are implemented. FFTR and ICF may be used in any order for cycle extension. ICF and FHOOS are available at any point in the cycle.

CPR limits are provided for NEOC, EOC, and FFTR/Coastdown exposure windows. NEOC limits may be used until a core average exposure of 31,004.5 MWd/MTU is reached. EOC limits may be used at any point in the cycle up to a maximum core average exposure of 34,274.5 MWd/MTU. FFTR/Coastdown limits may be used at any point in the cycle up to a

maximum core average exposure of 35,793.3 MWd/MTU.

Pressurization transients use a minimum power for operability for the Power Load Unbalance (PLU) feature of the Digital EHC system. The PLU will generate a TCV fast closure for a load rejection at or above 40% generator power. Reload analyses assume PLU operation above 50% core thermal power to conservatively account for efficiency loss when operating at low power conditions with FHOOS.

The OPRM system will be armed with cycle specific setpoints provided in Reference 14. Setpoints are based upon cycle specific DIVOM analyses. Backup stability protection regions have been provided for periods when the primary OPRM system is determined to be inoperable. The regions have been confirmed to be bounding for the cycle, or expanded as necessary.

The GE ULTRA HD and ULTRA MD blade designs will be used for any replacements; CR82M-1 is also acceptable. The blade types are currently in use at all three BFN units. The NA-300 LPRM design will be used for any planned replacements. The LPRM type is currently in use at all three BFN units.

Summary of Evaluation:

The core design uses ATRIUM-10XM, ATRIUM-11 LUAs, and legacy ATRIUM-10 fuel. This is the second reload of ATRIUM-10XM for Unit 2. Legacy ATRIUM-10 and ATRIUM-10XM fuel utilizes BLEU material. Fresh ATRIUM-10XM bundles will be a mix of BLEU and CGU; ATRIUM-11 lead use assemblies utilize CGU only. Control Blade replacements use the previously approved GE ULTRA HD/MD or Westinghouse CR82M-1 designs. LPRM replacements will be made using the previously approved NA-300 design. All potential changes to the core and core components used in this reload have been evaluated and approved for use.

Cycle specific reload evaluations have been performed as required to establish the cycle specific core operating limits supplied in the COLR. Operation of the core within required limits will ensure the fuel performs without fuel failures expected for normal operation and releases within 10 CFR 20 guidelines for AOTs, and within 10 CFR 50.67 guidelines for DBAs. All reload analyses were determined to have been performed in accordance with approved methodologies. The use of ATRIUM-11 LUAs requires meeting the Technical Specification limits of section 4.2.1. Lead use assemblies are allowed in limited number and in non-limiting locations. The use of six ATRIUM-11 LUAs is less than 1% of the total core load meeting the intent of limited number of assemblies. The U2C20 core design shows ATRIUM-11 LUAs are loaded into non limiting locations. Based upon the results of this evaluation: Implement the activity per plant procedures without obtaining a License Amendment.

TVA-COLR-BF3C17, Revision 2 (BFE-3731, Revision 1)

A core operating limits report (COLR) is generated each cycle as a result of updated safety analyses. In response to PER 833515, cycle specific safety evaluations have been performed for the SLO pump seizure event. The Screening Review (SR) / Safety Evaluation (SE) address revisions to the Browns Ferry Unit 2 and Unit 3 COLRs. Cycle specific reload licensing safety analyses results are summarized in Reference 3 & 6;

the reports are added to FSAR, Appendix N. Cycle specific operating limits are provided in the new COLRs, per References 1 & 4.

One of the anticipated operational occurrences for BFN is the seizure of a recirculation pump. The pump seizure event is evaluated in the UFSAR for two loop operation, where it is shown to be a benign event that does not set core operating limits; therefore it is not evaluated on a cycle specific basis.

The reload safety analyses have been revised to incorporate the recirculation pump seizure event in combination with one recirculation pump out of service (RCPOOS). All other safety analyses remain unchanged. The nature of the change is such that there are no differences in implementation between BFN Unit 2 and BFN Unit 3; as such, both Units will be addressed together. Unless otherwise noted, the following discussion applies to both Unit 2 and Unit 3.

The Unit 3 Cycle 17 COLR is being revised to incorporate a power coastdown cycle exposure extension. The extension has been assessed by the fuel vendor and is discussed per Reference 31.

Summary of Evaluation:

The pump seizure event analyzed here assumes the reactor is operating with one recirculation pump inactive and an instantaneous seizure of the pump motor shaft of the active recirculation pump occurs. Analyses assumptions were constructed to achieve a balance between operating flexibility and margin to thermal limits. Maximum core power and flow are restricted to 50% of rated conditions and active recirculation drive flow is assumed to be ≤ 17.73 Mlb/hr.

Two loop operation transient results remain applicable. SLO MCRP operating limits are a combination of SLO pump seizure analysis results and a SLO MCPR adjustment adder applied to the corresponding two-loop operating limits.

SLO LHGR and SLO MAPLHGR operating limits remain unaffected by the SLO pump seizure event.

Cycle specific safety evaluations have been performed to revise the SLO operating limits supplied in the COLR to include SLO pump seizure. Operation of the core within required limits will ensure the fuel performs without fuel failures expected for normal operation and releases within 10 CFR 20 guidelines for AOTs, and within 10 CFR 50.67 guidelines for DBAs. All analyses and computer code updates were determined to have been performed in accordance with previously approved methodology.

The fuel vendor has assessed the continuing applicability of Reference 6 for an increased CAVEX limit of 33,000 MWd/MTU; documenting the disposition per Reference 31. The value of 32,724.6 MWd/MTU occurs under power coastdown conditions. The small increase to maximum CAVEX occurs during a time when core reactivity is decreasing. The decreased reactivity limits the magnitude of peak power during transients and accidents. In general, this situation means margins to thermal limits are increasing as CAVEX increases. Based upon the results of this evaluation: Implement the activity per plant procedures without obtaining a License Amendment.

ENCLOSURE 1

Tennessee Valley Authority
Browns Ferry Nuclear Plant
Units 1, 2, and 3

10 CFR 50.59(d)(2) Summary Report

BFE-4076-R0 UFSAR Update, Revision 0

The proposed activity is to decrease the assumed number of missing Boral plates in the normal storage condition from four (4) plates per rack to zero (0). In the accident condition the number of assumed missing Boral plates is changed from four (4) to one (1). The change is based on actual testing and manufacturing documentation showing there is a 95% probability at 95% confidence that all plates are present in the spent fuel storage pool.

Summary of Evaluation

The altered assumption was implemented in 2003 based on the TS 421 LAR approval.
The Framatome ANP criticality analyses show the assumptions meet the 10 CFR 50.68, and Technical Specification requirements. Based upon the results of this evaluation: Implement the activity per plant procedures without obtaining a License Amendment.

ENCLOSURE 2

Tennessee Valley Authority
Browns Ferry Nuclear Plant
Units 1, 2, and 3
Technical Specifications (TS) Bases Changes and Additions

Unit 1 Technical Specifications Bases Changes

Bases Page No.	Amendment / Revision No.	Effective Date
B 2.0-3	Revision 104	10-20-2016
B 2.0-4	Revision 097	02-05-2016
B 2.0-4	Revision 104	10-20-2016
B 2.0-5	Revision 104	10-20-2016
B 2.0-7	Revision 097	02-05-2016
B 2.0-7	Revision 104	10-20-2016
B 3.0-4	Revision 093	10-23-2015
B 3.1-31	Revision 094	11-06-2015
B 3.2-1	Revision 104	10-20-2016
B 3.2-2	Revision 104	10-20-2016
B 3.2-3	Revision 104	10-20-2016
B 3.2-3a	Revision 104	10-20-2016
B 3.2-4	Revision 104	10-20-2016
B 3.2-5	Revision 104	10-20-2016
B 3.2-5a	Revision 104	10-20-2016
B 3.2-7	Revision 104	10-20-2016
B 3.2-10	Revision 104	10-20-2016
B 3.2-10a	Revision 104	10-20-2016
B 3.3-103	Revision 096	03-03-2016
B 3.3-118	Revision 107	05-02-2017
B 3.3-196	Revision 097	02-05-2016
B 3.3-206	Revision 098	02-05-2016
B 3.6-20	Revision 105	09-01-2016
B 3.6-36	Revision 104	10-25-2016
B 3.7-3	Revision 102	09-01-2016
B 3.7-32	Revision 103	10-15-2016
B 3.7-36	Revision 103	10-15-2016
B 3.8-2	Revision 100	06-09-2016
B 3.8-7	Revision 100	06-09-2016
B 3.8-25	Revision 101	08-18-2016
B 3.8-26	Revision 101	08-18-2016
B 3.8-49	Revision 095	01-21-2016
B 3.8-52	Revision 095	01-21-2016
B 3.8-53	Revision 095	01-21-2016
B 3.8-54	Revision 095	01-21-2016
B 3.8-54a	Revision 095	01-21-2016
B 3.8-55	Revision 095	01-21-2016
B 3.8-56	Revision 095	01-21-2016
B 3.8-56a	Revision 095	01-21-2016
B 3.8-56b	Revision 095	01-21-2016
B 3.8-56c	Revision 095	01-21-2016

ENCLOSURE 2

Tennessee Valley Authority
Browns Ferry Nuclear Plant
Units 1, 2, and 3
Technical Specifications (TS) Bases Changes and Additions

Unit 2 Technical Specifications Bases Changes

Bases Page No.	Amendment / Revision No.	Effective Date
B 2.0-4	Revision 097	02-05-2016
B 2.0-7	Revision 097	02-05-2016
B 3.0-4	Revision 093	10-23-2015
B 3.2-2	Revision 106	10-27-2016
B 3.2-3	Revision 106	10-27-2016
B 3.2-3a	Revision 106	10-27-2016
B 3.2-3b	Deleted by Revision 106	10-27-2016
B 3.2-5	Revision 106	10-27-2016
B 3.2-5a	Revision 106	10-27-2016
B 3.2-6	Revision 106	10-27-2016
B 3.2-7	Revision 106	10-27-2016
B 3.2-8	Revision 106	10-27-2016
B 3.2-11	Revision 106	10-27-2016
B 3.2-13a	Revision 106	10-27-2016
B 3.3-106	Revision 096	03-03-2016
B 3.3-121	Revision 107	05-02-2017
B 3.3-199	Revision 097	02-05-2016
B 3.3-209	Revision 098	02-05-2016
B 3.4-55	Revision 092	07-30-2015
B 3.4-55a	Revision 092	07-30-2015
B 3.4-56	Revision 092	07-30-2015
B 3.4-63	Revision 092	07-30-2015
B 3.4-66a	Revision 092	07-30-2015
B 3.6-20	Revision 105	09-01-2016
B 3.6-36	Revision 105	09-01-2016
B 3.7-3	Revision 102	09-01-2016
B 3.7-32	Revision 103	10-15-2016
B 3.7-36	Revision 103	10-15-2016
B 3.8-2	Revision 100	06-09-2016
B 3.8-7	Revision 100	06-09-2016
B 3.8-25	Revision 101	08-18-2016
B 3.8-26	Revision 101	08-18-2016
B 3.8-49	Revision 095	01-21-2016
B 3.8-52	Revision 095	01-21-2016
B 3.8-53	Revision 095	01-21-2016
B 3.8-54	Revision 095	01-21-2016
B 3.8-54a	Revision 095	01-21-2016
B 3.8-55	Revision 095	01-21-2016
B 3.8-56	Revision 095	01-21-2016
B 3.8-56a	Revision 095	01-21-2016
B 3.8-56b	Revision 095	01-21-2016
B 3.8-56c	Revision 095	01-21-2016

ENCLOSURE 2

Tennessee Valley Authority
Browns Ferry Nuclear Plant
Units 1, 2, and 3
Technical Specifications (TS) Bases Changes and Additions

Unit 3 Technical Specifications Bases Changes

Bases Page No.	Revision No.	Effective Date
B 2.0-3	Revision 097	02-05-2016
B 2.0-4	Revision 097	02-05-2016
B 2.0-5	Revision 097	02-05-2016
B 2.0-5	Revision 106	11-07-2016
B 2.0-7	Revision 097	02-05-2016
B 3.0-4	Revision 093	10-23-2015
B 3.2-1	Revision 097	02-05-2016
B 3.2-2	Revision 106	10-27-2016
B 3.2-3	Revision 106	10-27-2016
B 3.2-3a	Revision 106	10-27-2016
B 3.2-3b	Deleted by Revision 106	10-27-2016
B 3.2-5	Revision 106	10-27-2016
B 3.2-5a	Revision 097	02-05-2016
B 3.2-5a	Revision 106	10-27-2016
B 3.2-6	Revision 106	10-27-2016
B 3.2-7	Revision 106	10-27-2016
B 3.2-8	Revision 106	10-27-2016
B 3.2-11	Revision 097	02-05-2016
B 3.2-13a	Revision 106	10-27-2016
B 3.3-106	Revision 096	03-03-2016
B 3.3-121	Revision 107	05-02-2017
B 3.3-199	Revision 097	02-05-2016
B 3.3-209	Revision 098	02-05-2016
B 3.4-55	Revision 099	02-05-2016
B 3.4-55a	Revision 099	02-05-2016
B 3.4-56	Revision 099	02-05-2016
B 3.4-63	Revision 099	02-05-2016
B 3.4-66a	Revision 099	02-05-2016
B 3.6-20	Revision 105	09-01-2016
B 3.6-36	Revision 105	09-01-2016
B 3.7-3	Revision 102	09-01-2016
B 3.7-32	Revision 103	10-15-2016
B 3.7-36	Revision 103	10-15-2016
B 3.8-1	Revision 100	06-09-2016
B 3.8-7	Revision 100	06-09-2016
B 3.8-25	Revision 101	08-18-2016
B 3.8-26	Revision 101	08-18-2016
B 3.8-49	Revision 095	01-21-2016
B 3.8-52	Revision 095	01-21-2016
B 3.8-53	Revision 095	01-21-2016
B 3.8-54	Revision 095	01-21-2016
B 3.8-54a	Revision 095	01-21-2016
B 3.8-55	Revision 095	01-21-2016
B 3.8-56	Revision 095	01-21-2016
B 3.8-56a	Revision 095	01-21-2016
B 3.8-56b	Revision 095	01-21-2016

ENCLOSURE 2

Tennessee Valley Authority
Browns Ferry Nuclear Plant
Units 1, 2, and 3
Technical Specifications (TS) Bases Changes and Additions

B 3.8-56c	Revision 095	01-21-2016
B 3.8-87	Revision 093	10-23-2015
B 3.8-88	Revision 093	10-23-2015

EFFECTIVE PAGE LISTING Implementation Date: October 30, 2017

Page No.	Amendment / Revision No.	Effective Date
Title Page	Amendment No.	Initial
Effective Page Listing	Revision 108	10-27-2017
i	Revision 40	10-26-2006
ii	0	Initial
iii	0	Initial
iv	Revision 29	01-25-2005
v	0	Initial
vi	0	Initial
B 2.0-1	Revision 68	10-18-2012
B 2.0-2	0	Initial
B 2.0-3	Revision 104	10-20-2016
B 2.0-4	Revision 104	10-20-2016
B 2.0-5	Revision 104	10-20-2016
B 2.0-6	0	Initial
B 2.0-7	Revision 104	10-20-2016
B 2.0-8	Revision 29	01-25-2005
B 2.0-9	0	Initial
B 2.0-10	Revision 29	01-25-2005
B 2.0-11	Revision 97	02-05-2016
B 3.0-1	0	Initial
B 3.0-2	0	Initial
B 3.0-3	0	Initial
B 3.0-4	Revision 93	10-23-2015
B 3.0-4a	Revision 0	Initial
B 3.0-5	0	Initial
B 3.0-6	Amendment 249	12-01-2003
B 3.0-7	Amendment 249	12-01-2003
B 3.0-8	Amendment 249 and Revision 24	12-01-2003 and 01-29-2004
B 3.0-8a	Amendment 249 and Revision 24	12-01-2003 and 01-29-2004
B 3.0-8b	Amendment 249	12-01-2003
B 3.0-9	0	Initial
B 3.0-10	Amendment 239	11-21-2000
B 3.0-10a	Revision 12	03-06-2001

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B 3.0-10b	Amendment 239	11-21-2000
B 3.0-11	0	Initial
B 3.0-12	0	Initial
B 3.0-13	0	Initial
B 3.0-14	0	Initial
B 3.0-15	Revision 83	09-21-2011
B 3.0-15a	Revision 83	09-21-2011
B 3.0-16	Amendment 243	12-23-2002
B 3.0-16a	Amendment 243	12-23-2002
B 3.0-17	Amendment 249	12-01-2003
B 3.0-18	Amendment 249 and Revision 24	12-01-2003 and 01-29-2004
B 3.1-1	Revision 68	10-18-2012
B 3.1-2	0	Initial
B 3.1-3	0	Initial
B 3.1-4	0	Initial
B 3.1-5	0	Initial
B 3.1-6	0	Initial
B 3.1-7	0	Initial
B 3.1-8	Revision 68	10-18-2012
B 3.1-9	0	Initial
B 3.1-10	0	Initial
B 3.1-11	0	Initial
B 3.1-12	0	Initial
B 3.1-13	0	Initial
B 3.1-14	0	Initial
B 3.1-15	0	Initial
B 3.1-16	0	Initial
B 3.1-17	0	Initial
B 3.1-18	0	Initial
B 3.1-19	Amendment 274	06-26-2009
B 3.1-20	0	Initial
B 3.1-21	0	Initial
B 3.1-22	0	Initial
B 3.1-23	Amendment 274	06-26-2009
B 3.1-24	0	Initial
B 3.1-25	Amendment 274	06-26-2009
B 3.1-26	0	Initial

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B 3.1-27	Revision 68	10-18-2012
B 3.1-28	Revision 9	12-15-1999
B 3.1-29	Revision 9	12-15-1999
B 3.1-30	Amendment 239	11-21-2000
B 3.1-31	Revision 94	11-06-2015
B 3.1-32	0	Initial
B 3.1-33	Revision 12	03-06-2001
B 3.1-34	Revision 68	10-18-2012
B 3.1-35	0	Initial
B 3.1-36	0	Initial
B 3.1-37	0	Initial
B 3.1-38	0	Initial
B 3.1-39	0	Initial
B 3.1-40	0	Initial
B 3.1-41	Revision 68	10-18-2012
B 3.1-42	Revision 68	10-18-2012
B 3.1-42a	Amendment 276	11-19-2009
B 3.1-43	0	Initial
B 3.1-44	0	Initial
B 3.1-45	0	Initial
B 3.1-46	Revision 68	10-18-2012
B 3.1-47	Revision 29	01-25-2005
B 3.1-48	Revision 50	05-03-2007
B 3.1-49	Revision 29	01-25-2005
B 3.1-50	Revision 29	01-25-2005
B 3.1-51	Revision 29	01-25-2005
B 3.1-52	Revision 50	05-03-2007
B 3.1-53	Revision 29	01-25-2005
B 3.1-54	Revision 29	01-25-2005
B 3.1-55	Revision 50	05-03-2007
B 3.1-56	Revision 43	01-17-2007
B 3.1-57	Revision 29	01-25-2005
B 3.1-58	Revision 29	01-25-2005
B 3.1-59	Revision 29	01-25-2005
B 3.1-60	Revision 29	01-25-2005
B 3.1-61	Revision 29	01-25-2005
B 3.1-62	Revision 29	01-25-2005

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B 3.1-63	Revision 43	01-17-2007
B 3.2-1	Revision 104	10-20-2016
B 3.2-2	Revision 104	10-20-2016
B 3.2-3	Revision 104	10-20-2016
B 3.2-3a	Revision 104	10-20-2016
B 3.2-4	Revision 104	10-20-2016
B 3.2-5	Revision 104	10-20-2016
B 3.2-5a	Revision 104	10-20-2016
B 3.2-6	0	Initial
B 3.2-7	Revision 104	10-20-2016
B 3.2-8	Revision 68	10-18-2012
B 3.2-9	Revision 68	10-18-2012
B 3.2-10	Revision 104	10-20-2016
B 3.2-10a	Revision 104	10-20-2016
B 3.2-11	Revision 29	01-25-2005
B 3.2-12	Revision 68	10-18-2012
B 3.2-12a	Revision 68	10-18-2012
B 3.2-13	0	Initial
B 3.2-14	0	Initial
B 3.3-1	0	Initial
B 3.3-2	0	Initial
B 3.3-3	0	Initial
B 3.3-4	Revision 41	11-09-2006
B 3.3-4a	Revision 41	11-09-2006
B 3.3-5	0	Initial
B 3.3-6	0	Initial
B 3.3-7	0	Initial
B 3.3-8	0	Initial
B 3.3-9	Revision 50	05-03-2007
B 3.3-10	Revision 45	02-27-2007
B 3.3-11	Revision 45	02-27-2007
B 3.3-12	Revision 86	10-29-2014
B 3.3-13	Revision 86	10-29-2014
B 3.3-14	Revision 40	10-26-2006
B 3.3-15	Revision 40	10-26-2006
B 3.3-16	Revision 45	02-27-2007
B 3.3-16a	Revision 45	02-27-2007

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B 3.3-16b	Revision 45	02-27-2007
B 3.3-17	0	Initial
B 3.3-17a	Revision 41	11-09-2006
B 3.3-18	0	Initial
B 3.3-19	Revision 41	11-09-2006
B 3.3-19a	Revision 41	11-09-2006
B 3.3-20	0	Initial
B 3.3-21	0	Initial
B 3.3-22	0	Initial
B 3.3-23	0	Initial
B 3.3-24	0	Initial
B 3.3-25	0	Initial
B 3.3-26	Revision 41	11-09-2006
B 3.3-26a	Revision 41	11-09-2006
B 3.3-27	0	Initial
B 3.3-28	0	Initial
B 3.3-29	0	Initial
B 3.3-30	Revision 45	02-27-2007
B 3.3-31	Revision 40	10-26-2006
B 3.3-32	Revision 45	02-27-2007
B 3.3-32a	Revision 40	10-26-2006
B 3.3-33	Revision 45	02-27-2007
B 3.3-34	Revision 45	02-27-2007
B 3.3-34a	Revision 45	02-27-2007
B 3.3-35	0	Initial
B 3.3-36	Revision 40	10-26-2006
B 3.3-37	Revision 40	10-26-2006
B 3.3-38	Revision 52	05-11-2007
B 3.3-39	Revision 40	10-26-2006
B 3.3-40	Revision 43	01-17-2007
B 3.3-41	Revision 45	02-27-2007
B 3.3-42	Revision 45	02-27-2007
B 3.3-43	Revision 43	01-17-2007
B 3.3-43a	Revision 45	02-27-2007
B 3.3-44	Revision 45	02-27-2007
B 3.3-44a	Revision 45	02-27-2007
B 3.3-45	0	Initial

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B 3.3-46	Revision 40	10-26-2006
B 3.3-47	0	Initial
B 3.3-48	0	Initial
B 3.3-49	0	Initial
B 3.3-50	0	Initial
B 3.3-51	0	Initial
B 3.3-52	0	Initial
B 3.3-53	0	Initial
B 3.3-54	0	Initial
B 3.3-55	0	Initial
B 3.3-56	0	Initial
B 3.3-57	Revision 40	10-26-2006
B 3.3-58	Revision 40	10-26-2006
B 3.3-59	Revision 68	10-18-2012
B 3.3-60	Revision 40	10-26-2006
B 3.3-61	Revision 68	10-18-2012
B 3.3-62	0	Initial
B 3.3-63	0	Initial
B 3.3-64	0	Initial
B 3.3-65	0	Initial
B 3.3-66	0	Initial
B 3.3-67	Revision 40	10-26-2006
B 3.3-68	Revision 43	01-17-2007
B 3.3-69	Revision 43	01-17-2007
B 3.3-70	Revision 43	01-17-2007
B 3.3-70a	Revision 43	01-17-2007
B 3.3-71	Revision 68	10-18-2012
B 3.3-71a	Revision 68	10-18-2012
B 3.3-72	0	Initial
B 3.3-73	0	Initial
B 3.3-74	0	Initial
B 3.3-75	0	Initial
B 3.3-76	0	Initial
B 3.3-77	0	Initial
B 3.3-78	0	Initial
B 3.3-79	Revision 43	01-17-2007
B 3.3-80	Revision 43	01-17-2007

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B 3.3-81	0	Initial
B 3.3-82	0	Initial
B 3.3-83	0	Initial
B 3.3-84	0	Initial
B 3.3-85	0	Initial
B 3.3-86	Revision 53	05-18-2007
B 3.3-87	Revision 32	04-06-2005
B 3.3-88	0	Initial
B 3.3-89	Amendment 249	12-01-2003
B 3.3-90	0	Initial
B 3.3-91	Revision 32	04-06-2005
B 3.3-92	Revision 32	04-06-2005
B 3.3-93	0	Initial
B 3.3-94	Revision 43	01-17-2007
B 3.3-95	0	Initial
B 3.3-96	0	Initial
B 3.3-97	0	Initial
B 3.3-98	Revision 4	04-09-1999
B 3.3-99	Amendment 249	12-01-2003
B 3.3-100	Revision 43	01-17-2007
B 3.3-101	Revision 43	01-17-2007
B 3.3-102	Revision 82	10-10-2013
B 3.3-103	Revision 96	03-03-2016
B 3.3-104	Revision 54	05-19-2007
B 3.3-104a	Revision 82	10-10-2013
B 3.3-105	Revision 68	10-18-2012
B 3.3-106	Revision 68	10-18-2012
B 3.3-107	0	Initial
B 3.3-108	Revision 68	10-18-2012
B 3.3-109	Revision 68	10-18-2012
B 3.3-110	Revision 68	10-18-2012
B 3.3-111	0	Initial
B 3.3-112	Revision 68	10-18-2012
B 3.3-113	Revision 68	10-18-2012
B 3.3-114	0	Initial
B 3.3-115	Revision 43	01-17-2007

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B 3.3-116	Revision 53	05-18-2007
B 3.3-117	0	Initial
B 3.3-118	Revision 107	05-02-2017
B 3.3-119	0	Initial
B 3.3-120	0	Initial
B 3.3-121	0	Initial
B 3.3-122	0	Initial
B 3.3-123	0	Initial
B 3.3-124	0	Initial
B 3.3-125	0	Initial
B 3.3-126	Revision 43	01-17-2007
B 3.3-127	Revision 43	01-17-2007
B 3.3-128	0	Initial
B 3.3-129	0	Initial
B 3.3-130	0	Initial
B 3.3-131	0	Initial
B 3.3-132	0	Initial
B 3.3-133	0	Initial
B 3.3-134	0	Initial
B 3.3-135	0	Initial
B 3.3-136	Revision 41	11-09-2006
B 3.3-137	Revision 41	11-09-2006
B 3.3-138	0	Initial
B 3.3-139	Revision 41	11-09-2006
B 3.3-140	Revision 41	11-09-2006
B 3.3-141	Revision 41	11-09-2006
B 3.3-141a	Revision 41	11-09-2006
B 3.3-141b	Revision 41	11-09-2006
B 3.3-141c	Revision 41	11-09-2006
B 3.3-142	0	Initial
B 3.3-143	Revision 47	03-22-2007
B 3.3-144	Revision 47	03-22-2007
B 3.3-145	Revision 85	09-30-2014
B 3.3-145a	Revision 65	05-29-2012
B 3.3-145b	Revision 47	03-22-2007
B 3.3-146	Revision 41	11-09-2006
B 3.3-146a	Revision 41	11-09-2006
B 3.3-147	0	Initial

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B 3.3-148	Revision 41	11-09-2006
B 3.3-148a	Revision 41	11-09-2006
B 3.3-149	Revision 41	11-09-2006
B 3.3-149a	Revision 41	11-09-2006
B 3.3-150	0	Initial
B 3.3-151	0	Initial
B 3.3-152	0	Initial
B 3.3-153	0	Initial
B 3.3-154	Revision 41	11-09-2006
B 3.3-154a	Revision 41	11-09-2006
B 3.3-155	Revision 41	11-09-2006
B 3.3-155a	Revision 41	11-09-2006
B 3.3-156	0	Initial
B 3.3-156a	Revision 41	11-09-2006
B 3.3-157	0	Initial
B 3.3-158	0	Initial
B 3.3-159	0	Initial
B 3.3-160	0	Initial
B 3.3-161	0	Initial
B 3.3-162	0	Initial
B 3.3-163	0	Initial
B 3.3-164	0	Initial
B 3.3-165	0	Initial
B 3.3-166	0	Initial
B 3.3-167	0	Initial
B 3.3-168	0	Initial
B 3.3-169	0	Initial
B 3.3-170	0	Initial
B 3.3-171	0	Initial
B 3.3-172	Revision 13	04-11-2001
B 3.3-173	Revision 13	04-11-2001
B 3.3-174	Revision 65	05-29-2012
B 3.3-175	0	Initial
B 3.3-176	0	Initial
B 3.3-177	0	Initial
B 3.3-178	Revision 41	11-09-2006
B 3.3-179	Revision 41	11-09-2006

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B 3.3-179a	Revision 41	11-09-2006
B 3.3-180	0	Initial
B 3.3-181	0	Initial
B 3.3-182	0	Initial
B 3.3-183	0	Initial
B 3.3-184	0	Initial
B 3.3-185	Revision 43	01-17-2007
B 3.3-186	Revision 43	01-17-2007
B 3.3-187	0	Initial
B 3.3-188	0	Initial
B 3.3-189	0	Initial
B 3.3-190	0	Initial
B 3.3-191	0	Initial
B 3.3-192	0	Initial
B 3.3-193	0	Initial
B 3.3-194	0	Initial
B 3.3-195	Revision 29	01-25-2005
B 3.3-196	Revision 97	02-05-2016
B 3.3-196a	Revision 41	11-09-2006
B 3.3-197	Revision 29	01-25-2005
B 3.3-198	0	Initial
B 3.3-199	Revision 29	01-25-2005
B 3.3-200	Revision 29	01-25-2005
B 3.3-201	0	Initial
B 3.3-202	0	Initial
B 3.3-203	0	Initial
B 3.3-204	0	Initial
B 3.3-205	0	Initial
B 3.3-206	Revision 98	02-05-2016
B 3.3-207	0	Initial
B 3.3-208	0	Initial
B 3.3-209	0	Initial
B 3.3-210	Revision 29	01-25-2005
B 3.3-211	0	Initial
B 3.3-212	0	Initial
B 3.3-213	0	Initial
B 3.3-214	0	Initial

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B 3.3-215	0	Initial
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B 3.3-217	0	Initial
B 3.3-218	0	Initial
B 3.3-219	0	Initial
B 3.3-220	0	Initial
B 3.3-221	Revision 43	01-17-2007
B 3.3-222	0	Initial
B 3.3-223	Revision 35	02-14-2006
B 3.3-224	0	Initial
B 3.3-225	0	Initial
B 3.3-226	0	Initial
B 3.3-227	0	Initial
B 3.3-228	Revision 35	02-14-2006
B 3.3-229	Revision 35	02-14-2006
B 3.3-230	0	Initial
B 3.3-231	0	Initial
B 3.3-232	0	Initial
B 3.3-233	0	Initial
B 3.3-234	0	Initial
B 3.3-235	Revision 43	01-17-2007
B 3.3-236	Revision 43	01-17-2007
B 3.3-237	Revision 35	02-14-2006
B 3.3-238	Revision 35	02-14-2006
B 3.3-239	0	Initial
B 3.3-240	0	Initial
B 3.3-241	0	Initial
B 3.3-242	Revision 35	02-14-2006
B 3.3-243	Revision 29	01-25-2005
B 3.3-244	0	Initial
B 3.3-245	0	Initial
B 3.3-246	0	Initial
B 3.3-247	Revision 27	07-14-2004
B 3.3-248	0	Initial
B 3.3-249	0	Initial
B 3.3-250	0	Initial
B 3.3-251	Revision 43	01-17-2007

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B 3.3-252	Revision 43	01-17-2007
B 3.3-253	0	Initial
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B 3.3-256	0	Initial
B 3.3-257	0	Initial
B 3.3-258	0	Initial
B 3.3-259	0	Initial
B 3.3-260	0	Initial
B 3.3-261	0	Initial
B 3.3-262	0	Initial
B 3.3-263	Amendment 235	11-30-1998
B 3.3-264	0	Initial
B 3.3-265	0	Initial
B 3.3-266	0	Initial
B 3.3-267	0	Initial
B 3.3-268	0	Initial
B 3.3-269	0	Initial
B 3.3-270	0	Initial
B 3.3-271	0	Initial
B 3.3-272	0	Initial
B 3.3-273	0	Initial
B 3.3-274	Revision 43	01-17-2007
B 3.4-1	Revision 49	04-30-2007
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B 3.4-7	Revision 45	02-27-2007
B 3.4-8	Revision 45	02-27-2007
B 3.4-9	Revision 45	02-27-2007
B 3.4-10	Revision 85	10-29-2014
B 3.4-11	0	Initial
B 3.4-12	0	Initial
B 3.4-13	0	Initial
B 3.4-14	Amendment 236	12-23-1998

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B 3.4-15	0	Initial
B 3.4-16	0	Initial
B 3.4-17	0	Initial
B 3.4-18	0	Initial
B 3.4-19	0	Initial
B 3.4-20	0	Initial
B 3.4-21	0	Initial
B 3.4-22	Revision 81	10-16-2013
B 3.4-23	0	Initial
B 3.4-24	0	Initial
B 3.4-25	0	Initial
B 3.4-26	0	Initial
B 3.4-27	0	Initial
B 3.4-28	0	Initial
B 3.4-29	0	Initial
B 3.4-30	0	Initial
B 3.4-31	0	Initial
B 3.4-32	0	Initial
B 3.4-33	0	Initial
B 3.4-34	Amendment 249	12-01-2003
B 3.4-35	0	Initial
B 3.4-36	Revision 43	01-17-2007
B 3.4-37	Revision 29	01-25-2005
B 3.4-38	Revision 29	01-25-2005
B 3.4-39	Amendment 249	12-01-2003
B 3.4-40	Revision 29	01-25-2005
B 3.4-41	Revision 29	01-25-2005
B 3.4-42	0	Initial
B 3.4-43	0	Initial
B 3.4-44	0	Initial
B 3.4-45	Amendment 249	12-01-2003
B 3.4-46	0	Initial
B 3.4-47	0	Initial
B 3.4-48	0	Initial
B 3.4-49	0	Initial
B 3.4-50	Revision 91	06-17-2015
B 3.4-51	0	Initial

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B 3.4-52	0	Initial
B 3.4-53	Revision 91	06-17-2015
B 3.4-54	0	Initial
B 3.4-55	Amendment 287	02-02-2015
B 3.4-55a	Amendment 287	02-02-2015
B 3.4-56	Revision 15	01-24-2002
B 3.4-56a	Revision 38	09-21-2006
B 3.4-56b	Revision 38	09-21-2006
B 3.4-57	Revision 38	09-21-2006
B 3.4-58	Revision 38	09-21-2006
B 3.4-59	0	Initial
B 3.4-60	0	Initial
B 3.4-61	0	Initial
B 3.4-62	0	Initial
B 3.4-63	Revision 38	09-21-2006
B 3.4-64	0	Initial
B 3.4-65	Revision 38	09-21-2006
B 3.4-66	Revision 38	09-21-2006
B 3.4-67	Revision 50	05-03-2007
B 3.4-68	Revision 50	05-03-2007
B 3.4-69	Revision 50	05-03-2007
B 3.4-70	0	Initial
B 3.5-1	0	Initial
B 3.5-2	0	Initial
B 3.5-3	Revision 65	05-29-2012
B 3.5-4	Revision 47	03-22-2007
B 3.5-5	Revision 80	10-04-2013
B 3.5-6	0	Initial
B 3.5-7	0	Initial
B 3.5-8	Amendment 249	12-01-2003
B 3.5-8a	Amendment 249	12-01-2003
B 3.5-9	0	Initial
B 3.5-10	Amendment 240	03-12-2001
B 3.5-11	Amendment 240	03-12-2001
B 3.5-12	0	Initial
B 3.5-13	0	Initial
B 3.5-14	Revision 46	03-14-2007

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B 3.5-17	Revision 43	01-17-2007
B 3.5-18	Revision 43	01-17-2007
B 3.5-19	Revision 43	01-17-2007
B 3.5-20	0	Initial
B 3.5-21	Revision 43	01-17-2007
B 3.5-22	0	Initial
B 3.5-23	0	Initial
B 3.5-24	Revision 46	03-14-2007
B 3.5-25	0	Initial
B 3.5-26	0	Initial
B 3.5-27	0	Initial
B 3.5-28	0	Initial
B 3.5-29	0	Initial
B 3.5-30	Revision 6	08-17-1999
B 3.5-31	Revision 50	05-03-2007
B 3.5-32	Amendment 249	12-01-2003
B 3.5-32a	Amendment 249	12-01-2003
B 3.5-33	0	Initial
B 3.5-34	0	Initial
B 3.5-35	Revision 53	05-18-2007
B 3.5-36	Revision 43	01-17-2007
B 3.5-37	Revision 43	01-17-2007
B 3.5-38	0	Initial
B 3.6-1	0	Initial
B 3.6-2	0	Initial
B 3.6-3	Revision 49	04-30-2007
B 3.6-4	0	Initial
B 3.6-5	0	Initial
B 3.6-6	Revision 43	01-17-2007
B 3.6-7	0	Initial
B 3.6-8	Revision 49	04-30-2007
B 3.6-9	0	Initial
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B 3.6-11	0	Initial
B 3.6-12	0	Initial

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B 3.6-15	0	Initial
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B 3.6-17	0	Initial
B 3.6-18	0	Initial
B 3.6-19	0	Initial
B 3.6-20	Revision 105	09-01-2016
B 3.6-21	Revision 3	03-19-1999
B 3.6-22	0	Initial
B 3.6-23	0	Initial
B 3.6-24	0	Initial
B 3.6-25	0	Initial
B 3.6-26	0	Initial
B 3.6-27	0	Initial
B 3.6-28	0	Initial
B 3.6-29	0	Initial
B 3.6-30	0	Initial
B 3.6-31	0	Initial
B 3.6-32	0	Initial
B 3.6-33	Revision 29	01-25-2005
B 3.6-34	Revision 84	08-29-2014
B 3.6-34a	Revision 40	10-26-2006
B 3.6-35	Revision 62	01-12-2012
B 3.6-36	Revision 105	09-01-2016
B 3.6-37	Revision 50	05-03-2007
B 3.6-38	0	Initial
B 3.6-39	0	Initial
B 3.6-40	0	Initial
B 3.6-41	0	Initial
B 3.6-42	0	Initial
B 3.6-43	0	Initial
B 3.6-44	0	Initial
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B 3.6-48	Revision 43	01-17-2007

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B 3.6-53	0	Initial
B 3.6-54	0	Initial
B 3.6-55	0	Initial
B 3.6-56	Revision 43	01-17-2007
B 3.6-57	0	Initial
B 3.6-58	Revision 85	09-30-2014
B 3.6-59	0	Initial
B 3.6-60	0	Initial
B 3.6-61	0	Initial
B 3.6-62	0	Initial
B 3.6-63	Revision 85	09-30-2014
B 3.6-64	0	Initial
B 3.6-65	0	Initial
B 3.6-66	0	Initial
B 3.6-67	0	Initial
B 3.6-68	0	Initial
B 3.6-69	0	Initial
B 3.6-70	0	Initial
B 3.6-71	Amendment 241	06-08-2001
B 3.6-72	Amendment 241	06-08-2001
B 3.6-73	Revision 81	10-16-2013
B 3.6-74	0	Initial
B 3.6-75	0	Initial
B 3.6-76	0	Initial
B 3.6-77	0	Initial
B 3.6-78	0	Initial
B 3.6-79	0	Initial
B 3.6-80	0	Initial
B 3.6-81	0	Initial
B 3.6-82	0	Initial
B 3.6-83	0	Initial
B 3.6-84	0	Initial
B 3.6-85	Revision 3	03-19-1999
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B 3.6-87	0	Initial
B 3.6-88	0	Initial
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B 3.6-90	0	Initial
B 3.6-91	0	Initial
B 3.6-92	0	Initial
B 3.6-93	Revision 34	09-07-2005
B 3.6-94	Revision 34	09-07-2005
B 3.6-95	0	Initial
B 3.6-96	0	Initial
B 3.6-97	0	Initial
B 3.6-98	0	Initial
B 3.6-99	0	Initial
B 3.6-100	0	Initial
B 3.6-101	0	Initial
B 3.6-102	Revision 29	01-25-2005
B 3.6-103	Revision 29	01-25-2005
B 3.6-104	Revision 29	01-25-2005
B 3.6-105	Revision 29	01-25-2005
B 3.6-106	Revision 29	01-25-2005
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B 3.6-109	Revision 29	01-25-2005
B 3.6-110	0	Initial
B 3.6-111	0	Initial
B 3.6-112	Revision 29	01-25-2005
B 3.6-113	Revision 29	01-25-2005
B 3.6-114	0	Initial
B 3.6-115	Revision 29	01-25-2005
B 3.6-116	Revision 29	01-25-2005
B 3.6-117	0	Initial
B 3.6-118	Revision 29	01-25-2005
B 3.6-119	Revision 29	01-25-2005
B 3.6-120	108	10-27-2017
B 3.6-121	235	11-30-98
B 3.7-1	Revision 73	01-03-2013
B 3.7-2	Revision 73	01-03-2013
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B 3.7-4	Revision 73	01-03-2013
B 3.7-4a	Revision 73	01-03-2013
B 3.7-5	Revision 73	01-03-2013
B 3.7-5a	Revision 73	01-03-2013
B 3.7-5b	Revision 73	01-03-2013
B 3.7-6	Revision 73	01-03-2013
B 3.7-6a	Revision 73	01-03-2013
B 3.7-7	Revision 73	01-03-2013
B 3.7-8	Revision 73	01-03-2013
B 3.7-9	Revision 73	01-03-2013
B 3.7-10	0	Initial
B 3.7-11	0	Initial
B 3.7-12	0	Initial
B 3.7-13	0	Initial
B 3.7-14	Revision 69	10-05-2012
B 3.7-15	0	Initial
B 3.7-16	Revision 69	10-05-2012
B 3.7-17	Revision 67	08-10-2012
B 3.7-18	Revision 67	08-10-2012
B 3.7-18a	Revision 90	02-26-2015
B 3.7-19	Revision 67	08-10-2012
B 3.7-19a	Revision 67	08-10-2012
B 3.7-20	Revision 67	08-10-2012
B 3.7-21	Amendment 275	10-16-2009
B 3.7-21a	Revision 67	08-10-2012
B 3.7-22	Revision 67	08-10-2012
B 3.7-23	Revision 67	08-10-2012
B 3.7-24	Revision 108	10-27-2017
B 3.7-25	Amendment 275	10-16-2009
B 3.7-25a	Amendment 275	10-16-2009
B 3.7-25b	Revision 67	08-10-2012
B 3.7-26	0	Initial
B 3.7-27	0	Initial
B 3.7-28	0	Initial
B 3.7-29	0	Initial
B 3.7-30	0	Initial
B 3.7-31	Amendment 235	11-30-1998
B 3.7-32	Revision 103	10-15-2016

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B 3.7-33	Revision 68	10-18-2012
B 3.7-34	Revision 68	10-18-2012
B 3.7-35	Revision 43	01-17-2007
B 3.7-36	Revision 103	10-15-2016
B 3.7-37	Revision 29	01-25-2005
B 3.7-38	0	Initial
B 3.7-39	Revision 29	01-25-2005
B 3.8-1	Revision 52	05-11-2007
B 3.8-2	Revision 100	06-09-2016
B 3.8-3	Revision 42	11-16-2006
B 3.8-3a	Revision 42	11-16-2006
B 3.8-4	Revision 47	03-22-2007
B 3.8-4a	Revision 47	03-22-2007
B 3.8-5	Amendment 280	10-05-2011
B 3.8-5a	Amendment 280	10-05-2011
B 3.8-6	Revision 52	05-11-2007
B 3.8-7	Revision 100	06-09-2016
B 3.8-8	Revision 52	05-11-2007
B 3.8-9	Revision 52	05-11-2007
B 3.8-10	0	Initial
B 3.8-11	Amendment 249	12-01-2003
B 3.8-12	0	Initial
B 3.8-13	0	Initial
B 3.8-14	Amendment 280	10-05-2011
B 3.8-15	Amendment 280	10-05-2011
B 3.8-15a	Amendment 280	10-05-2011
B 3.8-16	Amendment 280	10-05-2011
B 3.8-17	Amendment 280	10-05-2011
B 3.8-18	Amendment 280	10-05-2011
B 3.8-18a	Amendment 280	10-05-2011
B 3.8-19	Revision 47	03-22-2007
B 3.8-20	0	Initial
B 3.8-21	0	Initial
B 3.8-22	0	Initial
B 3.8-23	0	Initial
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B 3.8-25	Revision 101	08-18-2016
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B 3.8-29	0	Initial
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B 3.8-31	0	Initial
B 3.8-32	0	Initial
B 3.8-33	Amendment 235	11-30-1998
B 3.8-34	Amendment 235	11-30-1998
B 3.8-35	Amendment 235	11-30-1998
B 3.8-36	Revision 42	11-16-2006
B 3.8-37	Revision 42	11-16-2006
B 3.8-38	0	Initial
B 3.8-39	0	Initial
B 3.8-40	0	Initial
B 3.8-41	Revision 52	05-11-2007
B 3.8-42	Revision 52	05-11-2007
B 3.8-43	Revision 52	05-11-2007
B 3.8-44	Revision 52	05-11-2007
B 3.8-45	0	Initial
B 3.8-46	0	Initial
B 3.8-47	0	Initial
B 3.8-48	0	Initial
B 3.8-49	Revision 95	01-21-2016
B 3.8-50	0	Initial
B 3.8-51	0	Initial
B 3.8-52	Revision 95	01-21-2016
B 3.8-53	Revision 95	01-21-2016
B 3.8-54	Revision 95	01-21-2016
B 3.8-54a	Revision 95	01-21-2016
B 3.8-55	Revision 95	01-21-2016
B 3.8-56	Revision 95	01-21-2016
B 3.8-56a	Revision 95	01-21-2016
B 3.8-56b	Revision 95	01-21-2016
B 3.8-56c	Revision 95	01-21-2016
B 3.8-57	0	Initial
B 3.8-58	0	Initial
B 3.8-59	0	Initial
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B 3.8-62	0	Initial
B 3.8-63	0	Initial
B 3.8-64	0	Initial
B 3.8-65	Amendment 235	11-30-1998
B 3.8-66	Revision 58	10-01-2008
B 3.8-67	0	Initial
B 3.8-68	235	11-30-1998
B 3.8-69	0	Initial
B 3.8-70	0	Initial
B 3.8-71	0	Initial
B 3.8-72	0	Initial
B 3.8-73	0	Initial
B 3.8-74	0	Initial
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B 3.8-81	0	Initial
B 3.8-82	0	Initial
B 3.8-83	0	Initial
B 3.8-84	0	Initial
B 3.8-85	0	Initial
B 3.8-86	Revision 33	08-04-2005
B 3.8-87	Revision 56	04-03-2008
B 3.8-87a	Revision 56	04-03-2008
B 3.8-88	0	Initial
B 3.8-89	Revision 36	06-22-2006
B 3.8-90	Revision 33	08-04-2005
B 3.8-91	0	Initial
B 3.8-92	0	Initial
B 3.8-93	Revision 33	08-04-2005
B 3.8-94	Revision 33	08-04-2005
B 3.8-95	Revision 33	08-04-2005
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B 3.8-100	Revision 33	08-04-2005
B 3.8-101	0	Initial
B 3.8-102	Revision 33	08-04-2005
B 3.8-103	0	Initial
B 3.8-104	0	Initial
B 3.8-105	0	Initial
B 3.8-106	0	Initial
B 3.8-107	0	Initial
B 3.8-108	0	Initial
B 3.9-1	0	Initial
B 3.9-2	0	Initial
B 3.9-3	0	Initial
B 3.9-4	Amendment 242	03-06-2002
B 3.9-5	Revision 60	10-16-2009
B 3.9-5a	Revision No. 16	03-21-2002
B 3.9-6	0	Initial
B 3.9-7	0	Initial
B 3.9-8	0	Initial
B 3.9-9	0	Initial
B 3.9-10	Revision 60	10-16-2009
B 3.9-11	0	Initial
B 3.9-12	0	Initial
B 3.9-13	Revision 60	10-16-2009
B 3.9-14	0	Initial
B 3.9-15	0	Initial
B 3.9-16	0	Initial
B 3.9-17	0	Initial
B 3.9-18	Revision 60	10-16-2009
B 3.9-19	0	Initial
B 3.9-20	0	Initial
B 3.9-21	0	Initial
B 3.9-22	Revision 60	10-16-2009
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B 3.9-24	0	Initial
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B 3.9-30	0	Initial
B 3.9-31	0	Initial
B 3.9-32	Revision 91	06-17-2015
B 3.9-33	0	Initial
B 3.9-34	0	Initial
B 3.9-35	0	Initial
B 3.9-36	Amendment 251	09-27-2004
B 3.9-37	Amendment 251	09-27-2004
B 3.9-38	Amendment 251	09-27-2004
B 3.10-1	Revision 51	05-10-2007
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B 3.10-3	Revision 51	05-10-2007
B 3.10-4	Revision 51	05-10-2007
B 3.10-4a	Revision 51	05-10-2007
B 3.10-5	0	Initial
B 3.10-6	0	Initial
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B 3.10-9	0	Initial
B 3.10-10	0	Initial
B 3.10-11	0	Initial
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B 3.10-13	0	Initial
B 3.10-14	0	Initial
B 3.10-15	0	Initial
B 3.10-16	0	Initial
B 3.10-17	0	Initial
B 3.10-18	0	Initial
B 3.10-19	0	Initial
B 3.10-20	0	Initial
B 3.10-21	Revision 4	04-09-1999
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B 3.10-28	0	Initial
B 3.10-29	0	Initial
B 3.10-30	0	Initial
B 3.10-31	0	Initial
B 3.10-32	0	Initial
B 3.10-33	0	Initial
B 3.10-34	0	Initial
B 3.10-35	0	Initial
B 3.10-36	Revision 68	10-18-2012
B 3.10-37	Revision 68	10-18-2012
B 3.10-38	Revision 68	10-18-2012
B 3.10-39	0	Initial
B 3.10-40	Revision 68	10-18-2012
B 3.10-41	0	Initial
B 3.10-42	Revision 68	10-18-2012
B 3.10-43	Revision 40	10-26-2006
B 3.10-44	0	Initial
B 3.10-45	0	Initial
B 3.10-46	0	Initial
B 3.10-47	Revision 40	10-26-2006
B 3.10-48	Revision 68	10-18-2012
B 3.10-48a	Revision 68	10-18-2012

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity

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

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

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES

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.

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

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

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

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

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

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BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model combining all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved AREVA critical power correlations. One specific uncertainty included in the SL is the uncertainty inherent in the critical power correlation. References 2, 3, 4, 5, and 6 describe the uncertainties and methodologies used in determining the MCPR SL.

(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. ANP-10307PA Revision 0, AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, AREVA NP, June 2011.
5. ANP-10298PA Revision 0, ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, March 2010.
6. ANP-3140(P) Revision 0, Browns Ferry Units 1, 2, and 3 Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, August 2012.
7. 10 CFR 50.67.
8. ANP-3245P Revision 1, Browns Ferry Evaluation of PRFO Low Pressure Technical Specification Value, AREVA Inc., February 2014.

BASES (continued)

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 50.67, "Accident Source Term," limits (Ref. 7). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. EMF-2209(P)(A), SPCB Critical Power Correlation, (as identified in the COLR).
3. EMF-2245(P)(A), Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel, (as identified in the COLR).
4. ANP-10307PA Revision 0, AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, AREVA NP, June 2011.
5. ANP-10298PA Revision 0, ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, March 2010.
6. ANP-3140(P) Revision 0, Browns Ferry Units 1, 2, and 3 Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, August 2012.
7. 10 CFR 50.67.
8. ANP-3245P Revision 1, Browns Ferry Evaluation of PRFO Low Pressure Technical Specification Value, AREVA Inc., February 2014.

BASES

LCO 3.0.3 (continued)

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The LCO phrase, "Action shall be initiated within 1 hour ..." does not mean that a change in load must be commenced by the end of the 1 hour period. The action initiated at the end of the 1 hour period may be administrative in nature, such as preparing shutdown procedures. If corrective measures which would allow exiting LCO 3.0.3 are not complete at the end of 1 hour, but there is reasonable assurance that they will be completed with enough time remaining to allow for an orderly unit shutdown, if required, commencing a load decrease may be delayed until that time. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

(continued)

BASES

LCO 3.0.3 (continued)

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met.

b. A Condition exists for which the Required Actions have now been performed.

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.2

Additional testing of a sample of control rods is required to verify the continued performance of the scram function during the cycle. A representative sample contains at least 10% of the control rods. This sample remains representative if no more than 7.5% of the control rods in the sample tested are determined to be "slow." With more than 7.5% of the sample declared to be "slow" per the criteria in Table 3.1.4-1, additional control rods are tested until this 7.5% criterion (i.e., 7.5% of the entire sample) is satisfied, or until the total number of "slow" control rods (throughout the core from all Surveillances) exceeds the LCO limit. For planned testing, the control rods selected for the sample should be different for each test. Data from inadvertent scrams should be used whenever possible to avoid unnecessary testing at power, even if the control rods with data may have been previously tested in a sample. The 200 day Frequency is based on operating experience that has shown control rod scram times do not significantly change over an operating cycle. This Frequency is also reasonable based on the additional Surveillances done on the CRDs at more frequent intervals in accordance with LCO 3.1.3 and LCO 3.1.5, "Control Rod Scram Accumulators."

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

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

APPLICABLE SAFETY ANALYSES

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

(Deleted by Tech Spec Bases Revision 104)

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

(Deleted by Tech Spec Bases Revision 104)

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

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 operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

(continued)

BASES

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA and transient analyses may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR out of specification.

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal 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.

(continued)

BASES

REFERENCES

1. (Deleted by Tech Spec Revision 104).

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.

(continued)

BASES

REFERENCES (continued)

11. EMF-2361(P)(A), Rev. 0, "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP Inc., as supplemented by the site-specific approval in NRC safety evaluation dated April 27, 2012, and July 31, 2014.

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

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

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

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

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

(continued)

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 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 (Ref. 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 stop valve closure and 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

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 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. (Deleted by Tech Spec Revision 104).
3. FSAR, Chapter 3.
4. FSAR, Chapter 14.
5. FSAR, Appendix N.
6. (Deleted by Tech Spec Revision 104).
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. (Deleted by Tech Spec Revision 104).

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

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

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

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

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

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

BASES

LCO 3.3.11	The primary containment hydrogen monitoring instrumentation allows the operators to detect trends in hydrogen concentration to diagnose the course of beyond design basis accidents. High hydrogen concentration is measured, continuously recorded, and displayed in the control room by a single instrument channel. The analyzer has the capability for sampling both the drywell and the suppression chamber. LCO 3.3.11 requires the primary containment hydrogen analyzer be OPERABLE.
APPLICABILITY	The primary containment hydrogen analyzer is required to be OPERABLE when primary containment is inerted, except as allowed by the relaxations during startup and shutdown addressed below. The primary containment must be inert in MODE 1, since this is the condition with the highest probability of an event that could produce hydrogen. Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and de-inerted as soon as possible in the plant shutdown. As long as reactor power is < 15% RTP, the potential for an event that generates significant hydrogen is low and the primary containment need not be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a startup, or within the last 24 hours before a shutdown, is low enough that these "windows," when the primary containment is not inerted, are also justified. The 24 hour time period is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.
ACTIONS	A.1 Seven days to restore the primary containment hydrogen analyzer capability is reasonable given the requirement to be available for use in diagnosing beyond design basis events.

Table B 3.3.3.2-1 (Page 2 of 4)
Backup Control System Instrumentation and Controls

FUNCTION	NUMBER REQUIRED
Transfer/Control Parameter (continued)	
14. RHRSW Pumps (0-43-23-15, -19, -23, -27) (0-HS-23-15C, -19C, -23C, -27C); EECW Pumps (north header) (0-43-23-1, -85, -8, -91) (0-HS-23-1C, -85C, -8C, -91C); EECW Pumps (south header) (0-43-23-15, -88, -23, -94) (0-HS-23-15C, -88C, -23C, -94C)	note e
15. RHRSW Discharge Valves for RHR Loop II Heat Exchangers (1-XS-23-46, -52) (1-HS-23-46C, -52C)	2, note f
16. RCW Pumps 1D and 3D (Trip Function Only) (1-XS-24-16, 3-XS-24-16) (1-HS-24-16C, 3-HS-24-16C)	2, note g
17. Recirculation System Sample Line Isolation Valves (1-XS-43-13, -14) (1-HS-43-13C, -14C)	1, note i
18. EECW Sectionalizing Valves (0-XS-67-13, -14, -17, -18, -21, -22, -25, -26) (0-HS-67-13C, -14C, -17C, -18C, -21C, -22C, -25C, -26C)	8, 1 per valve, note j
19. (Removed by Revision 88)	
20. Recirculation Pump 1A Discharge Valve (1-XS-68-3) (1-HS-68-3C)	1
21. RWCU Drain to Main Condenser Hotwell Isolation Valve (1-XS-69-16) (1-HS-69-16C)	1
note e: There are 12 RHRSW pumps. All are equipped with emergency transfer switches. Backup Control must be available for 2 OPERABLE pumps aligned for EECW service (supports all units). Backup control for an additional 2 OPERABLE pumps must be available for RHRSW for RHR Loop II.	
note f: 1 Discharge Valve per RHR Loop II Heat Exchanger for a total of 2.	
note g: 1 per pump. Trip function necessary to prevent spurious start overloading 4-kV Buses/Diesel Generators.	
note h: Deleted.	
note i: 1 Recirculation System Sample Line Isolation Valve required, may be either inboard valve or outboard valve.	
note j: Not required if valve breaker remains open, per NFPA 805 requirement, except when required for valve testing or operation.	

BASES

BACKGROUND (continued)	There are two motor breakers provided for each of the two recirculation pumps for a total of four breakers. The output of each trip system is provided to one of the two breakers for each recirculation pump.
APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY	The ATWS-RPT initiates an RPT to aid in preserving the integrity of the fuel cladding following events in which a scram does not, but should, occur. Based on its contribution to the reduction of overall plant risk, however, the instrumentation meets Criterion 4 of the NRC Policy Statement (Ref. 3). The OPERABILITY of the ATWS-RPT is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have a required number of OPERABLE channels in each trip system, with their setpoints within the specified Allowable Value of SR 3.3.4.2.3. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint). ATWS-RPT Channel OPERABILITY also includes the associated recirculation pump motor breakers. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. Allowable Values are specified for each ATWS-RPT Function specified in the LCO. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between 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 predetermined values of output at which an action should take place. The setpoints are compared to the actual process

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

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

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

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

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

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

5.g. SLC System Initiation

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 4). An isolation signal for both RWCU isolation valves is initiated when the SLC pump start handswitch is not in the stop position.

There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch.

The SLC System Initiation Function is required to be OPERABLE in MODES 1 and 2, because these are the only MODES where the reactor can be critical, and in MODE 3 because this MODE uses the SLC System sodium pentaborate as a buffering solution to maintain the pH level at or above 7 in the suppression pool in the event of a LOCA. These MODES are consistent with the Applicability for the SLC System (LCO 3.1.7).

As noted (footnote (a) to Table 3.3.6.1-1), the SLC initiation signal provides input to the isolation logic for both RWCU isolation valves.

5.h. Reactor Vessel Water Level - Low, Level 3
(LIS-3-203A-D)

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some interfaces with the reactor vessel occurs to isolate the potential sources of a break. The isolation of the RWCU System on Level 3 supports actions to ensure that the fuel peak cladding

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The PCIVs LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory, and establishing the primary containment boundary during major accidents. As part of the primary containment boundary, PCIV OPERABILITY supports leak tightness of primary containment. Therefore, the safety analysis of any event requiring isolation of primary containment is applicable to this LCO.

The DBAs that result in a release of radioactive material and are mitigated by PCIVs are a LOCA and a main steam line break (MSLB). In the analysis for each of these accidents, it is assumed that PCIVs are either closed or close within the required isolation times following event initiation. This ensures that potential paths to the environment through PCIVs (including primary containment purge valves) are minimized. Of the events analyzed in Reference 1, the LOCA is the most limiting event due to radiological consequences.

The closure time of the main steam isolation valves (MSIVs) is a significant variable from a radiological standpoint. The MSIVs are required to close within 3 to 5 seconds since the 5 second closure time is consistent with or conservative to the times assumed in the analyses. The safety analyses assume that the purge valves were closed at event initiation. Likewise, it is assumed that the primary containment is isolated such that release of fission products to the environment is controlled.

The DBA analysis assumes that primary containment is isolated within a certain time period (based on PCIV closure times provided in Reference 2 and inboard MSIV closure times of 2 minutes for a LOCA per Reference 9) and that leakage is terminated, except for the maximum allowable leakage rate, L_a. The primary containment isolation total response time includes signal delay, diesel generator startup (for loss of offsite power), and PCIV stroke times.

(continued)

BASES

REFERENCES

1. FSAR, Section 14.6.
2. BFN Technical Instruction (TI), 0-TI-360.
3. 10 CFR 50, Appendix J, Option B.
4. FSAR, Section 5.2.
5. FSAR, Section 14.6.5.
6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
7. FSAR Table 5.2-2.
8. General Electric NEDO-32977-A (Boiling Water Reactor Owners Group Topical Report, B21-00658-01), "Excess Flow Check Valve Testing Relaxation", dated June 2000.
9. MDQ0000012016000566, Revision 0, "Main Steam Isolation Valve (MSIV) Loss of Coolant Accident (LOCA) Closure Analysis," dated September 2016.

BASES

APPLICABLE SAFETY ANALYSES (continued)	With two and three units fueled, a worse case single failure could also include the loss of two RHRSW pumps caused by losing a 4 kV shutdown board since there are certain alignment configurations that allow two RHRSW pumps to be powered from the same 4 kV shutdown board. As discussed in the FSAR, Section 14.6.3.3.2 (Ref. 4) for these analyses, manual initiation of the OPERABLE RHRSW subsystems and the associated RHR System is assumed to occur 10 minutes after a DBA. The analyses assume that there are two RHRSW subsystems operating in each unit, with one RHRSW pump in each subsystem capable of producing 4000 gpm of flow. In this case, the maximum suppression chamber water temperature and pressure are 187.3°F (as reported in Reference 6) and 30.5 psig, respectively, well below the design temperature of 281°F and maximum allowable pressure of 62 psig. The RHRSW System satisfies Criterion 3 of the NRC Policy Statement (Ref 5).
LCO	Four RHRSW subsystems are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst case single active failure occurs coincident with the loss of offsite power. Additionally, since the RHRSW pumps are shared between the three BFN units, the number of OPERABLE pumps required is also dependent on the number of units fueled. An RHRSW subsystem is considered OPERABLE when: a. At least one RHRSW pump (i.e., one required RHRSW pump) is OPERABLE; and b. An OPERABLE flow path is capable of taking suction from the intake structure and transferring the water to the associated RHR heat exchanger at the assumed flow rate.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Turbine Bypass System

BASES

BACKGROUND

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is 25% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of nine valves connected to the main steam lines between the main steam isolation valves and the turbine stop valve bypass valve chest. Each of these nine valves is operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Pressure Regulator and Turbine Generator Control System, as discussed in the FSAR, Section 7.11 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows from the bypass chest, through connecting piping, to the pressure breakdown assemblies, where a series of orifices are used to further reduce the steam pressure before the steam enters the condenser.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)	<u>SR 3.7.5.3</u> This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analysis. The response time limits are specified in the cycle specific transient analyses performed to support the preparation of FSAR, Appendix N, Supplemental Reload Licensing Report (Ref. 4). The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown the 24 month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint.
REFERENCES	1. FSAR, Section 7.11. 2. FSAR, Section 14.5.1.1. 3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993. 4. FSAR, Appendix N.

BASES

BACKGROUND (continued)

switchyards from the transmission network. One of the two required qualified offsite power circuits must be from the 500 kV transmission network to meet the requirements of 10 CFR 50, Appendix A, GDC 17 (Ref. 1). Four basic circuits from the transmission network to the safety related Division I (A and B 4.16 kV shutdown boards) and Division II (C and D 4.16 kV shutdown boards), are as follows:

1. From the 500 kV switchyard, through unit station service transformer (USST) 1B to a 4.16 kV unit board. That unit board feeds 4.16 kV shutdown bus 1 or 2, which then feeds two of the Unit 1 and 2 4.16 kV shutdown boards (A and B or C and D);

2. From the 500 kV switchyard, through USST 2B to a 4.16 kV unit board. That unit board feeds 4.16 kV shutdown bus 1 or 2, which then feeds two of the Unit 1 and 2 4.16 kV shutdown boards (A and B or C and D);

3. From the 161 kV transmission network, through common station service transformer (CSST) A to start bus 1A or 1B, then to a 4.16 kV unit board. That unit board feeds 4.16 kV shutdown bus 1 or 2, which then feeds two of the Unit 1 and 2 4.16 kV shutdown boards (A and B or C and D); and

4. From the 161 kV transmission network, through CSST B to start bus 1A or 1B, and then to a 4.16 kV unit board. That unit board feeds 4.16 kV shutdown bus 1 or 2, which then feeds two of the Unit 1 and 2 4.16 kV shutdown boards (A and B or C and D).

(continued)

BASES

LCO (continued)

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the A and B (Division I) or C and D (Division II) 4.16 kV shutdown boards. One of the two required qualified offsite power circuits must be from the 500 kV transmission network to meet the requirements of 10 CFR 50, Appendix A, GCD 17 (Ref. 1). Specific circuits are described below:

1. From the 500 kV switchyard through unit station service transformer (USST) 1B to 4.16 kV unit board 1A, to 4.16 kV shutdown bus 1, to 4.16 kV shutdown boards A and B; and/or, to 4.16 kV unit board 1B, to 4.16 kV shutdown bus 2, to 4.16 kV shutdown boards C and D.

2. From the 500 kV switchyard through USST 2B to 4.16 kV unit board 2A, to 4.16 kV shutdown bus 2, to 4.16 kV shutdown boards C and D; and/or, to 4.16 kV unit board 2B, to 4.16 kV shutdown bus 1, to 4.16 kV shutdown boards A and B.

3. From the 161 kV transmission network, through common station service transformer (CSST) A to start bus 1A or 1B, to 4.16 kV unit board 1A or 2B, to 4.16 kV shutdown bus 1, to 4.16 kV shutdown boards A and B; or alternately, to 4.16 kV unit board 1B or 2A, to 4.16 kV shutdown bus 2, to 4.16 kV shutdown boards C and D.

4. From the 161 kV transmission network, through CSST B to start bus 1A or 1B, to 4.16 kV unit board 1A or 2B, to 4.16 kV shutdown bus 1, to 4.16 kV shutdown boards A and B; or alternately, to 4.16 kV unit board 1B or 2A, to 4.16 kV shutdown bus 2, to 4.16 kV shutdown boards C and D.

(continued)

BASES

ACTIONS (continued)

J.1

Condition J corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

K.1

Required Action K.1 is intended to provide assurance that a loss of offsite power, during the period that a required Unit 3 DG is inoperable, does not result in a complete loss of safety function of critical systems (i.e., SGT, CREVS, RHRSW, or EECW). These features are designed with redundant safety related divisions (i.e., single division systems are not included). Redundant required features failures consist of inoperable features associated with a division redundant to the division that has an inoperable Unit 3 DG.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action the Completion Time only begins on discovery that both:

a. An inoperable required Unit 3 DG exists; and

b. A redundant required feature supported by another DG, is inoperable.

(continued)

BASES

ACTIONS

K.1 (continued)

If, at any time during the existence of this Condition (a required Unit 3 DG inoperable), a required redundant feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering a required Unit 3 DG inoperable coincident with an inoperable required redundant feature that is associated with the OPERABLE DGs results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period.

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

BASES

BACKGROUND

Each diesel generator (DG) is provided with three interconnected storage tanks having a minimum usable fuel oil volume (35,280 gallons) sufficient to operate that DG for a period of 7 days while the DG is supplying maximum post loss of coolant accident (LOCA) load demand discussed in FSAR, Section 8.5.3.4 (Ref. 1) and Regulatory Guide 1.137 (Ref. 2). A transfer pump is located at the fuel oil storage tanks which can supply fuel oil from two 71,000-gallon fuel oil storage tanks to the 7-day storage tanks. Only the 7-day storage tanks are subject to this LCO, the Actions, and the Surveillance Requirements. In addition, it is possible to transfer fuel from one 7-day storage tank to any other by using transfer pumps. This onsite fuel oil capacity is sufficient to operate the DGs for longer than the time to replenish the onsite supply from outside sources.

Fuel oil is transferred from the 7-day storage tank to the day tank by either of two transfer pumps associated with each diesel generator. This is accomplished automatically by level switches on the day tank. Redundancy of pumps and piping precludes the failure of one pump, or the rupture of any pipe, valve, or tank to result in the loss of more than one DG. All 7-day tanks are embedded in the substructure of the Standby Diesel Generator Building.

For proper operation of the standby DGs, it is necessary to ensure the proper quality of the stored fuel oil. The fuel oil property monitored is the total particulate concentration. Periodic testing of the stored fuel oil total particulate concentration is a method to monitor the potential degradation related to long term storage and the potential impact to fuel filter plugging as a result of high particulate levels.

(continued)

BASES (continued)

APPLICABILITY

The AC sources (LCO 3.8.1 and LCO 3.8.2) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an abnormal operational transient or a postulated DBA. Because stored diesel fuel oil, lube oil, and starting air subsystem support LCO 3.8.1 and LCO 3.8.2, stored diesel fuel oil, lube oil, and starting air are required to be within limits when the associated DG is required to be OPERABLE.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each DG. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DG subsystem. Complying with the Required Actions for one inoperable DG subsystem may allow for continued operation, and subsequent inoperable DG subsystem(s) governed by separate Condition entry and application of associated Required Actions.

A.1

In this condition, the 7-day fuel oil supply for a DG is not available. However, the Condition is restricted to fuel oil level reductions that maintain at least a 6-day supply. The fuel oil level equivalent to a 6-day supply is 30,240 gallons. These circumstances may be caused by events such as:

a. Full load operation required for an inadvertent start while at minimum required level; or

b. Feed and bleed operations that may be necessitated by increasing particulate levels or any number of other oil quality degradations.

(continued)

BASES

ACTIONS

A.1 (continued)

This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of the fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required level prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

B.1

In this Condition, the 7-day lube oil inventory, i.e., sufficient lube oil to support 7 days of continuous DG operation at full load conditions, is not available. However, the Condition is restricted to lube oil volume reductions that maintain at least a 6-day supply. The lube oil inventory equivalent to a 6-day supply is 150 gallons. This restriction allows sufficient time for obtaining the requisite replacement volume. A period of 48 hours is considered sufficient to complete restoration of the required volume prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the low rate of usage, the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

C.1

This Condition is entered as a result of a failure to meet the acceptance criterion for particulates. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence

(continued)

ACTIONS

C.1 (continued)

of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, since particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and since proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion Time allows for further evaluation, re-sampling, and re-analysis of the DG fuel oil.

D.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.3 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or a combination of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is high likelihood that the DG would still be capable of performing its intended function.

E.1

Only one of the two redundant air starting systems is required to support associated DG operability. With the starting air receiver pressure < 165 psig in the required starting air system, sufficient capacity to start the associated DG may not exist. The associated DG may be incapable of performing its intended function and must be immediately declared inoperable. This declaration also requires entry into applicable Conditions and Required Actions for an inoperable DG, LCO 3.8.1, "AC Sources - Operating."

(continued)

BASES

ACTIONS

F.1

With a Required Action and associated Completion Time not met, or the stored diesel fuel oil, lube oil, or starting air subsystem inoperable for reasons other than addressed by Conditions A through E, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support each DG's operation for 7 days at full load. The fuel oil level equivalent to a 7-day supply is 35,280 gallons when calculated in accordance with References 2 and 6. The required fuel storage volume is determined using the most limiting energy content of the stored fuel. Using the known correlation of diesel fuel oil absolute specific gravity or API gravity to energy content, the required diesel generator output, and the corresponding fuel consumption rate, the on site fuel storage volume required for 7 days of operation can be determined. SR 3.8.3.3 requires new fuel to be tested to verify that the absolute specific gravity or API gravity is within the range assumed in the diesel fuel oil consumption calculations. The 7-day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

The 31 day Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

SR 3.8.3.2

This Surveillance ensures that sufficient lubricating oil inventory is available to support at least 7 days of full load operation for each DG. The lube oil inventory equivalent to a 7-day supply is 175 gallons and is based on the DG manufacturer's consumption values for the run time of the DG.

A 31 day Frequency is adequate to ensure that a sufficient lube oil supply is onsite, since DG starts and run time are closely monitored by the plant staff.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.3

The tests listed below are a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate detrimental impact on diesel engine combustion. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. These tests are to be conducted prior to adding the new fuel to the 7-day storage tank(s), but in no case is the time between receipt of new fuel and conducting the tests to exceed 31 days following addition to the 7-day storage tanks. The tests, limits, and applicable ASTM Standards are as follows:

a. Sample the new fuel oil in accordance with ASTM D4057-12 (Ref. 7)

b. Verify in accordance with the tests specified in ASTM D975-14a (Ref. 7) that the sample has an absolute specific gravity at 60/60°F of ≥ 0.83 and ≤ 0.89 or an API gravity at 60°F of ≥ 27° and ≤ 39° when tested in accordance with ASTM D1298-12b (Ref. 7), a kinematic viscosity at 40°C of ≥ 1.9 centistokes and ≤ 4.1 centistokes, and a flash point of ≥ 125°F

c. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176-04 or a water and sediment content within limits when tested in accordance with ASTM D2709-96 (Ref. 7)

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO because the fuel oil is not added to the 7-day storage tanks.

Either prior to adding new fuel oil to the 7-day storage tanks or within 31 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975-14a (Ref. 7) are met for

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.3

new fuel oil when tested in accordance with ASTM D975-14a (Ref. 7), except that the analysis for sulfur may be performed in accordance with ASTM D4294-10 (Ref. 7). If these tests are not completed prior to adding new fuel oil to the 7-day storage tanks, the 31 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on DG operation. This Surveillance ensures the availability of high quality fuel oil for the DGs.

Fuel oil degradation during long term storage shows up as an increase in particulates, mostly due to oxidation. The presence of particulates does not mean that the fuel oil will not burn properly in a diesel engine. The particulates can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

Particulate concentrations should be determined in accordance with ASTM D6217-11 (Ref. 7). This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing. Because the 7-day tank consists of three interconnected tanks, samples are drawn from each tank.

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

SR 3.8.3.4

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The system design requirements provide for at least one start cycle from one of two redundant air start systems without recharging. A start cycle is defined by the DG vendor, but usually is measured in terms of time (seconds of cranking) or engine cranking speed. The pressure specified in this SR is the lowest pressure at which at least one start attempt can be accomplished using one of two redundant air start systems.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.4

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.

SR 3.8.3.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel storage tanks once every 31 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 2). This SR is for preventive maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during performance of the Surveillance.

REFERENCES

1. FSAR, Section 8.5.3.4.
2. Regulatory Guide 1.137, Revision 1, October 1979.
3. FSAR, Chapter 6.
4. FSAR, Chapter 14.
5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

REFERENCES (continued)

6. ANSI N195-1976, "Fuel Oil Systems for Standby Diesel
7. ASTM Standards, D4057-12; D4176-04; D2709-96, D1298-12b, D975-14a, D4294-10, D6217-11.

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B 3.0-1	0	Initial
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B 3.0-3	0	Initial
B 3.0-4	Revision 93	10-23-2015
B 3.0-4a	Revision 0	Initial
B 3.0-5	0	Initial
B 3.0-6	Amendment 286	12-01-2003
B 3.0-7	Amendment 286	12-01-2003
B 3.0-8	Amendment 286 and Revision 24	12-01-2003 and 01-29-2004
B 3.0-8a	Amendment 286 and Revision 240	12-01-2003 and 01-29-2004
B 3.0-8b	Amendment 286	12-01-2003
B 3.0-9	0	Initial
B 3.0-10	Amendment 266	11-21-2000

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B 3.0-10a	Revision 12	03-06-2001
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B 3.0-11	0	Initial
B 3.0-12	0	Initial
B 3.0-13	0	Initial
B 3.0-14	0	Initial
B 3.0-15	Revision 83	09-21-2011
B 3.0-15a	Revision 83	09-21-2011
B 3.0-16	Amendment 278	12-23-2002
B 3.0-16a	Amendment 278	12-23-2002
B 3.0-17	Amendment 286	12-01-2003
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B 3.1-1	Revision 61	12-07-2010
B 3.1-2	0	Initial
B 3.1-3	0	Initial
B 3.1-4	0	Initial
B 3.1-5	0	Initial
B 3.1-6	Revision 61	12-07-2010
B 3.1-7	0	Initial
B 3.1-8	Revision 61	12-07-2010
B 3.1-9	0	Initial
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B 3.1-19	Amendment 301	06-26-2009
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B 3.1-31	Revision 55	03-20-2008
B 3.1-32	0	Initial
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B 3.1-35	0	Initial
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B 3.1-41	Revision 61	12-07-2010
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B 3.1-42a	Amendment 303	11-19-2009
B 3.1-43	0	Initial
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B 3.1-46	Revision 61	12-07-2010
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B 3.2-1	Amendment 313	02-26-2015
B 3.2-2	Revision 106	10-27-2016
B 3.2-3	Revision 106	10-27-2016

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B 3.2-5	Revision 106	10-27-2016
B 3.2-5a	Revision 106	10-27-2016
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B 3.2-8	Revision 106	10-27-2016
B 3.2-9	0	Initial
B 3.2-10	Revision 31	04-06-2005
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B 3.2-11a	Revision 61	12-07-2010
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B 3.2-14	0	Initial
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B 3.3-1	0	Initial
B 3.3-2	Amendment 276	04-08-2002
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B 3.3-4	Revision 41	11-09-2006
B 3.3-4a	Revision 41	11-09-2006
B 3.3-5	0	Initial
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B 3.3-7	0	Initial
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B 3.3-9	Amendment 258	03-05-1999
B 3.3-9a	Amendment 258	03-05-1999
B 3.3-10	0	Initial
B 3.3-11	Revision 86	10-29-2014
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B 3.3-14	Amendment 258	03-05-1999
B 3.3-15	Amendment 258	03-05-1999
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B 3.3-16	0	Initial
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B 3.3-28	Amendment 276	04-08-2002
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B 3.3-101	Revision 4	04-09-1999
B 3.3-102	Amendment 286	12-01-2003
B 3.3-103	Amendment 255	11-30-1998
B 3.3-104	Amendment 255	11-30-1998
B 3.3-105	Revision 4	04-09-1999
B 3.3-106	Revision 96	03-03-2016
B 3.3-107	Revision 4	04-09-1999
B 3.3-107a	Revision 4	04-09-1999
B 3.3-108	Revision 31	04-06-2005
B 3.3-109	Revision 61	12-07-2010
B 3.3-110	0	Initial
B 3.3-111	Revision 61	12-07-2010
B 3.3-112	Revision 61	12-07-2010
B 3.3-113	Revision 31	04-06-2005
B 3.3-114	0	Initial
B 3.3-115	Revision 61	12-07-2010
B 3.3-116	Revision 61	12-07-2010
B 3.3-117	0	Initial
B 3.3-118	Amendment 255	11-30-1998
B 3.3-119	Revision 31	04-06-2005
B 3.3-120	0	Initial
B 3.3-121	Revision 107	05-02-2017
B 3.3-122	0	Initial
B 3.3-123	0	Initial
B 3.3-124	0	Initial
B 3.3-125	0	Initial
B 3.3-126	0	Initial
B 3.3-127	0	Initial
B 3.3-128	0	Initial
B 3.3-129	Amendment 255	11-30-1998
B 3.3-130	Amendment 255	11-30-1998
B 3.3-131	0	Initial

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B 3.3-132	0	Initial
B 3.3-133	0	Initial
B 3.3-134	0	Initial
B 3.3-135	0	Initial
B 3.3-136	0	Initial
B 3.3-137	0	Initial
B 3.3-138	0	Initial
B 3.3-139	Revision 41	11-09-2006
B 3.3-140	Revision 41	11-09-2006
B 3.3-141	0	Initial
B 3.3-142	Revision 41	11-09-2006
B 3.3-143	Revision 41	11-09-2006
B 3.3-144	Revision 41	11-09-2006
B 3.3-144a	Revision 41	11-09-2006
B 3.3-144b	Revision 41	11-09-2006
B 3.3-144c	Revision 41	11-09-2006
B 3.3-145	0	Initial
B 3.3-146	Revision 47	03-22-2007
B 3.3-147	Revision 47	03-22-2007
B 3.3-148	Revision 65	05-29-2012
B 3.3-148a	Revision 65	05-29-2012
B 3.3-148b	Revision 47	03-22-2007
B 3.3-149	Revision 41	11-09-2006
B 3.3-149a	Revision 41	11-09-2006
B 3.3-150	0	Initial
B 3.3-151	Revision 41	11-09-2006
B 3.3-151a	Revision 41	11-09-2006
B 3.3-152	Revision 41	11-09-2006
B 3.3-152a	Revision 41	11-09-2006
B 3.3-153	0	Initial
B 3.3-154	0	Initial
B 3.3-155	0	Initial
B 3.3-156	0	Initial
B 3.3-157	Revision 41	11-09-2006
B 3.3-157a	Revision 41	11-09-2006
B 3.3-158	Revision 41	11-09-2006
B 3.3-158a	Revision 41	11-09-2006
B 3.3-159	0	Initial
B 3.3-159a	Revision 41	11-09-2006
B 3.3-160	0	Initial

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B 3.3-161	0	Initial
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B 3.3-167	0	Initial
B 3.3-168	0	Initial
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B 3.3-171	0	Initial
B 3.3-172	0	Initial
B 3.3-173	0	Initial
B 3.3-174	0	Initial
B 3.3-175	Revision 13	04-11-2001
B 3.3-176	Revision 13	04-11-2001
B 3.3-177	Revision 65	05-29-2012
B 3.3-178	0	Initial
B 3.3-179	0	Initial
B 3.3-180	0	Initial
B 3.3-181	Revision 41	11-09-2006
B 3.3-182	Revision 41	11-09-2006
B 3.3-182a	Revision 41	11-09-2006
B 3.3-183	0	Initial
B 3.3-184	0	Initial
B 3.3-185	0	Initial
B 3.3-186	0	Initial
B 3.3-187	0	Initial
B 3.3-188	Amendment 255	11-30-1998
B 3.3-189	Amendment 255	11-30-1998
B 3.3-190	0	Initial
B 3.3-191	0	Initial
B 3.3-192	0	Initial
B 3.3-193	0	Initial
B 3.3-194	0	Initial
B 3.3-195	0	Initial
B 3.3-196	0	Initial
B 3.3-197	0	Initial
B 3.3-198	Revision 29	01-25-2005
B 3.3-199	Revision 97	02-05-2016

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B 3.3-199a	Revision 41	11-09-2006
B 3.3-200	Revision 29	01-25-2005
B 3.3-201	Amendment 277	11-26-2002
B 3.3-202	Revision 29	01-25-2005
B 3.3-203	Revision 29	01-25-2005
B 3.3-204	0	Initial
B 3.3-205	0	Initial
B 3.3-206	0	Initial
B 3.3-207	0	Initial
B 3.3-208	0	Initial
B 3.3-209	Revision 98	02-05-2016
B 3.3-210	0	Initial
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B 3.3-212	0	Initial
B 3.3-213	Revision 29	01-25-2005
B 3.3-214	0	Initial
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B 3.3-216	0	Initial
B 3.3-217	0	Initial
B 3.3-218	0	Initial
B 3.3-219	0	Initial
B 3.3-220	0	Initial
B 3.3-221	0	Initial
B 3.3-222	0	Initial
B 3.3-223	0	Initial
B 3.3-224	Amendment 255	11-30-1998
B 3.3-225	0	Initial
B 3.3-226	Revision 35	02-14-2006
B 3.3-227	0	Initial
B 3.3-228	0	Initial
B 3.3-229	0	Initial
B 3.3-230	0	Initial
B 3.3-231	Revision 35	02-14-2006
B 3.3-232	Revision 35	02-14-2006
B 3.3-233	0	Initial
B 3.3-234	0	Initial
B 3.3-235	0	Initial
B 3.3-236	0	Initial
B 3.3-237	0	Initial
B 3.3-238	Revision 43	01-17-2007

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B 3.3-239	Revision 29	01-25-2005
B 3.3-240	Revision 35	02-14-2006
B 3.3-241	Revision 35	02-14-2006
B 3.3-242	0	Initial
B 3.3-243	0	Initial
B 3.3-244	0	Initial
B 3.3-245	Revision 35	02-14-2006
B 3.3-246	Revision 29	01-25-2005
B 3.3-247	0	Initial
B 3.3-248	0	Initial
B 3.3-249	0	Initial
B 3.3-250	Revision 27	07-14-2004
B 3.3-251	0	Initial
B 3.3-252	0	Initial
B 3.3-253	0	Initial
B 3.3-254	Revision 43	01-17-2007
B 3.3-255	Amendment 255	11-30-1998
B 3.3-256	0	Initial
B 3.3-257	0	Initial
B 3.3-258	0	Initial
B 3.3-259	0	Initial
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B 3.3-261	0	Initial
B 3.3-262	0	Initial
B 3.3-263	0	Initial
B 3.3-264	0	Initial
B 3.3-265	0	Initial
B 3.3-266	Amendment 255	11-30-1998
B 3.3-267	0	Initial
B 3.3-268	0	Initial
B 3.3-269	0	Initial
B 3.3-270	0	Initial
B 3.3-271	0	Initial
B 3.3-272	0	Initial
B 3.3-273	0	Initial
B 3.3-274	0	Initial
B 3.3-275	0	Initial
B 3.3-276	0	Initial
B 3.3-277	Amendment 255	11-30-1998
B 3.4-1	Revision 20	03-20-2003

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B 3.4-2	Revision 20	03-20-2003
B 3.4-3	Revision 87	10-29-2014
B 3.4-4	Amendment 258 and Revision 3	03-05-1999 and 03-19-1999
B 3.4-5	Amendment 258	03-05-1999
B 3.4-6	Amendment 258	03-05-1999
B 3.4-7	Amendment 258	03-05-1999
B 3.4-8	Amendment 258	03-05-1999
B 3.4-9	Amendment 258	03-05-1999
B 3.4-10	Revision 87	10-29-2014
B 3.4-11	0	Initial
B 3.4-12	0	Initial
B 3.4-13	0	Initial
B 3.4-14	Amendment 256	12-23-1998
B 3.4-15	0	Initial
B 3.4-16	0	Initial
B 3.4-17	0	Initial
B 3.4-18	0	Initial
B 3.4-19	0	Initial
B 3.4-20	0	Initial
B 3.4-21	0	Initial
B 3.4-22	Revision 81	10-16-2013
B 3.4-23	0	Initial
B 3.4-24	0	Initial
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B 3.4-26	0	Initial
B 3.4-27	0	Initial
B 3.4-28	0	Initial
B 3.4-29	0	Initial
B 3.4-30	0	Initial
B 3.4-31	0	Initial
B 3.4-32	Revision 77	02-15-2011
B 3.4-33	0	Initial
B 3.4-34	Amendment 286	12-01-2003
B 3.4-35	0	Initial
B 3.4-36	Amendment 255	11-30-1998
B 3.4-37	Revision 29	01-25-2005
B 3.4-38	Revision 29	01-25-2005
B 3.4-39	Amendment 286	12-01-2003
B 3.4-40	Revision 29	01-25-2005

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B 3.4-41	Revision 29	01-25-2005
B 3.4-42	0	Initial
B 3.4-43	0	Initial
B 3.4-44	0	Initial
B 3.4-45	Amendment 286	12-01-2003
B 3.4-46	0	Initial
B 3.4-47	0	Initial
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B 3.4-51	0	Initial
B 3.4-52	0	Initial
B 3.4-53	0	Initial
B 3.4-54	0	Initial
B 3.4-55	Revision 92	07-30-2015
B 3.4-55a	Revision 92	07-30-2015
B 3.4-56	Revision 92	07-30-2015
B 3.4-56a	Revision 26	03-17-2004
B 3.4-56b	Revision 26	03-17-2004
B 3.4-57	Revision 26	03-17-2004
B 3.4-58	Revision 26	03-17-2004
B 3.4-59	0	Initial
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B 3.4-62	0	Initial
B 3.4-63	Revision 92	07-30-2015
B 3.4-64	0	Initial
B 3.4-65	Revision 26	03-17-2004
B 3.4-66	Revision 26	03-17-2004
B 3.4-66a	Revision 92	07-30-2015
B 3.4-67	Revision 31	04-06-2005
B 3.4-68	Amendment 254	09-08-1998
B 3.4-69	Amendment 254	09-08-1998
B 3.4-70	0	Initial
B 3.5-1	0	Initial
B 3.5-2	0	Initial
B 3.5-3	Revision 76	04-25-2013
B 3.5-3a	Revision 65	05-29-2012
B 3.5-4	Revision 47	03-22-2007
B 3.5-5	Revision 80	10-04-2013

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B 3.5-6	0	Initial
B 3.5-7	0	Initial
B 3.5-8	Amendment 286	12-01-2003
B 3.5-8a	Amendment 286	12-01-2003
B 3.5-9	0	Initial
B 3.5-10	Amendment 269	03-12-2001
B 3.5-11	Amendment 269	03-12-2001
B 3.5-12	0	Initial
B 3.5-13	0	Initial
B 3.5-14	0	Initial
B 3.5-15	0	Initial
B 3.5-16	Revision 81	10-16-2013
B 3.5-17	Amendment 255	11-30-1998
B 3.5-18	Amendment 255	11-30-1998
B 3.5-19	Amendment 255	11-30-1998
B 3.5-20	0	Initial
B 3.5-21	Revision 76	04-25-2013
B 3.5-22	0	Initial
B 3.5-23	0	Initial
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B 3.5-25	0	Initial
B 3.5-26	0	Initial
B 3.5-27	0	Initial
B 3.5-28	0	Initial
B 3.5-29	0	Initial
B 3.5-30	Revision 6	08-17-1999
B 3.5-31	Amendment 254	09-08-1998
B 3.5-32	Amendment 286	12-01-2003
B 3.5-32a	Amendment 286	12-01-2003
B 3.5-33	0	Initial
B 3.5-34	0	Initial
B 3.5-35	Revision 53	05-18-2007
B 3.5-36	Amendment 255	11-30-1998
B 3.5-37	Amendment 255	11-30-1998
B 3.5-38	0	Initial
B 3.6-1	0	Initial
B 3.6-2	0	Initial
B 3.6-3	Amendment 254	09-08-1998
B 3.6-4	0	Initial
B 3.6-5	0	Initial

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B 3.6-6	Amendment 255	11-30-1998
B 3.6-7	0	Initial
B 3.6-8	Amendment 254	09-08-1998
B 3.6-9	0	Initial
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B 3.6-16	0	Initial
B 3.6-17	0	Initial
B 3.6-18	0	Initial
B 3.6-19	0	Initial
B 3.6-20	Revision 105	09-01-2016
B 3.6-21	Revision 3	03-19-1999
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B 3.6-25	0	Initial
B 3.6-26	0	Initial
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B 3.6-29	0	Initial
B 3.6-30	0	Initial
B 3.6-31	0	Initial
B 3.6-32	0	Initial
B 3.6-33	Revision 29	01-25-2005
B 3.6-34	Revision 84	08-29-2014
B 3.6-34a	Amendment 268	01-29-2001
B 3.6-35	Revision 62	01-12-2012
B 3.6-36	Revision 105	09-01-2016
B 3.6-37	Amendment 254	09-08-1998
B 3.6-38	0	Initial
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B 3.6-40	0	Initial
B 3.6-41	0	Initial
B 3.6-42	0	Initial
B 3.6-43	0	Initial
B 3.6-44	0	Initial

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B 3.6-45	0	Initial
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B 3.6-47	0	Initial
B 3.6-48	Amendment 255	11-30-1998
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B 3.6-56	Amendment 255	11-30-1998
B 3.6-57	0	Initial
B 3.6-58	Revision 85	09-30-2014
B 3.6-59	0	Initial
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B 3.6-62	0	Initial
B 3.6-63	Revision 85	09-30-2014
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B 3.6-68	0	Initial
B 3.6-69	0	Initial
B 3.6-70	0	Initial
B 3.6-71	Amendment 272	06-08-2001
B 3.6-72	Amendment 272	06-08-2001
B 3.6-73	Revision 81	10-16-2013
B 3.6-74	0	Initial
B 3.6-75	0	Initial
B 3.6-76	0	Initial
B 3.6-77	0	Initial
B 3.6-78	0	Initial
B 3.6-79	0	Initial
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B 3.6-82	0	Initial
B 3.6-83	0	Initial
B 3.6-84	0	Initial

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B 3.6-85	Revision 3	03-19-1999
B 3.6-86	0	Initial
B 3.6-87	0	Initial
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B 3.6-89	0	Initial
B 3.6-90	0	Initial
B 3.6-91	0	Initial
B 3.6-92	0	Initial
B 3.6-93	Amendment 286	12-01-2003
B 3.6-94	Amendment 265	05-24-2000
B 3.6-95	Amendment 265	05-24-2000
B 3.6-96	Amendment 265	05-24-2000
B 3.6-97	0	Initial
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B 3.6-99	0	Initial
B 3.6-100	0	Initial
B 3.6-101	0	Initial
B 3.6-102	Revision 29	01-25-2005
B 3.6-103	Revision 29	01-25-2005
B 3.6-104	Revision 29	01-25-2005
B 3.6-105	Revision 29	01-25-2005
B 3.6-106	Revision 29	01-25-2005
B 3.6-107	Revision 29	01-25-2005
B 3.6-108	Revision 29	01-25-2005
B 3.6-109	Revision 29	01-25-2005
B 3.6-110	0	Initial
B 3.6-111	0	Initial
B 3.6-112	Revision 29	01-25-2005
B 3.6-113	Revision 29	01-25-2005
B 3.6-114	0	Initial
B 3.6-115	Revision 29	01-25-2005
B 3.6-116	Revision 29	01-25-2005
B 3.6-117	0	Initial
B 3.6-118	Revision 29	01-25-2005
B 3.6-119	Revision 29	01-25-2005
B 3.6-120	Revision 108	10-27-2017
B 3.6-121	Amendment 255	11-30-1998
B 3.7-1	Revision 73	01-03-2013
B 3.7-2	Revision 73	01-03-2013
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B 3.7-4	Revision 73	01-03-2013
B 3.7-4a	Revision 73	01-03-2013
B 3.7-5	Revision 73	01-03-2013
B 3.7-5a	Revision 73	01-03-2013
B 3.7-5b	Revision 73	01-03-2013
B 3.7-6	Revision 73	01-03-2013
B 3.7-6a	Revision 73	01-03-2013
B 3.7-7	Revision 73	01-03-2013
B 3.7-8	Revision 73	01-03-2013
B 3.7-9	Revision 73	01-03-2013
B 3.7-10	Revision 69	10-05-2012
B 3.7-11	0	Initial
B 3.7-12	0	Initial
B 3.7-13	Amendment 254	09-08-1998
B 3.7-14	Revision 69	10-05-2012
B 3.7-15	0	Initial
B 3.7-16	Amendment 255	11-30-1998
B 3.7-17	Revision 90	02-26-2015
B 3.7-17a	Amendment 302	10-16-2009
B 3.7-18	Revision 90	02-26-2015
B 3.7-19	Revision 67	08-10-2012
B 3.7-19a	Revision 67	08-10-2012
B 3.7-20	Revision 67	08-10-2012
B 3.7-21	Amendment 302	10-16-2009
B 3.7-21a	Revision 67	08-10-2012
B 3.7-22	Revision 67	08-10-2012
B 3.7-23	Revision 67	08-10-2012
B 3.7-24	Revision 108	10-27-2017
B 3.7-25	Amendment 302	10-16-2009
B 3.7-25a	Revision 67	08-10-2012
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B 3.7-29	0	Initial
B 3.7-30	0	Initial
B 3.7-31	Amendment 255	11-30-1998
B 3.7-32	Revision 103	10-15-2015
B 3.7-33	Revision 61	12-07-2010
B 3.7-34	Revision 31	04-06-2005
B 3.7-35	Amendment 255	11-30-1998

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B 3.7-36	Revision 103	10-15-2016
B 3.7-37	Revision 29	01-25-2005
B 3.7-38	0	Initial
B 3.7-39	Revision 29	01-25-2005
B 3.8-1	Revision 52	05-11-2007
B 3.8-2	Revision 100	06-09-2016
B 3.8-3	Revision 42	11-16-2006
B 3.8-3a	Revision 42	11-16-2006
B 3.8-4	Revision 47	03-22-2007
B 3.8-4a	Revision 47	03-22-2007
B 3.8-5	Amendment 307	10-05-2011
B 3.8-5a	Amendment 307	10-05-2011
B 3.8-6	Revision 52	05-11-2007
B 3.8-7	Revision 100	06-09-2016
B 3.8-8	Revision 52	05-11-2007
B 3.8-9	Revision 52	05-11-2007
B 3.8-10	0	Initial
B 3.8-11	Amendment 286	12-01-2003
B 3.8-12	0	Initial
B 3.8-13	0	Initial
B 3.8-14	Amendment 307	10-05-2011
B 3.8-15	Amendment 307	10-05-2011
B 3.8-15a	Amendment 307	10-05-2011
B 3.8-16	Amendment 307	10-05-2011
B 3.8-17	Amendment 307	10-05-2011
B 3.8-18	Amendment 307	10-05-2011
B 3.8-18a	Amendment 307	10-05-2011
B 3.8-19	Amendment 307	10-05-2011
B 3.8-20	Revision 7	09-17-1999
B 3.8-21	0	Initial
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B 3.8-24	0	Initial
B 3.8-25	Revision 101	08-18-2016
B 3.8-26	Revision 101	08-18-2016
B 3.8-27	Revision 28	08-26-2004
B 3.8-28	Revision 28	08-26-2004
B 3.8-29	0	Initial
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B 3.8-34	Amendment 255	11-30-1998
B 3.8-35	Amendment 255	11-30-1998
B 3.8-36	Revision 42	11-16-2006
B 3.8-37	Revision 42	11-16-2006
B 3.8-38	0	Initial
B 3.8-39	0	Initial
B 3.8-40	0	Initial
B 3.8-41	Revision 52	05-11-2007
B 3.8-42	Revision 52	05-11-2007
B 3.8-43	Revision 52	05-11-2007
B 3.8-44	Revision 52	05-11-2007
B 3.8-45	0	Initial
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B 3.8-48	0	Initial
B 3.8-49	Revision 95	01-21-2016
B 3.8-50	0	Initial
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B 3.8-52	Revision 95	01-21-2016
B 3.8-53	Revision 95	01-21-2016
B 3.8-54	Revision 95	01-21-2016
B 3.8-54a	Revision 95	01-21-2016
B 3.8-55	Revision 95	01-21-2016
B 3.8-56	Revision 95	01-21-2016
B 3.8-56a	Revision 95	01-21-2016
B 3.8-56b	Revision 95	01-21-2016
B 3.8-56c	Revision 95	01-21-2016
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B 3.8-58	0	Initial
B 3.8-59	0	Initial
B 3.8-60	0	Initial
B 3.8-61	0	Initial
B 3.8-62	0	Initial
B 3.8-63	0	Initial
B 3.8-64	0	Initial
B 3.8-65	Amendment 255	11-30-1998
B 3.8-66	Revision 58	10-01-2008
B 3.8-67	0	Initial

(continued)

Page No.	Amendment / Revision No.	Effective Date
B 3.8-68	Amendment 255	11-30-1998
B 3.8-69	0	Initial
B 3.8-70	0	Initial
B 3.8-71	0	Initial
B 3.8-72	0	Initial
B 3.8-73	0	Initial
B 3.8-74	0	Initial
B 3.8-75	0	Initial
B 3.8-76	0	Initial
B 3.8-77	0	Initial
B 3.8-78	0	Initial
B 3.8-79	0	Initial
B 3.8-80	0	Initial
B 3.8-81	0	Initial
B 3.8-82	0	Initial
B 3.8-83	0	Initial
B 3.8-84	0	Initial
B 3.8-85	0	Initial
B 3.8-86	Revision 76	04-25-2013
B 3.8-87	Revision 91	06-17-2015
B 3.8-87a	Revision 76	04-25-2013
B 3.8-88	0	Initial
B 3.8-89	Revision 36	06-22-2006
B 3.8-90	0	Initial
B 3.8-91	0	Initial
B 3.8-92	0	Initial
B 3.8-93	Revision 76	04-25-2013
B 3.8-94	0	Initial
B 3.8-95	0	Initial
B 3.8-96	0	Initial
B 3.8-97	0	Initial
B 3.8-98	0	Initial
B 3.8-99	0	Initial
B 3.8-100	0	Initial
B 3.8-101	0	Initial
B 3.8-102	0	Initial
B 3.8-103	0	Initial
B 3.8-104	0	Initial
B 3.8-105	Revision 76	04-25-2013
B 3.8-106	0	Initial

(continued)

Page No.	Amendment / Revision No.	Effective Date
B 3.8-107	0	Initial
B 3.8-108	0	Initial
B 3.9-1	0	Initial
B 3.9-2	0	Initial
B 3.9-3	0	Initial
B 3.9-4	Amendment 274	03-06-2002
B 3.9-5	Amendment 274 and Revision 16	03-06-2002 and 03-21-2002
B 3.9-5a	Revision 60	10-16-2009
B 3.9-6	0	Initial
B 3.9-7	0	Initial
B 3.9-8	0	Initial
B 3.9-9	0	Initial
B 3.9-10	Revision 60	10-16-2009
B 3.9-11	0	Initial
B 3.9-12	0	Initial
B 3.9-13	Revision 60	10-16-2009
B 3.9-14	0	Initial
B 3.9-15	0	Initial
B 3.9-16	0	Initial
B 3.9-17	0	Initial
B 3.9-18	Revision 60	10-16-2009
B 3.9-19	0	Initial
B 3.9-20	0	Initial
B 3.9-21	0	Initial
B 3.9-22	Revision 60	10-16-2009
B 3.9-23	Revision 29	01-25-2005
B 3.9-24	0	Initial
B 3.9-25	Revision 29	01-25-2005
B 3.9-26	0	Initial
B 3.9-27	0	Initial
B 3.9-28	0	Initial
B 3.9-29	0	Initial
B 3.9-30	0	Initial
B 3.9-31	0	Initial
B 3.9-32	0	Initial
B 3.9-33	0	Initial
B 3.9-34	0	Initial
B 3.9-35	0	Initial
B 3.9-36	Amendment 290	09-27-2004

(continued)

Page No.	Amendment / Revision No.	Effective Date
B 3.9-37	Amendment 290	09-27-2004
B 3.9-38	Amendment 290	09-27-2004
B 3.10-1	Revision 51	05-10-2007
B 3.10-2	Revision 51	05-10-2007
B 3.10-3	Revision 51	05-10-2007
B 3.10-4	Revision 51	05-10-2007
B 3.10-4a	Revision 51	05-10-2007
B 3.10-5	0	Initial
B 3.10-6	0	Initial
B 3.10-7	0	Initial
B 3.10-8	0	Initial
B 3.10-9	0	Initial
B 3.10-10	0	Initial
B 3.10-11	0	Initial
B 3.10-12	0	Initial
B 3.10-13	0	Initial
B 3.10-14	0	Initial
B 3.10-15	0	Initial
B 3.10-16	0	Initial
B 3.10-17	0	Initial
B 3.10-18	0	Initial
B 3.10-19	0	Initial
B 3.10-20	0	Initial
B 3.10-21	Revision 4	04-09-1999
B 3.10-22	0	Initial
B 3.10-23	0	Initial
B 3.10-24	0	Initial
B 3.10-25	0	Initial
B 3.10-26	0	Initial
B 3.10-27	Revision 4	04-09-1999
B 3.10-28	0	Initial
B 3.10-29	0	Initial
B 3.10-30	0	Initial
B 3.10-31	0	Initial

(continued)

Page No.	Amendment / Revision No.	Effective Date
B 3.10-32	0	Initial
B 3.10-33	0	Initial
B 3.10-34	0	Initial
B 3.10-35	0	Initial
B 3.10-36	Revision 61	12-07-2010
B 3.10-37	Revision 31	04-06-2005
B 3.10-38	Revision 61	12-07-2010
B 3.10-39	0	Initial
B 3.10-40	Revision 61	12-07-2010
B 3.10-41	0	Initial
B 3.10-42	Revision 61	12-07-2010
B 3.10-43	0	Initial
B 3.10-44	0	Initial
B 3.10-45	0	Initial
B 3.10-46	0	Initial
B 3.10-47	0	Initial
B 3.10-48	Revision 31	04-06-2005
B 3.10-49	Revision 61	12-07-2010

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

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

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

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

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

(continued)

BASES (continued)

SAFETY LIMIT VIOLATIONS	Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 50.67, "Accident Source Term," limits (Ref. 7). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. EMF-2209(P)(A), SPCB Critical Power Correlation, (as identified in the COLR).
3. EMF-2245(P)(A), Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel, (as identified in the COLR).
4. ANP-10307PA Revision 0, AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, AREVA NP, June 2011.
5. ANP-10298PA Revision 0, ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, March 2010.
6. ANP-3140(P) Revision 0, Browns Ferry Units 1, 2, and 3 Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, August 2012.
7. 10 CFR 50.67.
8. ANP-3245P Revision 1, Browns Ferry Evaluation of PRFO Low Pressure Technical Specification Value, AREVA Inc., February 2014.

BASES

LCO 3.0.3 (continued)

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The LCO phrase, "Action shall be initiated within 1 hour ..." does not mean that a change in load must be commenced by the end of the 1 hour period. The action initiated at the end of the 1 hour period may be administrative in nature, such as preparing shutdown procedures. If corrective measures which would allow exiting LCO 3.0.3 are not complete at the end of 1 hour, but there is reasonable assurance that they will be completed with enough time remaining to allow for an orderly unit shutdown, if required, commencing a load decrease may be delayed until that time. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

(continued)

BASES

LCO 3.0.3 (continued)

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met.

b. A Condition exists for which the Required Actions have now been performed.

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	(Deleted by Tech Spec Bases Revision 106)

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

AREVA Fuel

For AREVA fuel, 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 ALPHGR (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 simple loop operation, a MAPLHGR multiplier is applied to the MAPLHGR limit (Ref. 11). The multiplier is documented in the COLR.

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

(continued)

BASES

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 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 operating with substantial margin to the APLHGR limits; thus this LCO is not required.

(continued)

BASES (continued)

REFERENCES

1. (Deleted by Tech Spec Revision 74)
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 5, January 2002.
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), Rev. 0, "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP Inc., as supplemented by the site specific approval in NRC safety evaluation, February 15, 2013 and July 31, 2014.
12. EMF-2292(P)(A), "ATRIUM™ -10: Appendix K Spray Heat Transfer Coefficients," (as identified in the COLR).

(continued)

BASES

REFERENCES (continued)

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

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

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

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND	MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during abnormal operational transients. Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion. The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.
APPLICABLE SAFETY ANALYSES	The analytical methods and assumptions used in evaluating the abnormal operational transients to establish the operating limit MCPR are presented in References 11, 12, 13, 14, and 15. To ensure 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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 (Ref. 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 stop valve closure and 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. Additionally 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.

(continued)

BASES (continued)

REFERENCES

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

(continued)

BASES (continued)

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.
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

LCO 3.3.11	The primary containment hydrogen monitoring instrumentation allows the operators to detect trends in hydrogen concentration to diagnose the course of beyond design basis accidents. High hydrogen concentration is measured, continuously recorded, and displayed in the control room by a single instrument channel. The analyzer has the capability for sampling both the drywell and the suppression chamber. LCO 3.3.11 requires the primary containment hydrogen analyzer to be OPERABLE.
APPLICABILITY	The primary containment hydrogen analyzer is required to be OPERABLE when primary containment is inerted, except as allowed by the relaxations during startup and shutdown addressed below. The primary containment must be inert in MODE 1, since this is the condition with the highest probability of an event that could produce hydrogen. Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and de-inerted as soon as possible in the plant shutdown. As long as reactor power is < 15% RTP, the potential for an event that generates significant hydrogen is low and the primary containment need not be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a startup, or within the last 24 hours before a shutdown, is low enough that these "windows," when the primary containment is not inerted, are also justified. The 24 hour time period is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.
ACTIONS	A.1 Seven days to restore the instrument is reasonable given the requirements to be available for use in diagnosing beyond design basis events.

Table B 3.3.3.2-1 (Page 2 of 4)
Backup Control System Instrumentation and Controls

FUNCTION	NUMBER REQUIRED
<u>Transfer/Control Parameter (continued)</u>	
14. RHRSW Pumps (0-43-23-1, -5, -8, -12) (0-HS-23-1C, -5C, -8C, -12C); EECW Pumps (north header) (0-43-23-1, -85, -8, -91) (0-HS-23-1C, -85C, -8C, -91C); EECW Pumps (south header) (0-43-23-15, -88, -23, -94) (0-HS-23-15C, -88C, -23C, -94C)	note e
15. RHRSW Discharge Valves for RHR Loop I Heat Exchangers (2-XS-23-34, -40) (2-HS-23-34C, -40C)	2, note f
16. RCW Pumps 1D and 3D (Trip Function Only) (1-XS-24-16, 3-XS-24-16) (1-HS-24-16C, 3-HS-24-16C)	2, note g
17. Recirculation System Sample Line Isolation Valves (2-XS-43-13, -14) (2-HS-43-13C, -14C)	1, note i
18. EECW Sectionalizing Valves (0-XS-67-13, -14, -17, -18, -21, -22, -25, -26) (0-HS-67-13C, -14C, -17C, -18C, -21C, -22C, -25C, -26C)	8, 1 per valve, note j
19. (Deleted by Revision 88)	
20. Recirculation Pump 2B Discharge Valve (2-XS-68-79) (2-HS-68-79C)	1
21. RWCU Drain to Main Condenser Hotwell Isolation Valve (2-XS-69-16) (2-HS-69-16C)	1
note e: There are 12 RHRSW pumps. All are equipped with emergency transfer switches. Backup Control must be available for 2 OPERABLE pumps aligned for EECW service (supports all units). Backup control for an additional 2 OPERABLE pumps must be available for RHRSW for RHR Loop I.	
note f: 1 Discharge Valve per RHR Loop I Heat Exchanger for a total of 2.	
note g: 1 per pump. Trip function necessary to prevent spurious start overloading 4-kV Buses/Diesel Generators.	
note h: Deleted.	
note i: 1 Recirculation System Sample Line Isolation Valve required, may be either inboard valve or outboard valve.	
Note j: Not required if valve breaker remains open, per NFPA 805 requirement, except when required for valve testing or operation.	

BASES

BACKGROUND (continued)

There are two motor breakers provided for each of the two recirculation pumps for a total of four breakers. The output of each trip system is provided to one of the two breakers for each recirculation pump.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The ATWS-RPT initiates an RPT to aid in preserving the integrity of the fuel cladding following events in which a scram does not, but should, occur. Based on its contribution to the reduction of overall plant risk, however, the instrumentation meets Criterion 4 of the NRC Policy Statement (Ref. 3).

The OPERABILITY of the ATWS-RPT is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have a required number of OPERABLE channels in each trip system, with their setpoints within the specified Allowable Value of SR 3.3.4.2.3. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint). ATWS-RPT Channel OPERABILITY also includes the associated recirculation pump motor breakers. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Allowable Values are specified for each ATWS-RPT Function specified in the LCO. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between 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 predetermined values of output at which an action should take place. The setpoints are compared to the actual process

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

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

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

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

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

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

5.g. SLC System Initiation

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 4). An isolation signal for both RWCU isolation valves is initiated when the SLC pump start handswitch is not in the stop position.

There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch.

The SLC System Initiation Function is required to be OPERABLE in MODES 1 and 2, because these are the only MODES where the reactor can be critical, and in MODE 3 because this MODE uses the SLC System sodium pentaborate as a buffering solution to maintain the pH level at or above 7 in the suppression pool in the event of a LOCA. These MODES are consistent with the Applicability for the SLC System (LCO 3.1.7).

As noted (footnote (a) to Table 3.3.6.1-1), the SLC initiation signal provides input to the isolation logic for both RWCU isolation valves.

5.h. Reactor Vessel Water Level - Low, Level 3 (LIS-3-203A-D)

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some interfaces with the reactor vessel occurs to isolate the potential sources of a break. The isolation of the RWCU System on Level 3 supports actions to ensure that the fuel peak cladding

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The pressure-temperature (P-T) curves included in the Technical Specifications have been developed to present steam-dome pressure versus minimum vessel metal temperature, incorporating appropriate non-beltline limits and irradiation embrittlement effects in the beltline. There are two sets of curves provided for Unit 2. The first set applies to operation up to and including 38 effective full power years (EFPY), and the second set applies to operation greater than 38 EFPY and less than or equal to 48 EFPY, where 48 EFPY represents the end of the renewed license and 38 EFPY is provided as a midpoint between the current EFPY and 48 EFPY. The P-T curves are provided in Figure 3.4.9-1 and Figure 3.4.9-2, respectively. Figure 3.4.9-1 contains P-T limit curves for mechanical heatup or cooldown following nuclear shutdown (bottom head and upper RPV/beltline) and for core operation (criticality). Figure 3.4.9-2 contains P-T limit curves for inservice leakage and hydrostatic testing (bottom head and upper RPV/beltline). The maximum rate of change of reactor coolant temperature is contained in SR 3.4.9.5, SR 3.4.9.6, and SR 3.4.9.7.

(continued)

BASES

BACKGROUND (continued)

The P-T curves incorporate a fluence calculated in accordance with GE Licensing Topical Report NEDC-32983P (Ref. 10), which has been approved by the NRC and is in compliance with Regulatory Guide 1.190 (Ref. 11). The fluence represents an Extended Power Uprate (EPU) for the rated power of 3952 MW_t, and is conservatively applied for the rated power of 3458 MW_t. The 1998 Edition of the ASME Section XI Boiler and Pressure Vessel Code including 2000 Addenda was used in accordance with 10 CFR 50.55a.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

(continued)

BASES

BACKGROUND (continued)

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, abnormal operational transients, and system hydrostatic tests. It mandates the use of the ASME Code, Section III, Appendix G (Ref. 2).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 3), Appendix H of 10 CFR 50 (Ref. 4), and BWRVIP-86-A, Revision 1 (Ref. 12). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 5.

Minimum reactor vessel temperature requirements for pressure-temperature (P/T) limits depend on the reactor vessel's controlling material (which is either the material in the closure flange or the material in the beltline region with the highest reference temperature) and on the reactor's operating condition (i.e., hydrostatic pressure and leak tests, or normal operation including anticipated operation occurrences), vessel pressure, whether or not fuel is in the vessel, and whether the core is critical. The beltline region of the reactor vessel is the region that directly surrounds the effective height of the active core and adjacent regions that are predicted to experience sufficient radiation exposure to be considered in the selection of the most limiting material with regard to radiation exposure. The metal temperature of the controlling material, in the region of the controlling material which has the least favorable combination of stress and temperature, must exceed the appropriate minimum temperature requirements for the reactor vessel condition and pressure specified in 10 CFR 50, Appendix G.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1 (continued)

This SR has been modified by three Notes. Note 1 requires this Surveillance to be performed only during system heatup and cooldown operations or inservice leakage and hydrostatic testing. Also, Note 1 only requires this SR to be performed during inservice leakage and hydrostatic testing when reactor pressure is > 313 psig. Note 2 allows the limits of Figure 3.4.9-2 to be applied during nonnuclear heatup and ambient loss cooldown associated with inservice leak and hydrostatic testing provided that the heatup and cooldown rates are ≤ 15°F/hr. Note 3 provides that the limits of Figures 3.4.9-1 and 3.4.9-2 do not apply when the tension from the reactor head flange bolting studs is removed.

SR 3.4.9.2

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before withdrawing control rods that will make the reactor critical.

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the control rod withdrawal.

(continued)

BASES

REFERENCES

12. "BWRVIP-86, Revision 1-A: BWR Vessel and Internals project. Updated BWR Integrated Surveillance Program (ISP) Implementation Plan," EPRI Technical Report 1016575, May 2013.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The PCIVs LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory, and establishing the primary containment boundary during major accidents. As part of the primary containment boundary, PCIV OPERABILITY supports leak tightness of primary containment. Therefore, the safety analysis of any event requiring isolation of primary containment is applicable to this LCO.

The DBAs that result in a release of radioactive material and are mitigated by PCIVs are a LOCA and a main steam line break (MSLB). In the analysis for each of these accidents, it is assumed that PCIVs are either closed or close within the required isolation times following event initiation. This ensures that potential paths to the environment through PCIVs (including primary containment purge valves) are minimized. Of the events analyzed in Reference 1, the LOCA is the most limiting event due to radiological consequences.

The closure time of the main steam isolation valves (MSIVs) is a significant variable from a radiological standpoint. The MSIVs are required to close within 3 to 5 seconds since the 5 second closure time is consistent with or conservative to the times assumed in the analyses. The safety analyses assume that the purge valves were closed at event initiation. Likewise, it is assumed that the primary containment is isolated such that release of fission products to the environment is controlled.

The DBA analysis assumes that primary containment is isolated within a certain time period (based on PCIV closure times provided in Reference 2 and inboard MSIV closure times of 2 minutes for a LOCA per Reference 9) and that leakage is terminated, except for the maximum allowable leakage rate, L_a. The primary containment isolation total response time includes signal delay, diesel generator startup (for loss of offsite power), and PCIV stroke times.

(continued)

BASES

REFERENCES

1. FSAR, Section 14.6.
2. BFN Technical Instruction (TI), 0-TI-360.
3. 10 CFR 50, Appendix J, Option B.
4. FSAR, Section 5.2.
5. FSAR, Section 14.6.5.
6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
7. FSAR Table 5.2-2.
8. General Electric NEDO-32977-A (Boiling Water Reactor Owners Group Topical Report, B21-00658-01), "Excess Flow Check Valve Testing Relaxation", dated June 2000.
9. MDQ0000012016000566, Revision 0, "Main Steam Isolation Valve (MSIV) Loss of Coolant Accident (LOCA) Closure Analysis," dated September 2016.

BASES

APPLICABLE SAFETY ANALYSES (continued)	With two and three units fueled, a worst case single failure could also include the loss of two RHRSW pumps caused by losing a 4 kV shutdown board since there are certain alignment configurations that allow two RHRSW pumps to be powered from the same 4 kV shutdown board. As discussed in the FSAR, Section 14.6.3.3.2 (Ref. 4) for these analyses, manual initiation of the OPERABLE RHRSW subsystems and the associated RHR System is assumed to occur 10 minutes after a DBA. The analyses assume that there are two RHRSW subsystems operating in each unit, with one RHRSW pump in each subsystem capable of producing 4000 gpm of flow. In this case, the maximum suppression chamber water temperature and pressure are 177°F (as reported in Reference 3) and 36.3 psig, respectively, well below the design temperature of 281°F and maximum allowable pressure of 62 psig. The RHRSW System together with the UHS satisfies Criterion 3 of the NRC Policy Statement (Ref 5).
LCO	Four RHRSW subsystems are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst case single active failure occurs coincident with the loss of offsite power. Additionally, since the RHRSW pumps are shared between the three BFN units, the number of OPERABLE pumps required is also dependent on the number of units fueled. An RHRSW subsystem is considered OPERABLE when: a. At least one RHRSW pump (i.e., one required RHRSW pump) is OPERABLE; and b. An OPERABLE flow path is capable of taking suction from the intake structure and transferring the water to the associated RHR heat exchanger at the assumed flow rate.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Turbine Bypass System

BASES

BACKGROUND

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is 25% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of nine valves connected to the main steam lines between the main steam isolation valves and the turbine stop valve bypass valve chest. Each of these nine valves is operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Pressure Regulator and Turbine Generator Control System, as discussed in the FSAR, Section 7.11 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows from the bypass chest, through connecting piping, to the pressure breakdown assemblies, where a series of orifices are used to further reduce the steam pressure before the steam enters the condenser.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.3

This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analysis. The response time limits are specified in the cycle specific transient analyses performed to support the preparation of FSAR, Appendix N, Supplemental Reload Licensing Report (Ref. 4). The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown the 24 month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 7.11.
2. FSAR, Section 14.5.1.1 and 14.5.1.2.
3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
4. FSAR, Appendix N.

BASES

BACKGROUND (continued)

switchyards from the transmission network. One of the two required qualified offsite circuits must be from the 500 kV transmission network to meet the requirements of 10 CFR 50, Appendix A, GDC 17 (Ref. 1). Four basic circuits from the transmission network to the safety related Division I (A and B 4.16 kV shutdown boards) and Division II (C and D 4.16 kV shutdown boards), are as follows:

1. From the 500 kV switchyard, through unit station service transformer (USST) 1B to a 4.16 kV unit board. That unit board feeds 4.16 kV shutdown bus 1 or 2, which then feeds two of the Unit 1 and 2 4.16 kV shutdown boards (A and B or C and D);

2. From the 500 kV switchyard, through USST 2B to a 4.16 kV unit board. That unit board feeds 4.16 kV shutdown bus 1 or 2, which then feeds two of the Unit 1 and 2 4.16 kV shutdown boards (A and B or C and D);

3. From the 161 kV transmission network, through common station service transformer (CSST) A to start bus 1A or 1B, then to a 4.16 kV unit board. That unit board feeds 4.16 kV shutdown bus 1 or 2, which then feeds two of the Unit 1 and 2 4.16 kV shutdown boards (A and B or C and D); and

4. From the 161 kV transmission network, through CSST B to start bus 1A or 1B, and then to a 4.16 kV unit board. That unit board feeds 4.16 kV shutdown bus 1 or 2, which then feeds two of the Unit 1 and 2 4.16 kV shutdown boards (A and B or C and D).

(continued)

BASES

LCO (continued)

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the A and B (Division I) or C and D (Division II) 4.16 kV shutdown boards. One of the two required qualified offsite circuits must be from the 500 kV transmission network to meet the requirements of 10 CFR 50, Appendix A, GDC 17 (Ref. 1). Specific circuits are described below:

1. From the 500 kV switchyard through unit station service transformer (USST) 1B to 4.16 kV unit board 1A, to 4.16 kV shutdown bus 1, to 4.16 kV shutdown boards A and B; and/or, to 4.16 kV unit board 1B, to 4.16 kV shutdown bus 2, to 4.16 kV shutdown boards C and D.

2. From the 500 kV switchyard through USST 2B to 4.16 kV unit board 2A, to 4.16 kV shutdown bus 2, to 4.16 kV shutdown boards C and D; and/or, to 4.16 kV unit board 2B, to 4.16 kV shutdown bus 1, to 4.16 kV shutdown boards A and B.

3. From the 161 kV transmission network, through common station service transformer (CSST) A to start bus 1A or 1B, to 4.16 kV unit board 1A or 2B, to 4.16 kV shutdown bus 1, to 4.16 kV shutdown boards A and B; or alternately, to 4.16 kV unit board 1B or 2A, to 4.16 kV shutdown bus 2, to 4.16 kV shutdown boards C and D.

4. From the 161 kV transmission network, through CSST B to start bus 1A or 1B, to 4.16 kV unit board 1A or 2B, to 4.16 kV shutdown bus 1, to 4.16 kV shutdown boards A and B; or alternately, to 4.16 kV unit board 1B or 2A, to 4.16 kV shutdown bus 2, to 4.16 kV shutdown boards C and D.

(continued)

BASES

ACTIONS (continued)

J.1

Condition J corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

K.1

Required Action K.1 is intended to provide assurance that a loss of offsite power, during the period that a required Unit 3 DG is inoperable, does not result in a complete loss of safety function of critical systems (i.e., SGT, CREVS, RHRSW, or EECW). These features are designed with redundant safety related divisions (i.e., single division systems are not included). Redundant required features failures consist of inoperable features associated with a division redundant to the division that has an inoperable Unit 3 DG.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action the Completion Time only begins on discovery that both:

a. An inoperable required Unit 3 DG exists; and

b. A redundant required feature supported by another DG, is inoperable.

(continued)

BASES

ACTIONS

K.1 (continued)

If, at any time during the existence of this Condition (a required Unit 3 DG inoperable), a required redundant feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering a required Unit 3 DG inoperable coincident with an inoperable required redundant feature that is associated with the OPERABLE DGs results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period.

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

BASES

BACKGROUND

Each diesel generator (DG) is provided with three interconnected storage tanks having a minimum usable fuel oil volume (35,280 gallons) sufficient to operate that DG for a period of 7 days while the DG is supplying maximum post loss of coolant accident (LOCA) load demand discussed in FSAR, Section 8.5.3.4 (Ref. 1) and Regulatory Guide 1.137 (Ref. 2). A transfer pump is located at the fuel oil storage tanks which can supply fuel oil from two 71,000-gallon fuel oil storage tanks to the 7-day storage tanks. Only the 7-day storage tanks are subject to this LCO, the Actions, and the Surveillance Requirements. In addition, it is possible to transfer fuel from one 7-day storage tank to any other by using transfer pumps. This onsite fuel oil capacity is sufficient to operate the DGs for longer than the time to replenish the onsite supply from outside sources.

Fuel oil is transferred from the 7-day storage tank to the day tank by either of two transfer pumps associated with each diesel generator. This is accomplished automatically by level switches on the day tank. Redundancy of pumps and piping precludes the failure of one pump, or the rupture of any pipe, valve, or tank to result in the loss of more than one DG. All 7-day tanks are embedded in the substructure of the Standby Diesel Generator Building.

For proper operation of the standby DGs, it is necessary to ensure the proper quality of the stored fuel oil. The fuel oil property monitored is the total particulate concentration. Periodic testing of the stored fuel oil total particulate concentration is a method to monitor the potential degradation related to long term storage and the potential impact to fuel filter plugging as a result of high particulate levels.

(continued)

BASES (continued)

APPLICABILITY

The AC sources (LCO 3.8.1 and LCO 3.8.2) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an abnormal operational transient or a postulated DBA. Because stored diesel fuel oil, lube oil, and starting air subsystem support LCO 3.8.1 and LCO 3.8.2, stored diesel fuel oil, lube oil, and starting air are required to be within limits when the associated DG is required to be OPERABLE.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each DG. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DG subsystem. Complying with the Required Actions for one inoperable DG subsystem may allow for continued operation, and subsequent inoperable DG subsystem(s) governed by separate Condition entry and application of associated Required Actions.

A.1

In this condition, the 7-day fuel oil supply for a DG is not available. However, the Condition is restricted to fuel oil level reductions that maintain at least a 6-day supply. The fuel oil level equivalent to a 6-day supply is 30,240 gallons. These circumstances may be caused by events such as:

a. Full load operation required for an inadvertent start while at minimum required level; or

b. Feed and bleed operations that may be necessitated by increasing particulate levels or any number of other oil quality degradations.

(continued)

BASES

ACTIONS

A.1 (continued)

This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of the fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required level prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

B.1

In this Condition, the 7-day lube oil inventory, i.e., sufficient lube oil to support 7 days of continuous DG operation at full load conditions is not available. However, the Condition is restricted to lube oil volume reductions that maintain at least a 6-day supply. The lube oil inventory equivalent to a 6-day supply is 150 gallons. This restriction allows sufficient time for obtaining the requisite replacement volume. A period of 48 hours is considered sufficient to complete restoration of the required volume prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the low rate of usage, the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

C.1

This Condition is entered as a result of a failure to meet the acceptance criterion for particulates. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence

(continued)

BASES

ACTIONS

C.1 (continued)

of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, since particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and since proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion Time allows for further evaluation, re-sampling, and re-analysis of the DG fuel oil.

D.1

With the new fuel oil properties in the Bases for SR 3.8.3.3 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or a combination of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is high likelihood that the DG would still be capable of performing its intended function.

E.1

Only one of the two redundant air starting systems is required to support associated DG operability. With the starting air receiver pressure < 165 psig in the required starting air system, sufficient capacity to start the associated DG may not exist. The associated DG may be incapable of performing its intended function and must be immediately declared inoperable. This declaration also requires entry into applicable Conditions and Required Actions for an inoperable DG, LCO 3.8.1, "AC Sources - Operating."

(continued)

BASES

ACTIONS

F.1

With a Required Action and associated Completion Time not met, or the stored diesel fuel oil, lube oil, or starting air subsystem inoperable for reasons other than addressed by Conditions A through E, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support each DG's operation for 7 days at full load. The fuel oil level equivalent to a 7-day supply is 35,280 gallons when calculated in accordance with References 2 and 6. The required fuel storage volume is determined using the most limiting energy content of the stored fuel. Using the known correlation of diesel fuel oil absolute specific gravity or API gravity to energy content, the required diesel generator output, and the corresponding fuel consumption rate, the on site fuel storage volume required for 7 days of operation can be determined. SR 3.8.3.3 requires new fuel to be tested to verify that the absolute specific gravity or API gravity is within the range assumed in the diesel fuel oil consumption calculations. The 7-day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

The 31 day Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

SR 3.8.3.2

This Surveillance ensures that sufficient lubricating oil inventory is available to support at least 7 days of full load operation for each DG. The lube oil inventory equivalent to a 7-day supply is 175 gallons and is based on the DG manufacturer's consumption values for the run time of the DG.

A 31 day Frequency is adequate to ensure that a sufficient lube oil supply is onsite, since DG starts and run time are closely monitored by the plant staff.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.3

The tests listed below are a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate detrimental impact on diesel engine combustion. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. These tests are to be conducted prior to adding the new fuel to the 7-day storage tank(s), but in no case is the time between receipt of new fuel and conducting the tests to exceed 31 days following addition to the 7-day storage tanks. The tests, limits, and applicable ASTM Standards are as follows:

a. Sample the new fuel oil in accordance with ASTM D4057-12 (Ref. 7)

b. Verify in accordance with the tests specified in ASTM D975-14a (Ref. 7) that the sample has an absolute specific gravity at 60/60°F of ≥ 0.83 and ≤ 0.89 or an API gravity at 60°F of ≥ 27° and ≤ 39° when tested in accordance with ASTM D1298-12b (Ref. 7), a kinematic viscosity at 40°C of ≥ 1.9 centistokes and ≤ 4.1 centistokes, and a flash point of ≥ 125°F

c. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176-04 or a water and sediment content within limits when tested in accordance with ASTM D2709-96 (Ref. 7)

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO because the fuel oil is not added to the 7-day storage tank.

Either prior to adding new fuel oil to the 7-day storage tanks or within 31 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified in Table 1 of ASTM D975-14a (Ref. 7) are met for new fuel oil when tested in accordance with ASTM D975-14a (Ref. 7).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.3

except that the analysis for sulfur may be performed in accordance with ASTM D4294-10 (Ref. 7). If these tests are not completed prior to adding new fuel oil to the 7-day storage tanks, the 31 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on DG operation. This Surveillance ensures the availability of high quality fuel oil for the DGs.

Fuel oil degradation during long term storage shows up as an increase in particulates, mostly due to oxidation. The presence of particulates does not mean that the fuel oil will not burn properly in a diesel engine. The particulates can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

Particulate concentrations should be determined in accordance with ASTM D6217-11 (Ref. 7). This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing. Because the 7-day tank consists of three interconnected tanks, samples are drawn from each tank.

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

SR 3.8.3.4

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The system design requirements provide for at least one start cycle from one of two redundant air start systems without recharging. A start cycle is defined by the DG vendor, but usually is measured in terms of time (seconds of cranking) or engine cranking speed. The pressure specified in this SR is the lowest pressure at which at least one start attempt can be accomplished using one of two redundant air start systems.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.4

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.

SR 3.8.3.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel storage tanks once every 31 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 2). This SR is for preventive maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during performance of the Surveillance.

REFERENCES

1. FSAR, Section 8.5.3.4.
2. Regulatory Guide 1.137, Revision 1, October 1979.
3. FSAR, Chapter 6.
4. FSAR, Chapter 14.
5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

REFERENCES

6. ANSI N195-1976, " Fuel Oil Systems for Standby Diesel Generators," Section 5.4, April 1976.
7. ASTM Standards: D4057-12; D4176-04; D2709-96; D1298-12b; D975-14a; D4294-10; D6217-11.

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B 3.6-27	0	Initial
B 3.6-28	0	Initial
B 3.6-29	0	Initial
B 3.6-30	0	Initial
B 3.6-31	0	Initial
B 3.6-32	0	Initial
B 3.6-33	Revision 29	01-25-2005
B 3.6-34	Revision 84	08-29-2014

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B 3.6-34a	228	01-29-2001
B 3.6-35	Revision 62	01-12-2012
B 3.6-36	Revision 105	09-01-2016
B 3.6-37	214	9-08-98
B 3.6-38	0	Initial
B 3.6-39	0	Initial
B 3.6-40	0	Initial
B 3.6-41	0	Initial
B 3.6-42	0	Initial
B 3.6-43	0	Initial
B 3.6-44	0	Initial
B 3.6-45	0	Initial
B 3.6-46	0	Initial
B 3.6-47	0	Initial
B 3.6-48	215	11-30-98
B 3.6-49	0	Initial
B 3.6-50	0	Initial
B 3.6-51	0	Initial
B 3.6-52	0	Initial
B 3.6-53	0	Initial
B 3.6-54	0	Initial
B 3.6-55	0	Initial
B 3.6-56	215	11-30-98
B 3.6-57	0	Initial
B 3.6-58	Revision 85	09-30-2014
B 3.6-59	0	Initial
B 3.6-60	0	Initial
B 3.6-61	0	Initial
B 3.6-62	0	Initial
B 3.6-63	Revision 85	09-30-2014
B 3.6-64	0	Initial
B 3.6-65	0	Initial
B 3.6-66	0	Initial
B 3.6-67	0	Initial
B 3.6-68	0	Initial
B 3.6-69	0	Initial
B 3.6-70	0	Initial
B 3.6-71	Amendment 230	06-08-2001
B 3.6-72	Amendment 230	06-08-2001

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B 3.6-73	Revision 81	10-16-2013
B 3.6-74	0	Initial
B 3.6-75	0	Initial
B 3.6-76	0	Initial
B 3.6-77	0	Initial
B 3.6-78	0	Initial
B 3.6-79	0	Initial
B 3.6-80	0	Initial
B 3.6-81	0	Initial
B 3.6-82	0	Initial
B 3.6-83	0	Initial
B 3.6-84	0	Initial
B 3.6-85	Revision 3	03-19-99
B 3.6-86	0	Initial
B 3.6-87	0	Initial
B 3.6-88	0	Initial
B 3.6-89	0	Initial
B 3.6-90	0	Initial
B 3.6-91	0	Initial
B 3.6-92	0	Initial
B 3.6-93	Amendment 244	12-01-2003
B 3.6-94	Amendment 225	05-24-2000
B 3.6-95	Amendment 225	05-24-2000
B 3.6-96	Amendment 225	05-24-2000
B 3.6-97	0	Initial
B 3.6-98	0	Initial
B 3.6-99	0	Initial
B 3.6-100	0	Initial
B 3.6-101	0	Initial
B 3.6-102	Revision 29	01-25-2005
B 3.6-103	Revision 29	01-25-2005
B 3.6-104	Revision 29	01-25-2005
B 3.6-105	Revision 29	01-25-2005
B 3.6-106	Revision 29	01-25-2005
B 3.6-107	Revision 29	01-25-2005
B 3.6-108	Revision 29	01-25-2005
B 3.6-109	Revision 29	01-25-2005
B 3.6-110	0	Initial
B 3.6-111	0	Initial

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B 3.6-112	Revision 29	01-25-2005
B 3.6-113	Revision 29	01-25-2005
B 3.6-114	0	Initial
B 3.6-115	Revision 29	01-25-2005
B 3.6-116	Revision 29	01-25-2005
B 3.6-117	0	Initial
B 3.6-118	Revision 29	01-25-2005
B 3.6-119	Revision 29	01-25-2005
B 3.6-120	Revision 108	10-27-2017
B 3.6-121	215	11-30-98
B 3.7-1	Revision 73	01-03-2013
B 3.7-2	Revision 73	01-03-2013
B 3.7-3	Revision 102	09-01-2016
B 3.7-4	Revision 73	01-03-2013
B 3.7-4a	Revision 73	01-03-2013
B 3.7-5	Revision 73	01-03-2013
B 3.7-5a	Revision 73	01-03-2013
B 3.7-5b	Revision 73	01-03-2013
B 3.7-6	Revision 73	01-03-2013
B 3.7-6a	Revision 73	01-03-2013
B 3.7-7	Revision 73	01-03-2013
B 3.7-8	Revision 73	01-03-2013
B 3.7-9	Revision 73	01-03-2013
B 3.7-10	Revision 69	10-05-2012
B 3.7-11	0	Initial
B 3.7-12	0	Initial
B 3.7-13	214	09-08-98
B 3.7-14	Revision 69	10-05-2012
B 3.7-15	0	Initial
B 3.7-16	215	11-30-98
B 3.7-17	Revision 90	02-26-2015
B 3.7-17a	261	10-16-2009
B 3.7-18	Revision 90	02-26-2015
B 3.7-19	Revision 67	08-10-2012
B 3.7-19a	Revision 67	08-10-2012
B 3.7-20	Revision 67	08-10-2012
B 3.7-21	261	10-16-2009
B 3.7-21a	Revision 67	08-10-2012
B 3.7-22	Revision 67	08-10-2012

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B 3.7-23	Revision 67	08-10-2012
B 3.7-24	Revision 108	10-27-2017
B 3.7-25	Revision 90	02-26-2015
B 3.7-25a	Revision 90	02-26-2015
B 3.7-26	0	Initial
B 3.7-27	0	Initial
B 3.7-28	0	Initial
B 3.7-29	0	Initial
B 3.7-30	0	Initial
B 3.7-31	215	11-30-98
B 3.7-32	Revision 103	10-15-2016
B 3.7-33	Revision 61	12-07-2010
B 3.7-34	Revision 25	03-12-2004
B 3.7-35	215	11-30-98
B 3.7-36	Revision 103	10-15-2016
B 3.7-37	Revision 29	01-25-2005
B 3.7-38	0	Initial
B 3.7-39	Revision 29	01-25-2005
B 3.8-1	Revision 100	06-09-2016
B 3.8-2	Revision 52	05-11-2007
B 3.8-2a	Revision 42	11-16-2006
B 3.8-3	0	Initial
B 3.8-4	Revision 57	04-10-2008
B 3.8-5	Amendment 266	10-05-2011
B 3.8-5a	Amendment 266	10-05-2011
B 3.8-6	Revision 52	05-11-2007
B 3.8-7	Revision 100	06-09-2016
B 3.8-8	Revision 52	05-11-2007
B 3.8-9	Revision 52	05-11-2007
B 3.8-10	0	Initial
B 3.8-11	Amendment 244	12-01-2003
B 3.8-12	0	Initial
B 3.8-13	0	Initial
B 3.8-14	Amendment 266	10-05-2011
B 3.8-15	Amendment 266	10-05-2011
B 3.8-15a	Amendment 266	10-05-2011
B 3.8-16	Amendment 266	10-05-2011
B 3.8-17	Amendment 266	10-05-2011
B 3.8-18	Revision 72	11-03-2011

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B 3.8-18a	Amendment 266	10-05-2011
B 3.8-19	Amendment 266	10-05-2011
B 3.8-20	Revision 7	09-17-99
B 3.8-21	0	Initial
B 3.8-22	0	Initial
B 3.8-23	0	Initial
B 3.8-24	0	Initial
B 3.8-25	Revision 101	08-18-2016
B 3.8-26	Revision 101	08-18-2016
B 3.8-27	Revision 28	08-26-2004
B 3.8-28	Revision 28	08-26-2004
B 3.8-29	0	Initial
B 3.8-30	0	Initial
B 3.8-31	0	Initial
B 3.8-32	0	Initial
B 3.8-33	215	11-30-98
B 3.8-34	215	11-30-98
B 3.8-35	215	11-30-98
B 3.8-36	0	Initial
B 3.8-37	215	11-30-98
B 3.8-38	0	Initial
B 3.8-39	0	Initial
B 3.8-40	0	Initial
B 3.8-41	Revision 52	05-11-2007
B 3.8-42	Revision 52	05-11-2007
B 3.8-43	Revision 52	05-11-2007
B 3.8-44	0	Initial
B 3.8-45	0	Initial
B 3.8-46	0	Initial
B 3.8-47	0	Initial
B 3.8-48	0	Initial
B 3.8-49	Revision 95	01-21-2016
B 3.8-50	0	Initial
B 3.8-51	0	Initial
B 3.8-52	Revision 95	01-21-2016
B 3.8-53	Revision 95	01-21-2016
B 3.8-54	Revision 95	01-21-2016
B 3.8-54a	Revision 95	01-21-2016
B 3.8-55	Revision 95	01-21-2016

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B 3.8-56	Revision 95	01-21-2016
B 3.8-56a	Revision 95	01-21-2016
B 3.8-56b	Revision 95	01-21-2016
B 3.8-56c	Revision 95	01-21-2016
B 3.8-57	0	Initial
B 3.8-58	0	Initial
B 3.8-59	0	Initial
B 3.8-60	0	Initial
B 3.8-61	0	Initial
B 3.8-62	0	Initial
B 3.8-63	0	Initial
B 3.8-64	0	Initial
B 3.8-65	215	11-30-98
B 3.8-66	Revision 58	10-01-2008
B 3.8-67	0	Initial
B 3.8-68	215	11-30-98
B 3.8-69	0	Initial
B 3.8-70	0	Initial
B 3.8-71	0	Initial
B 3.8-72	0	Initial
B 3.8-73	0	Initial
B 3.8-74	0	Initial
B 3.8-75	0	Initial
B 3.8-76	0	Initial
B 3.8-77	0	Initial
B 3.8-78	0	Initial
B 3.8-79	0	Initial
B 3.8-80	0	Initial
B 3.8-81	0	Initial
B 3.8-82	0	Initial
B 3.8-83	0	Initial
B 3.8-84	0	Initial
B 3.8-85	0	Initial
B 3.8-86	Revision 83	02-28-2014
B 3.8-87	Revision 93	10-23-2015
B 3.8-88	Revision 93	10-23-2015
B 3.8-89	0	Initial
B 3.8-90	Revision 36	06-22-2006
B 3.8-91	0	Initial

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B 3.8-92	0	Initial
B 3.8-93	0	Initial
B 3.8-94	Revision 83	02-28-2014
B 3.8-95	0	Initial
B 3.8-96	0	Initial
B 3.8-97	0	Initial
B 3.8-98	0	Initial
B 3.8-99	0	Initial
B 3.8-100	0	Initial
B 3.8-101	0	Initial
B 3.8-102	0	Initial
B 3.8-103	0	Initial
B 3.8-104	0	Initial
B 3.8-105	0	Initial
B 3.8-106	Revision 83	02-28-2014
B 3.8-107	0	Initial
B 3.8-108	0	Initial
B 3.8-109	0	Initial
B 3.9-1	0	Initial
B 3.9-2	0	Initial
B 3.9-3	0	Initial
B 3.9-4	Amendment No. 232	03-06-2002
B 3.9-5	Amendment No. 232 and Revision 16	03-06-2002 and 03-21-2002
B 3.9-5a	Revision 60	10-16-2009
B 3.9-6	0	Initial
B 3.9-7	0	Initial
B 3.9-8	0	Initial
B 3.9-9	0	Initial
B 3.9-10	Revision 60	10-16-2009
B 3.9-11	0	Initial
B 3.9-12	0	Initial
B 3.9-13	Revision 60	10-16-2009
B 3.9-14	0	Initial
B 3.9-15	0	Initial
B 3.9-16	0	Initial
B 3.9-17	0	Initial
B 3.9-18	Revision 60	10-16-2009
B 3.9-19	0	Initial

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B 3.9-20	0	Initial
B 3.9-21	0	Initial
B 3.9-22	Revision 60	10-16-2009
B 3.9-23	Revision 29	01-25-2005
B 3.9-24	0	Initial
B 3.9-25	Revision 29	01-25-2005
B 3.9-26	0	Initial
B 3.9-27	0	Initial
B 3.9-28	0	Initial
B 3.9-29	0	Initial
B 3.9-30	0	Initial
B 3.9-31	0	Initial
B 3.9-32	0	Initial
B 3.9-33	0	Initial
B 3.9-34	0	Initial
B 3.9-35	0	Initial
B 3.9-36	Amendment 249	09-27-2004
B 3.9-37	Amendment 249	09-27-2004
B 3.9-38	Amendment 249	09-27-2004
B 3.10-1	Revision 51	05-10-2007
B 3.10-2	Revision 51	05-10-2007
B 3.10-3	Revision 51	05-10-2007
B 3.10-4	Revision 51	05-10-2007
B 3.10-4a	Revision 51	05-10-2007
B 3.10-5	0	Initial
B 3.10-6	0	Initial
B 3.10-7	0	Initial
B 3.10-8	0	Initial
B 3.10-9	0	Initial
B 3.10-10	0	Initial
B 3.10-11	0	Initial
B 3.10-12	0	Initial
B 3.10-13	0	Initial
B 3.10-14	0	Initial
B 3.10-15	0	Initial
B 3.10-16	0	Initial
B 3.10-17	0	Initial
B 3.10-18	0	Initial
B 3.10-19	0	Initial

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B 3.10-20	0	Initial
B 3.10-21	Revision 4	04-09-99
B 3.10-22	0	Initial
B 3.10-23	0	Initial
B 3.10-24	0	Initial
B 3.10-25	0	Initial
B 3.10-26	0	Initial
B 3.10-27	Revision 4	04-09-99
B 3.10-28	0	Initial
B 3.10-29	0	Initial
B 3.10-30	0	Initial
B 3.10-31	0	Initial
B 3.10-32	0	Initial
B 3.10-33	0	Initial
B 3.10-34	0	Initial
B 3.10-35	0	Initial
B 3.10-36	Revision 61	12-07-2010
B 3.10-37	Revision 25	03-12-2004
B 3.10-38	Revision 61	12-07-2010
B 3.10-39	0	Initial
B 3.10-40	Revision 61	12-07-2010
B 3.10-41	0	Initial
B 3.10-42	Revision 61	12-07-2010
B 3.10-43	213	09-03-98
B 3.10-44	0	Initial
B 3.10-45	0	Initial
B 3.10-46	0	Initial
B 3.10-47	213	09-03-98
B 3.10-48	0	Initial
B 3.10-49	Revision 61	12-07-2010

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity

Critical power correlations are valid over a wide range of conditions per References 2 and 5, extending to expected conditions below 25% THERMAL POWER. For core thermal power levels at, or above 25% rated, the hot channel flow rate is expected to be >28,000 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 critical power correlations. For operation at low pressure/flow conditions, consistent with the low power region of the Power/Flow operating map, another basis is used as follows:

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

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

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

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

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model combining all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved AREVA critical power correlations. One specific uncertainty included in the SL is the uncertainty inherent in the critical power correlation. References 2, 3, 4, 5 and 6 describe the uncertainties and methodologies used in determining the MCPR SL.

(continued)

BASES

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, 4, 5 and 6 describe the uncertainties and methodologies used in determining the MCPR SL.

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BASES (continued)

SAFETY LIMIT VIOLATIONS	Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 50.67, "Accident Source Term," limits (Ref. 5). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. EMF-2209(P)(A), SPCB Critical Power Correlation, (as identified in the COLR).
3. EMF-2245(P)(A), Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel, (as identified in the COLR).
4. ANP-10307PA Revision 0, AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, AREVA NP, June 2011.
5. ANP-10298PA Revision 0, ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, March 2010.
6. ANP-3140(P) Revision 0, Browns Ferry Units 1, 2, and 3 Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, August 2012.
7. 10 CFR 50.67.
8. ANP-3245P Revision 1, Browns Ferry Evaluation of PRFO Low Pressure Technical Specification Value, AREVA Inc., February 2014.

BASES

LCO 3.0.3 (continued)

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The LCO phrase, "Action shall be initiated within 1 hour ..." does not mean that a change in load must be commenced by the end of the 1 hour period. The action initiated at the end of the 1 hour period may be administrative in nature, such as preparing shutdown procedures. If corrective measures which would allow exiting LCO 3.0.3 are not complete at the end of 1 hour, but there is reasonable assurance that they will be completed with enough time remaining to allow for an orderly unit shutdown, if required, commencing a load decrease may be delayed until that time. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

(continued)

BASES

LCO 3.0.3 (continued)

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met.

b. A Condition exists for which the Required Actions have now been performed.

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

(Deleted by Tech Spec Bases Revision 106)

(continued)

APPLICABLE SAFETY ANALYSES (continued)

AREVA Fuel

For AREVA fuel, 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 ALPHGR (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

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 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 operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

(continued)

BASES (continued)

REFERENCES

1. (Deleted by Tech Spec Bases Revision 74)
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 5, January 2002.
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), Rev. 0, "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP Inc., as supplemented by the site specific approval in NRC safety evaluation, February 15, 2013 and July 31, 2014.
12. EMF-2292(P)(A), "ATRIUM™ -10: Appendix K Spray Heat Transfer Coefficients," (as identified in the COLR).

(continued)

BASES

REFERENCES (continued)

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

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

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

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

BASES

REFERENCES (continued)

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

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. 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).

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

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during abnormal operational transients. Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the abnormal operational transients to establish the operating limit MCPR are presented in References 11, 12, 13, 14, and 15. To ensure 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

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 (Ref. 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 stop valve closure and 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. Additionally 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.

(continued)

BASES (continued)

REFERENCES

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

2. (Deleted).

3. FSAR, Chapter 3.

4. FSAR, Chapter 14.

5. FSAR, Appendix N.

6. (Deleted)

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

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

9. (Deleted)

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

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

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

(continued)

BASES (continued)

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.
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.

(continued)

BASES

LCO 3.3.11	The primary containment hydrogen monitoring instrumentation allows the operators to detect trends in hydrogen concentration to diagnose the course of beyond design basis accidents. High hydrogen concentration is measured, continuously recorded, and displayed in the control room by a single instrument channel. The analyzer has the capability for sampling both the drywell and the suppression chamber. LCO 3.3.11 requires the primary containment hydrogen analyzer to be OPERABLE.
APPLICABILITY	The primary containment hydrogen analyzer is required to be OPERABLE when primary containment is inerted, except as allowed by the relaxations during startup and shutdown addressed below. The primary containment must be inert in MODE 1, since this is the condition with the highest probability of an event that could produce hydrogen. Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and de-inerted as soon as possible in the plant shutdown. As long as reactor power is < 15% RTP, the potential for an event that generates significant hydrogen is low and the primary containment need not be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a startup, or within the last 24 hours before a shutdown, is low enough that these "windows," when the primary containment is not inerted, are also justified. The 24 hour time period is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.
ACTIONS	<u>A.1</u> Seven days to restore the instrument is reasonable given the requirements to be available for use in diagnosing beyond design basis events.

Table B 3.3.3.2-1 (Page 2 of 4)
Backup Control System Instrumentation and Controls

FUNCTION	NUMBER REQUIRED
Transfer/Control Parameter (continued)	
14. RHRSW Pumps (0-43-23-1, -5, -8, -12) (0-HS-23-1C, -5C, -8C, -12C) EECW Pumps (north header) (0-43-23-1, -85, -8, -91) (0-HS-23-1C, -85C, -8C, -91C) EECW Pumps (south header) (0-43-23-15, -88, -23, -94) (0-HS-23-15C, -88C, -23C, -94C)	note e
15. RHRSW Discharge Valves for RHR Loop I Heat Exchangers (3-XS-23-34, -40) (3-HS-23-34C, -40C)	2, note f
16. RCW Pumps 1D and 3D (Trip Function Only) (1-XS-24-16, 3-XS-24-16) (1-HS-24-16C, 3-HS-24-16C)	2, note g
17. Recirculation System Sample Line Isolation Valves (3-XS-43-13, -14) (3-HS-43-13C, 14C)	1, note i
18. EECW Sectionalizing Valves (0-XS-67-13, -14, -17, -18, -21, -22, -25, -26) (0-HS-67-13C, -14C, -17C, -18C, -21C, -22C, -25C, -26C)	8, 1 per valve, note j
19. (Removed by Revision 88.)	
20. Recirculation Pump 3B Discharge Valve (3-XS-68-79) (3-HS-68-79C)	1
21. RWCU Drain to Main Condenser Hotwell Isolation Valve (3-XS-69-16) (3-HS-69-16C)	1
note e: There are 12 RHRSW pumps. All are equipped with emergency transfer switches. Backup Control must be available for 2 OPERABLE pumps aligned for EECW service (supports all units). Backup control for an additional 2 OPERABLE pumps must be available for RHRSW for RHR Loop I.	
note f: 1 Discharge Valve per RHR Loop I Heat Exchanger for a total of 2.	
note g: 1 per pump. Trip function necessary to prevent spurious start overloading 4-kV Buses/Diesel Generators.	
note h: Deleted.	
note i: 1 Recirculation System Sample Line Isolation Valve required, may be either inboard valve or outboard valve.	
note j: Not required if valve breaker remains open, per NFPA 805 requirement, except when required for valve testing or operation.	

BASES

BACKGROUND (continued)	There are two motor breakers provided for each of the two recirculation pumps for a total of four breakers. The output of each trip system is provided to one of the two breakers for each recirculation pump.
APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY	The ATWS-RPT initiates an RPT to aid in preserving the integrity of the fuel cladding following events in which a scram does not, but should, occur. Based on its contribution to the reduction of overall plant risk, however, the instrumentation meets Criterion 4 of the NRC Policy Statement (Ref. 3).

The OPERABILITY of the ATWS-RPT is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have a required number of OPERABLE channels in each trip system, with their setpoints within the specified Allowable Value of SR 3.3.4.2.3. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint). ATWS-RPT Channel OPERABILITY also includes the associated recirculation pump motor breakers. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Allowable Values are specified for each ATWS-RPT Function specified in the LCO. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between 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 predetermined values of output at which an action should take place. The setpoints are compared to the actual process

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

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

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

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

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

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

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

5.g. SLC System Initiation

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 4). An isolation signal for both RWCU isolation valves is initiated when the SLC pump start handswitch is not in the stop position.

There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch.

The SLC System Initiation Function is required to be OPERABLE in MODES 1 and 2, because these are the only MODES where the reactor can be critical, and in MODE 3 because this MODE uses the SLC System sodium pentaborate as a buffering solution to maintain the pH level at or above 7 in the suppression pool in the event of a LOCA. These MODES are consistent with the Applicability for the SLC System (LCO 3.1.7).

As noted (footnote (a) to Table 3.3.6.1-1), the SLC initiation signal provides input to the isolation logic for both RWCU isolation valves.

5.h. Reactor Vessel Water Level - Low, Level 3 (LIS-3-203A-D)

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, isolation of some interfaces with the reactor vessel occurs to isolate the potential sources of a break. The isolation of the RWCU System on Level 3 supports actions to ensure that the fuel peak cladding

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The pressure-temperature (P-T) curves included in the Technical Specifications have been developed to present steam-dome pressure versus minimum vessel metal temperature, incorporating appropriate non-beltline limits and irradiation embrittlement effects in the beltline. There are two sets of curves provided for Unit 3. The first set applies to operation up to and including 38 effective full power years (EFPY), and the second set applies to operation greater than 38 EFPY and less than or equal to 54 EFPY, where 54 EFPY represents the end of the renewed license and 38 EFPY is provided as a midpoint between the current EFPY and 54 EFPY. The P-T curves are provided in Figure 3.4.9-1 and Figure 3.4.9-2, respectively. Figure 3.4.9-1 contains P-T limit curves for mechanical heatup or cooldown following nuclear shutdown (bottom head and upper RPV/beltline) and for core operation (criticality). Figure 3.4.9-2 contains P-T limit curves for inservice leakage and hydrostatic testing (bottom head and upper RPV/beltline). The maximum rate of change of reactor coolant temperature is contained in SR 3.4.9.5, SR 3.4.9.6, and SR 3.4.9.7.

(continued)

BASES

BACKGROUND (continued)

The P-T curves incorporate a fluence calculated in accordance with GE Licensing Topical Report NEDC-32983P (Ref. 10), which has been approved by the NRC and is in compliance with Regulatory Guide 1.190 (Ref. 11). The fluence represents an Extended Power Uprate (EPU) for the rated power of 3952 MW_t, and is conservatively applied for the rated power of 3458 MW_t. The 1998 Edition of the ASME Section XI Boiler and Pressure Vessel Code including 2000 Addenda was used in accordance with 10 CFR 50.55a.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

(continued)

BASES

BACKGROUND (continued)

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, abnormal operational transients, and system hydrostatic tests. It mandates the use of the ASME Code, Section III, Appendix G (Ref. 2).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 3), Appendix H of 10 CFR 50 (Ref. 4), and BWRVIP-86 Revision1-A (Ref.12). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 5.

Minimum reactor vessel temperature requirements for pressure-temperature (P/T) limits depend on the reactor vessel's controlling material (which is either the material in the closure flange or the material in the beltline region with the highest reference temperature) and on the reactor's operating condition (i.e., hydrostatic pressure and leak tests, or normal operation including anticipated operation occurrences), vessel pressure, whether or not fuel is in the vessel, and whether the core is critical. The beltline region of the reactor vessel is the region that directly surrounds the effective height of the active core and adjacent regions that are predicted to experience sufficient radiation exposure to be considered in the selection of the most limiting material with regard to radiation exposure. The metal temperature of the controlling material, in the region of the controlling material which has the least favorable combination of stress and temperature, must exceed the appropriate minimum temperature requirements for the reactor vessel condition and pressure specified in 10 CFR 50, Appendix G.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1 (continued)

This SR has been modified by three Notes. Note 1 requires this Surveillance to be performed only during system heatup and cooldown operations or inservice leakage and hydrostatic testing. Also, Note 1 only requires this SR to be performed during inservice leakage and hydrostatic testing when reactor pressure is > 313 psig. Note 2 allows the limits of Figure 3.4.9-2 to be applied during nonnuclear heatup and ambient loss cooldown associated with inservice leak and hydrostatic testing provided that the heatup and cooldown rates are ≤ 15°F/hr. Note 3 provides that the limits of Figures 3.4.9-1 and 3.4.9-2 do not apply when the tension from the reactor head flange bolting studs is removed.

SR 3.4.9.2

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before withdrawing control rods that will make the reactor critical.

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the control rod withdrawal.

(continued)

BASES

REFERENCES (continued)

12. "BWRVIP-86 Revision 1-A: BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan," EPRI Technical Report 1016575, May 2013.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The PCIVs LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory, and establishing the primary containment boundary during major accidents. As part of the primary containment boundary, PCIV OPERABILITY supports leak tightness of primary containment. Therefore, the safety analysis of any event requiring isolation of primary containment is applicable to this LCO.

The DBAs that result in a release of radioactive material and are mitigated by PCIVs are a LOCA and a main steam line break (MSLB). In the analysis for each of these accidents, it is assumed that PCIVs are either closed or close within the required isolation times following event initiation. This ensures that potential paths to the environment through PCIVs (including primary containment purge valves) are minimized. Of the events analyzed in Reference 1, the LOCA is the most limiting event due to radiological consequences.

The closure time of the main steam isolation valves (MSIVs) is a significant variable from a radiological standpoint. The MSIVs are required to close within 3 to 5 seconds since the 5 second closure time is consistent with or conservative to the times assumed in the analyses. The safety analyses assume that the purge valves were closed at event initiation. Likewise, it is assumed that the primary containment is isolated such that release of fission products to the environment is controlled.

The DBA analysis assumes that primary containment is isolated within a certain time period (based on PCIV closure times provided in Reference 2 and inboard MSIV closure times of 2 minutes for a LOCA per Reference 9) and that leakage is terminated, except for the maximum allowable leakage rate, L_a. The primary containment isolation total response time includes signal delay, diesel generator startup (for loss of offsite power), and PCIV stroke times.

(continued)

BASES

REFERENCES

1. FSAR, Section 14.6.
2. BFN Technical Instruction (TI), 0-TI-360.
3. 10 CFR 50, Appendix J, Option B.
4. FSAR, Section 5.2.
5. FSAR, Section 14.6.5.
6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
7. FSAR Table 5.2-2.
8. General Electric NEDO-32977-A (Boiling Water Reactor Owners Group Topical Report, B21-00658-01), "Excess Flow Check Valve Testing Relaxation", dated June 2000.
9. MDQ0000012016000566 Revision 0, "Main Steam Isolation Valve (MSIV) Loss of Coolant Accident (LOCA) Closure Analysis," dated September 2016.

BASES

APPLICABLE SAFETY ANALYSES (continued)

With two and three units fueled, a worst case single failure could also include the loss of two RHRSW pumps caused by losing a 4 kV shutdown board since there are certain alignment configurations that allow two RHRSW pumps to be powered from the same 4 kV shutdown board. As discussed in the FSAR, Section 14.6.3.3.2 (Ref. 4) for these analyses, manual initiation of the OPERABLE RHRSW subsystems and the associated RHR System is assumed to occur 10 minutes after a DBA. The analyses assume that there are two RHRSW subsystems operating in each unit, with one RHRSW pump in each subsystem capable of producing 4000 gpm of flow. In this case, the maximum suppression chamber water temperature and pressure are 177°F (as reported in Reference 3) and 36.3 psig, respectively, well below the design temperature of 281°F and maximum allowable pressure of 62 psig.

The RHRSW System together with the UHS satisfies Criterion 3 of the NRC Policy Statement (Ref 5).

LCO

Four RHRSW subsystems are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst case single active failure occurs coincident with the loss of offsite power. Additionally, since the RHRSW pumps are shared between the three BFN units, the number of OPERABLE pumps required is also dependent on the number of units fueled.

An RHRSW subsystem is considered OPERABLE when:

a. At least one RHRSW pump (i.e., one required RHRSW pump) is OPERABLE, and

b. An OPERABLE flow path is capable of taking suction from the intake structure and transferring the water to the associated RHR heat exchanger at the assumed flow rate.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Turbine Bypass System

BASES

BACKGROUND

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is 25% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of nine valves connected to the main steam lines between the main steam isolation valves and the turbine stop valve bypass valve chest. Each of these nine valves is operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Pressure Regulator and Turbine Generator Control System, as discussed in the FSAR, Section 7.11 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows from the bypass chest, through connecting piping, to the pressure breakdown assemblies, where a series of orifices are used to further reduce the steam pressure before the steam enters the condenser.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)	SR 3.7.5.3 This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analysis. The response time limits are specified in the cycle specific transient analyses performed to support the preparation of FSAR, Appendix N, Supplemental Reload Licensing Report (Ref. 4). The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown the 24 month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint.
REFERENCES	1. FSAR, Section 7.11. 2. FSAR, Section 14.5.1.1 and 14.5.1.2. 3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993. 4. FSAR, Appendix N.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.1 AC Sources - Operating

BASES

BACKGROUND

The unit Class 1E AC Electrical Power Distribution System AC circuits consist of the offsite power sources (preferred power sources, normal and alternates), and the onsite standby power sources (Unit 1 and 2 diesel generators (DGs) A, B, C, and D, and Unit 3 DGs 3A, 3B, 3C, and 3D). As required by 10 CFR 50, Appendix A, GDC 17 (Ref. 1), the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the Engineered Safety Feature (ESF) systems.

The Class 1E AC distribution system is divided into redundant divisions, so loss of any one division does not prevent the minimum safety functions from being performed. Each of four 4.16 kV shutdown boards has two offsite power circuits available and a single DG.

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the 3EA and 3EB (Division I) or 3EC and 3ED (Division II) 4.16 kV shutdown boards. Offsite power is supplied to the 161 kV and 500 kV switchyards from the transmission network. One of the two required qualified offsite power circuits must be pre-aligned to the Division I and/or Division II 4.16 kV shutdown boards to meet the requirements of 10 CFR 50, Appendix A, GDC 17 (Ref. 1). Three basic circuits from the transmission network to the safety related Division I (3EA and 3EB 4.16 kV shutdown boards) and Division II (3EC and 3ED 4.16 kV shutdown boards), are as follows:

(continued)

BASES

LCO (continued)

An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the 3EA and 3EB (Division I) or 3EC and 3ED (Division II) 4.16 kV shutdown boards. One of the two required qualified offsite power circuits must be pre-aligned to the Division I and/or Division II 4.16 kV shutdown boards to meet the requirements of 10 CFR 50, Appendix A, GDC 17 (Ref. 1). Specific circuits are described below:

1. From the 500 kV switchyard through unit station service transformer (USST) 3B to 4.16 kV unit board 3A and/or 3B. Each unit board feeds two of the Unit 3 4.16 kV shutdown boards (3EA and 3EB or 3EC and 3ED).

2. From the 161 kV transmission network, through common station service transformer (CSST) A to start bus 1A or 1B, then to a 4.16 kV unit board. That unit board feeds two of the Unit 3 4.16 kV shutdown boards (3EA and 3EB or 3EC and 3ED).

3. From the 161 kV transmission network, through CSST B to start bus 1A or 1B, and then to a 4.16 kV unit board. That unit board feeds two of the Unit 3 4.16 kV shutdown boards (3EA and 3EB or 3EC and 3ED).

(continued)

BASES

ACTIONS (continued)

J.1

Condition J corresponds to a level of degradation in which all redundancy in the AC electrical power supplies has been lost. At this severely degraded level, any further losses in the AC electrical power system will cause a loss of function. Therefore, no additional time is justified for continued operation. The unit is required by LCO 3.0.3 to commence a controlled shutdown.

K.1

Required Action K.1 is intended to provide assurance that a loss of offsite power, during the period that a required Unit 1 and 2 DG is inoperable, does not result in a complete loss of safety function of critical systems (i.e., SGT, CREVS, RHRSW, or EECW). These features are designed with redundant safety related divisions (i.e., single division systems are not included). Redundant required features failures consist of inoperable features associated with a division redundant to the division that has an inoperable Unit 1 and 2 DG.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action the Completion Time only begins on discovery that both:

a. An inoperable required Unit 1 and 2 DG exists; and

b. A redundant required feature supported by another DG, is inoperable.

(continued)

BASES

ACTIONS

K.1 (continued)

If, at any time during the existence of this Condition (a required Unit 1 and 2 DG inoperable), a required redundant feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering a required Unit 1 and 2 DG inoperable coincident with an inoperable redundant feature that is associated with the OPERABLE DGs results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE DGs and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. Thus, on a component basis, single failure protection for the required feature's function may have been lost; however, function has not been lost. The 4 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 4 hour Completion Time takes into account the capacity and capability of the remaining AC sources, reasonable time for repairs, and low probability of a DBA occurring during this period.

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

BASES

BACKGROUND

Each diesel generator (DG) is provided with three interconnected storage tanks having a minimum usable fuel oil volume (35,280 gallons) sufficient to operate that DG for a period of 7 days while the DG is supplying maximum post loss of coolant accident (LOCA) load demand discussed in FSAR, Section 8.5.3.4 (Ref. 1) and Regulatory Guide 1.137 (Ref. 2). A transfer pump is located at the fuel oil storage tanks which can supply fuel oil from two 71,000-gallon fuel oil storage tanks to the 7-day storage tanks. Only the 7-day storage tanks are subject to this LCO, the Actions, and the Surveillance Requirements. In addition, it is possible to transfer fuel from one 7-day storage tank to any other by using transfer pumps. This onsite fuel oil capacity is sufficient to operate the DGs for longer than the time to replenish the onsite supply from outside sources.

Fuel oil is transferred from the 7-day storage tank to the day tank by either of two transfer pumps associated with each diesel generator. This is accomplished automatically by level switches on the day tank. Redundancy of pumps and piping precludes the failure of one pump, or the rupture of any pipe, valve, or tank to result in the loss of more than one DG. All 7-day tanks are embedded in the substructure of the Standby Diesel Generator Building.

For proper operation of the standby DGs, it is necessary to ensure the proper quality of the stored fuel oil. The fuel oil property monitored is the total particulate concentration. Periodic testing of the stored fuel oil total particulate concentration is a method to monitor the potential degradation related to long term storage and the potential impact to fuel filter plugging as a result of high particulate levels.

(continued)

BASES (continued)

APPLICABILITY

The AC sources (LCO 3.8.1 and LCO 3.8.2) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an abnormal operational transient or a postulated DBA. Because stored diesel fuel oil, lube oil, and starting air subsystem support LCO 3.8.1 and LCO 3.8.2, stored diesel fuel oil, lube oil, and starting air are required to be within limits when the associated DG is required to be OPERABLE.

ACTIONS

The ACTIONS Table is modified by a Note indicating that separate Condition entry is allowed for each DG. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable DG subsystem. Complying with the Required Actions for one inoperable DG subsystem may allow for continued operation, and subsequent inoperable DG subsystem(s) governed by separate Condition entry and application of associated Required Actions.

A.1

In this condition, the 7-day fuel oil supply for a DG is not available. However, the Condition is restricted to fuel oil level reductions that maintain at least a 6-day supply. The fuel oil level equivalent to a 6-day supply is 30,240 gallons. These circumstances may be caused by events such as:

a. Full load operation required for an inadvertent start while at minimum required level; or

b. Feed and bleed operations that may be necessitated by increasing particulate levels or any number of other oil quality degradations.

(continued)

ACTIONS

A.1 (continued)

This restriction allows sufficient time for obtaining the requisite replacement volume and performing the analyses required prior to addition of the fuel oil to the tank. A period of 48 hours is considered sufficient to complete restoration of the required level prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

B.1

In this Condition, the 7-day lube oil inventory, i.e., sufficient lube oil to support 7 days of continuous DG operation at full load conditions, is not available. However, the Condition is restricted to lube oil volume reductions that maintain at least a 6-day supply. The lube oil inventory equivalent to a 6-day supply is 150 gallons. This restriction allows sufficient time for obtaining the requisite replacement volume. A period of 48 hours is considered sufficient to complete restoration of the required volume prior to declaring the DG inoperable. This period is acceptable based on the remaining capacity (> 6 days), the low rate of usage, the fact that procedures will be initiated to obtain replenishment, and the low probability of an event during this brief period.

C.1

This Condition is entered as a result of a failure to meet the acceptance criterion for particulates. Normally, trending of particulate levels allows sufficient time to correct high particulate levels prior to reaching the limit of acceptability. Poor sample procedures (bottom sampling), contaminated sampling equipment, and errors in laboratory analysis can produce failures that do not follow a trend. Since the presence

(continued)

ACTIONS

C.1 (continued)

of particulates does not mean failure of the fuel oil to burn properly in the diesel engine, since particulate concentration is unlikely to change significantly between Surveillance Frequency intervals, and since proper engine performance has been recently demonstrated (within 31 days), it is prudent to allow a brief period prior to declaring the associated DG inoperable. The 7 day Completion Time allows for further evaluation, re-sampling, and re-analysis of the DG fuel oil.

D.1

With the new fuel oil properties defined in the Bases for SR 3.8.3.3 not within the required limits, a period of 30 days is allowed for restoring the stored fuel oil properties. This period provides sufficient time to test the stored fuel oil to determine that the new fuel oil, when mixed with previously stored fuel oil, remains acceptable, or to restore the stored fuel oil properties. This restoration may involve feed and bleed procedures, filtering, or a combination of these procedures. Even if a DG start and load was required during this time interval and the fuel oil properties were outside limits, there is high likelihood that the DG would still be capable of performing its intended function.

E.1

Only one of the two redundant air starting systems is required to support associated DG operability. With the starting air receiver pressure < 165 psig in the required starting air system, sufficient capacity to start the associated DG may not exist. The associated DG may be incapable of performing its intended function and must be immediately declared inoperable. This declaration also requires entry into applicable Conditions and Required Actions for an inoperable DG, LCO 3.8.1, "AC Sources - Operating."

(continued)

BASES

ACTIONS

F.1

With a Required Action and associated Completion Time not met, or the stored diesel fuel oil, lube oil, or starting air subsystem inoperable for reasons other than addressed by Conditions A through E, the associated DG may be incapable of performing its intended function and must be immediately declared inoperable.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.3.1

This SR provides verification that there is an adequate inventory of fuel oil in the storage tanks to support each DG's operation for 7 days at full load. The fuel oil level equivalent to a 7-day supply is 35,280 gallons when calculated in accordance with References 2 and 6. The required fuel storage volume is determined using the most limiting energy content of the stored fuel. Using the known correlation of diesel fuel oil absolute specific gravity or API gravity to energy content, the required diesel generator output, and the corresponding fuel consumption rate, the onsite fuel storage volume required for 7 days of operation can be determined. SR 3.8.3.3 requires new fuel to be tested to verify that the absolute specific gravity or API gravity is within the range assumed in the diesel fuel oil consumption calculations. The 7-day period is sufficient time to place the unit in a safe shutdown condition and to bring in replenishment fuel from an offsite location.

The 31 day Frequency is adequate to ensure that a sufficient supply of fuel oil is available, since low level alarms are provided and unit operators would be aware of any large uses of fuel oil during this period.

SR 3.8.3.2

This Surveillance ensures that sufficient lubricating oil inventory is available to support at least 7 days of full load operation for each DG. The lube oil inventory equivalent to a 7-day supply is 175 gallons and is based on the DG manufacturer's consumption values for the run time of the DG.

A 31 day Frequency is adequate to ensure that a sufficient lube oil supply is onsite, since DG starts and run time are closely monitored by the plant staff.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.3

The tests listed below are a means of determining whether new fuel oil is of the appropriate grade and has not been contaminated with substances that would have an immediate detrimental impact on diesel engine combustion. If results from these tests are within acceptable limits, the fuel oil may be added to the storage tanks without concern for contaminating the entire volume of fuel oil in the storage tanks. These tests are to be conducted prior to adding the new fuel to the 7-day storage tank(s), but in no case is the time between receipt of new fuel and conducting the tests to exceed 31 days following addition to the 7-day storage tanks. The tests, limits, and applicable ASTM Standards are as follows:

a. Sample the new fuel oil in accordance with ASTM D4057-12 (Ref. 7)

b. Verify in accordance with the tests specified in ASTM D975-14a (Ref. 7) that the sample has an absolute specific gravity at 60/60°F of ≥ 0.83 and ≤ 0.89 or an API gravity at 60°F of ≥ 27° and ≤ 39° when tested in accordance with ASTM D1298-12b (Ref. 7), a kinematic viscosity at 40°C of ≥ 1.9 centistokes and ≤ 4.1 centistokes, and a flash point of ≥ 125°F

c. Verify that the new fuel oil has a clear and bright appearance with proper color when tested in accordance with ASTM D4176-04 or a water and sediment content within limits when tested in accordance with ASTM D2709-96 (Ref. 7)

Failure to meet any of the above limits is cause for rejecting the new fuel oil, but does not represent a failure to meet the LCO because the fuel oil is not added to the 7-day storage tank.

Either prior to adding new fuel oil to the 7-day storage tanks or within 31 days following the initial new fuel oil sample, the fuel oil is analyzed to establish that the other properties specified

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.3

in Table 1 of ASTM D975-14a (Ref. 7) are met for new fuel oil when tested in accordance with ASTM D975-14a (Ref. 7), except that the analysis for sulfur may be performed in accordance with ASTM D4294-10 (Ref. 7). If these tests are not completed prior to adding new fuel oil to the 7-day storage tanks, the 31 day period is acceptable because the fuel oil properties of interest, even if they were not within stated limits, would not have an immediate effect on DG operation. This Surveillance ensures the availability of high quality fuel oil for the DGs.

Fuel oil degradation during long term storage shows up as an increase in particulates, mostly due to oxidation. The presence of particulates does not mean that the fuel oil will not burn properly in a diesel engine. The particulates can cause fouling of filters and fuel oil injection equipment, however, which can cause engine failure.

Particulate concentrations should be determined in accordance with ASTM D6217-11 (Ref. 7). This method involves a gravimetric determination of total particulate concentration in the fuel oil and has a limit of 10 mg/l. It is acceptable to obtain a field sample for subsequent laboratory testing in lieu of field testing. Because the 7-day tank consists of three interconnected tanks, samples are drawn from each tank.

The Frequency of this test takes into consideration fuel oil degradation trends that indicate that particulate concentration is unlikely to change significantly between Frequency intervals.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.3.4

This Surveillance ensures that, without the aid of the refill compressor, sufficient air start capacity for each DG is available. The system design requirements provide for at least one start cycle from one of two redundant air start systems without recharging. A start cycle is defined by the DG vendor, but usually is measured in terms of time (seconds of cranking) or engine cranking speed. The pressure specified in this SR is the lowest pressure at which at least one start attempt can be accomplished using one of two redundant air start systems.

The 31 day Frequency takes into account the capacity, capability, redundancy, and diversity of the AC sources and other indications available in the control room, including alarms, to alert the operator to below normal air start pressure.

SR 3.8.3.5

Microbiological fouling is a major cause of fuel oil degradation. There are numerous bacteria that can grow in fuel oil and cause fouling, but all must have a water environment in order to survive. Removal of water from the fuel storage tanks once every 31 days eliminates the necessary environment for bacterial survival. This is the most effective means of controlling microbiological fouling. In addition, it eliminates the potential for water entrainment in the fuel oil during DG operation. Water may come from any of several sources, including condensation, ground water, rain water, contaminated fuel oil, and from breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 2). This SR is for preventive maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during performance of the Surveillance.

BASES

REFERENCES

1. FSAR, Section 8.5.3.4.
2. Regulatory Guide 1.137, Revision 1, October 1979.
3. FSAR, Chapter 6.
4. FSAR, Chapter 14.
5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
6. ANSI N195-1976, " Fuel Oil Systems for Standby Diesel Generators," Section 5.4, April 1976
7. ASTM Standards: D4057-12; D4176-04; D2709-96; D1298-12b; D975-14a; D4294-10; D6217-11.

BASES

LCO (continued)

one or more of these boards become inoperable due to a failure not affecting the OPERABILITY of a board listed in Table B 3.8.7-1 (e.g., a breaker supplying a single MCC fails open), the individual loads on the board would be considered inoperable, and the appropriate Conditions and Required Actions of the LCOs governing the individual loads would be entered. However, if one or more of these boards is inoperable due to a failure also affecting the OPERABILITY of a board listed in Table B 3.8.7-1 (e.g., loss of a 4.16 kV shutdown board, which results in de-energization of all boards powered from the 4.16 kV shutdown board), then although the individual loads are still considered inoperable, the Conditions and Required Actions of the LCO for the individual loads are not required to be entered, since LCO 3.0.6 allows this exception (i.e., the loads are inoperable due to the inoperability of a support system governed by a Technical Specification; the 4.16 kV shutdown board).

If the 480 V Shutdown Board 3B is placed on the alternate supply 4.16 kV Shutdown Board 3EB, a LOCA/LOOP with a failure of the 250 V DC Battery Board 3 would disable the normal supply 4.16 kV Shutdown Board 3EC, and would also prevent the 480 V Shutdown Board 3B from load shedding its 480 V loads which would overload the alternate supply Diesel Generator 3B. This would result in the loss of 4.16 kV Shutdown Boards 3EB and 3EC which would impact both divisions ECCS in Unit 3. Therefore, the time limitations and restrictions on the associated drawings shall be adhered to whenever 480 V Shutdown Board 3B is on its alternate supply.

When 480 V Shutdown Board 3A is aligned to the alternate supply 4.16 kV Shutdown Board 3EB, a LOCA/LOOP with a failure of the 250 V DC Battery Board 1 would disable the normal supply 4.16 kV Shutdown Board 3EA, and would also prevent the 480 V Shutdown Board 3A from load shedding its 480 V loads which would overload the alternate supply Diesel Generator 3B. This would result in the loss of diesel

(continued)

BASES

LCO (continued)

generators 3A and 3B, associated 4.16 kV shutdown boards, and RHRSW pumps. Therefore, the restrictions on the associated drawings shall be adhered to whenever 480 V Shutdown Board 3A is on its alternate supply.

The Unit 3 diesel auxiliary boards have an alternate power supply from the other 480 V shutdown board. Interlocks prevent paralleling normal and alternate feeder breakers. The boards are considered inoperable when powered from their alternate feeder breakers because a single failure of the power source would affect both divisions.

The Unit 3 480 V RMOV boards 3A, 3B, 3D, and 3E have an alternate power supply from the other 480 V shutdown board. Interlocks prevent paralleling normal and alternate feeder breakers. The boards are considered inoperable when powered from their alternate feeder breakers because a single failure of the power source would affect both divisions.

The Unit 3 250 V DC RMOV boards 3A, 3B, and 3C have alternate power supplies from another 250 V Unit DC board. Interlocks prevent paralleling normal and alternate feeder breakers. The boards are considered inoperable when powered from their alternate feeder breakers because a single failure of the power source could affect both divisions depending on the board alignment.

If a 4.16 kV or 480 V shutdown board is aligned to its alternate 250 V DC control power source a single failure of the alternate power source could affect both ECCS divisions and common equipment needed to support the other units depending on the board alignment. Therefore, the restrictions on the associated drawings shall be adhered to whenever a 4.16 kV or 480 V shutdown board is on its alternate control power supply.

(continued)

ENCLOSURE 3

Tennessee Valley Authority
Browns Ferry Nuclear Plant
Units 1, 2, and 3

Technical Requirements Manual (TRM) Changes and Additions

Unit 1 Technical Requirements Manual Changes

TRM Page No.	Revision No.	Effective Date
3.3-22	Revision 123	07-30-2015
3.3-38	Revision 124	10-27-2015
3.3-39	Revision 124	10-27-2015
3.3-48	Revision 124	10-27-2015
3.3-51	Revision 124	10-27-2015
3.7-8	Revision 125	12-03-2015
3.7-9	Revision 125	12-03-2015
3.7-9a	Revision 125	12-03-2015
3.7-10	Revision 125	12-03-2015
3.7-11	Revision 125	12-03-2015
3.7-12	Revision 125	12-03-2015
3.7-13	Revision 125	12-03-2015
3.7-14	Revision 125	12-03-2015
3.7-15	Revision 125	12-03-2015
App. B (TVA-COLR-BF1C11), Rev. 2	Revision 123	08-19-2015
App. B (TVA-COLR-BF1C12), Rev. 0	Revision 132	09-27-2016

Unit 1 Technical Requirements Manual Bases Changes

TRMB Page No.	Revision No.	Effective Date
B 3.3-42	Revision 124	10-27-2015
B 3.3-49	Revision 124	10-27-2015
B 3.3-53a	Revision 124	10-27-2015
B 3.3-63	Revision 130	08-03-2016
B 3.7-8	Revision 125	12-03-2015
B 3.7-12	Revision 125	12-03-2015
B 3.7-12a	Revision 125	12-03-2015
B 3.7-13	Revision 125	12-03-2015
B 3.7-14	Revision 125	12-03-2015
B 3.7-14a	Revision 125	12-03-2015
B 3.7-15	Revision 125	12-03-2015
B 3.7-15a	Revision 125	12-03-2015
B 3.7-16	Revision 125	12-03-2015
B 3.7-17	Revision 125	12-03-2015
B 3.7-18	Revision 125	12-03-2015

ENCLOSURE 3

Tennessee Valley Authority
Browns Ferry Nuclear Plant
Units 1, 2, and 3

Technical Requirements Manual (TRM) Changes and Additions

Unit 2 Technical Requirements Manual Changes

TRM Page No.	Revision No.	Effective Date
3.3-39	Revision 124	10-27-2015
3.3-40	Revision 124	10-27-2015
3.3-49	Revision 124	10-27-2015
3.3-52	Revision 124	10-27-2015
3.7-8	Revision 125	12-03-2015
3.7-9	Revision 125	12-03-2015
3.7-9a	Revision 125	12-03-2015
3.7-10	Revision 125	12-03-2015
3.7-11	Revision 125	12-03-2015
3.7-12	Revision 125	12-03-2015
3.7-13	Revision 125	12-03-2015
3.7-14	Revision 125	12-03-2015
3.7-15	Revision 125	12-03-2015
App. B (TVA-COLR-BF2C19) Rev. 3	Revision 121	07-22-2015
App. B (TVA-COLR-BF2C19) Rev. 4	Revision 126	01-21-2016
App. B (TVA-COLR-BF2C19) Rev. 5	Revision 131	08-18-2016
App. B (TVA-COLR-BF2C20) Rev. 0	Revision 133	02-15-2017

Unit 2 Technical Requirements Manual Bases Changes

TRM Bases Page No.	Revision No.	Effective Date
B 3.3-42	Revision 124	10-27-2015
B 3.3-49	Revision 124	10-27-2015
B 3.3-54	Revision 124	10-27-2015
B 3.3-63	Revision 130	08-03-2016
B 3.7-8	Revision 125	12-03-2015
B 3.7-12	Revision 125	12-03-2015
B 3.7-12a	Revision 125	12-03-2015
B 3.7-13	Revision 125	12-03-2015
B 3.7-14	Revision 125	12-03-2015
B 3.7-14a	Revision 125	12-03-2015
B 3.7-15	Revision 125	12-03-2015
B 3.7-15a	Revision 125	12-03-2015
B 3.7-16	Revision 125	12-03-2015
B 3.7-17	Revision 125	12-03-2015
B 3.7-18	Revision 125	12-03-2015

ENCLOSURE 3

Tennessee Valley Authority
Browns Ferry Nuclear Plant
Units 1, 2, and 3

Technical Requirements Manual (TRM) Changes and Additions

Unit 3 Technical Requirements Manual Changes

TRM Page No.	Revision No.	Effective Date
3.3-38	Revision 124	10-27-2015
3.3-39	Revision 124	10-27-2015
3.3-48	Revision 124	10-27-2015
3.3-51	Revision 124	10-27-2015
3.7-8	Revision 125	12-03-2015
3.7-9	Revision 125	12-03-2015
3.7-10	Revision 125	12-03-2015
3.7-11	Revision 125	12-03-2015
3.7-12	Revision 125	12-03-2015
3.7-13	Revision 125	12-03-2015
3.7-14	Revision 125	12-03-2015
3.7-15	Revision 125	12-03-2015
App. B (TVA-COLR-BF3C17) Rev. 2	Revision 126	01-21-2016
App. B (TVA-COLR-BF3C18) Rev. 0	Revision 127	03-07-2016
App. B (TVA-COLR-BF3C18) Rev. 1	Revision 128	03-16-2016
App. B (TVA-COLR-BF3C18) Rev. 2	Revision 129	06-23-2016

Unit 3 Technical Requirements Manual Bases Changes

TRM Bases Page No.	Revision No.	Effective Date
B 3.3-42	Revision 124	10-27-2015
B 3.3-49	Revision 124	10-27-2015
B 3.3-54	Revision 124	10-27-2015
B 3.3-63	Revision 130	08-03-2016
B 3.7-8	Revision 125	12-03-2015
B 3.7-11	Revision 125	12-03-2015
B 3.7-11a	Revision 125	12-03-2015
B 3.7-12	Revision 125	12-03-2015
B 3.7-13	Revision 125	12-03-2015
B 3.7-13a	Revision 125	12-03-2015
B 3.7-14	Revision 125	12-03-2015
B 3.7-14a	Revision 125	12-03-2015
B 3.7-15	Revision 125	12-03-2015
B 3.7-16	Revision 125	12-03-2015
B 3.7-17	Revision 125	12-03-2015

EFFECTIVE PAGE LISTING — Implementation Date: October 3, 2016

Page No.	Revision No.	Effective Date
Title Page	0	Initial
Effective Page Listing	Revision 132	09-27-2016
i	Revision 114	01-08-2015
ii	Revision 102	10-24-2013
1.0-1	Revision 114	01-08-2015
1.1-1	0	Initial
1.1-2	0	Initial
1.1-3	0	Initial
1.1-4	Revision 63	04-30-2007
1.1-5	0	Initial
1.1-6	0	Initial
1.2-1	0	Initial
1.2-2	0	Initial
1.2-3	0	Initial
1.3-1	0	Initial
1.3-2	0	Initial
1.3-3	0	Initial
1.3-4	0	Initial
1.3-5	0	Initial
1.3-6	0	Initial
1.3-7	0	Initial
1.3-8	0	Initial
1.3-9	0	Initial
1.3-10	0	Initial
1.3-11	0	Initial
1.3-12	0	Initial
1.3-13	0	Initial
1.4-1	Revision 20	03-13-2001
1.4-2	Revision 20	03-13-2001
1.4-2a	Revision 20	03-13-2001
1.4-3	0	Initial
1.4-4	0	Initial
1.4-5	Revision 20	03-13-2001
1.4-6	Revision 20	03-13-2001

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EFFECTIVE PAGE LISTING (continued)

Page No.	Revision No.	Effective Date
1.4-7	Revision 20	03-13-2001
3.0-1	Revision 111	10-10-2014
3.0-2	Revision 44	03-22-2004
3.0-3	Revision 111	10-10-2014
3.0-4	Revision 102	10-24-2013
3.0-5	Revision 44	03-22-2004
3.1-1	Revision 103	11-1-2013
3.3-1	Revision 79	10-18-2010
3.3-2	Revision 60	02-01-2007
3.3-3	Revision 45	09-10-2004
3.3-4	Revision 45	09-10-2004
3.3-5	Revision 111	10-13-2014
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App. A-1	0	Initial
App. B (TVA-COLR-BF1C12) Revision 0	Revision 132	09-27-2016

---NOTE---

1. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into Associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated function maintains initiation capability.
2. Refer to Table 3.3.3.5-1 to determine which TSRs apply for each function.

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.3.3.5.1	Perform CHANNEL FUNCTIONAL TEST.	92 days
TSR 3.3.3.5.2	Perform CHANNEL CALIBRATION.	24 months

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. As required by Required Action A.1 and referenced in Table 3.3.5-1.	D.1	Monitor torus temperature to observe any unexplained temperature increase which might be indicative of an open relief valve.	Once per 12 hours
	<u>AND</u>		
	D.2	Restore control room indication by either the Tailpipe Thermocouple Temperature or Acoustic Monitor to OPERABLE status for each relief valve.	30 days
	<u>AND</u>		
	D.3	When inoperable for more than 30 days, enter the condition into the Corrective Action Program.	24 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action A.1 and referenced in Table 3.3.5-1.	------------------NOTE---------------- Required Actions E.1.1 and E.1.2 are not applicable when in MODES 4 and 5. ---------------------------------------	
	E.1.1 Restore required control room indication channel to OPERABLE status.	72 hours
	<u>OR</u>	
	E.1.2 Initiate the preplanned alternate method of monitoring the parameter.	72 hours
	<u>AND</u>	
	E.2 When inoperable for more than seven days, enter the condition into the Corrective Action Program.	24 hours

(continued)

TR 3.3 INSTRUMENTATION

TR 3.3.7 Meteorological Monitoring Instrumentation

LCO 3.3.7 The meteorological monitoring instrumentation listed in Table 3.3.7-1 shall be OPERABLE.

APPLICABILITY: At all times

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The number of OPERABLE meteorological monitoring channels less than required by Table 3.3.7-1.	A.1 Restore inoperable channel(s) to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Enter the condition into the Corrective Action Program.	24 hours

TR 3.3 INSTRUMENTATION

TR 3.3.8 Seismic Monitoring Instrumentation

LCO 3.3.8 The seismic monitoring instruments listed in Table 3.3.8-1 shall be OPERABLE.

APPLICABILITY: At all times

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more seismic monitoring instruments in Panel 1-PNLA-9-44 or foundation instrument BFN-0-ACGR-052-0001 on the Unit 1 Reactor Building Base Slab inoperable.	A.1	Restore inoperable instrument(s) to an OPERABLE status.	30 days
B. One or more seismic monitoring instruments BFN-0-ACGR-052-0002 or BFN-0-ACGR-052-0003 inoperable.	B.1	Restore inoperable instrument(s) to an OPERABLE status.	60 days
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	Enter the condition into the Corrective Action Program.	24 hours

---NOTES---

1. As used in this Technical Requirement, "code snubber" shall mean snubbers that are identified by BFN ASME Code IST Program as ASME Class 1, 2, and 3 equivalent snubbers. It shall also mean those BFN safety-related snubbers that are not identified as ASME Class 1, 2, and 3 equivalent, but are treated as such.
2. Each code snubber, identified by those snubbers listed in plant procedures, shall be demonstrated OPERABLE by performance of the following inservice examination and test program requirements, which is derived from ASME OM Code Subsection ISTD.
3. As used in this Technical Requirement, "type of snubber" shall mean snubbers of the same dsign and manufacturer, irrespective of capacity.
4. As used in this Technical Requirement, "population or category" shall mean the total number of snubbers being visually inspected as a lot.

TECHNICAL SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
TSR 3.7.4.1	---NOTE--- At BFN, code snubbers are visually examined as one population or category, regardless of accessibility, in accordance with the schedule and interval determined by Table 3.7.4-1, which is Table ISTD-4252-1. The first inspection interval determined using Table 3.7.4-1 criteria shall be based on the previous inspection interval established by the requirements in effect before Revision 007 of these Technical Requirements were issued. --- Perform visual examination of required snubber(s) based on the acceptance criteria of plant visual examination procedures and the frequency based on Table 3.7.4-1, which are both based on ASME OM Code, Subsection ISTD. Visual examination shall confirm: a. No visible indications of damage, leakage,	In accordance with Table 3.7.4-1.

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.7.4.1 (continued)	corrosion, degradation or other external characteristics that may indicate impaired OPERABILITY, pin to pin, inclusive. b. Fasteners for the attachment of the snubber are functional. c. No indication of binding, misalignment, or deformation of the snubber. d. Hydraulic snubber fluid is at the recommended level and that vented reservoir is oriented such that fluid can gravitate to snubber body. e. The absence of weld arc strikes, paint, weld slag, adhesive, or other deposits on piston rod or support cylinder that could result in unacceptable snubber performance. f. Snubber spherical bearing is fully engaged in attachment lug. g. Snubber position setting is adequate Snubbers which appear inoperable as a result of visual inspection shall be classified unacceptable. Snubbers confirmed as unacceptable snubbers are adjusted, repaired, modified, or replaced, and counted in determination of the subsequent examination interval in accordance with Table 3.7.4-1, regardless if the affected snubber is functionally tested in the as-found condition and determined OPERABLE per the criteria of TSR 3.7.4.2. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable	

(continued)

SURVEILLANCE		FREQUENCY
TSR 3.7.4.1 (continued)	snubber. If continued operation cannot be justified, the system or train shall be declared inoperable. Additionally, snubbers attached to sections of safety related systems that have experienced unexpected potentially damaging transients since the last inspection period shall be evaluated for the possibility of concealed damage and functionally tested, if applicable, to confirm OPERABILITY. Snubbers which have been made inoperable as the result of unexpected transients, isolated damage, or other random events, when the provisions of TSR 3.7.4.5. and TSR 3.7.4.6 have been met and any other appropriate corrective action implemented, shall not be counted in determining the next visual inspection interval.	

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.7.4.2	Perform an in-place or bench functional test of a representative sample of 10% of the total of each type of code snubber(s). Sample population is rounded up to next whole integer. Safety-related snubbers that are not ASME Class 1, 2, or 3 equivalent snubbers shall not be included in the snubber population when selecting the initial or additional samples. a. As practical, the representative sample selected for functional testing shall include representation from the Defined Test Plan Group (DTPG) based on the significant features (i.e., the various designs, configurations, operating environments, and the range of size and capacity of snubbers within the types) and based on the ratio of the number of snubbers of each significant feature, to the total number of snubbers in the DTPG. b. The sample shall be generally representative as specified in ISTD-5311 (a), but may also be selected from snubbers concurrently scheduled for seal replacement or other similar activity related to service life monitoring. The snubbers shall be tested on a generally rotational basis to coincide with the service life monitoring. c. The representative sample should be weighed to include more snubbers from severe service areas such as near heavy equipment. d. After testing, at the time of reinstallation, the snubber shall meet visual examination attributes as described in ISTD-4110(a), -4110(c), -4110(d), and -4110(e). The stroke setting shall be verified.	In accordance with Inservice Testing Program

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.7.4.2 (continued)	Functional Test Acceptance Criteria: The snubber functional test shall verify that: a. Activation (restraining action) is achieved in both tension and compression within the specified range of velocity or accelerations. b. Snubber bleed or release, where required, is present in both compression and tension within the specified range. c. For mechanical snubbers, the drag force is within the specified limits, in tension and compression. d. For hydraulic snubbers, if required to verify proper assembly, drag force is within specific limits in tension and compression. e. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified. f. Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.	

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TSR 3.7.4.3	A failure analysis shall be made of each failure to meet the functional test acceptance criteria of TSR 3.7.4.2 to determine the cause of the failure. The result of this analysis shall be used, if applicable, in selecting snubbers to be tested in the subsequent lot in an effort to determine the OPERABILITY of other snubbers which may be subject to the same failure mode. Selection of snubbers for future testing may also be based on the failure analysis. For each failed snubber, perform in-place or bench functional test on an additional lot equal to 5% of the subject snubber DTPG. When establishing additional sample testing, failure analysis results of unacceptable snubbers are to be used to determine if establisheding a FMG is appropriate. Additional lot (i.e., sample) population is rounded up to next whole integer. Rounding up satisfies ISTD-5312 requirements that additional samples shall be at least one-half the size of the initial sample from that DTPG. Additional samples selected from DTPGs follow the composition of the original sample. However, additional samples selected from FMGs are random samples. Testing shall continue until no additional inoperable snubbers are found within subsequent lots or all snubbers of the original test type are tested or all suspect snubbers identified by the failure analysis have been tested, as applicable. The functional test criteria shall be as specified in TSR 3.7.4.2. Prior to functional testing the discovery of loose or missing attachment fasteners will be evaluated to determine whether the cause may be localized or generic. The result of the evaluation will be used to select other suspect snubbers for verifying the attachment fasteners, as applicable.	Once for each discovery of snubber failure to meet functional test acceptance criteria

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TSR 3.7.4.4	(Deleted by TRM Revision 125)	
TSR 3.7.4.5	Perform an engineering evaluation on the component or system which is restrained by the snubber(s) found inoperable due to not meeting their functional test acceptance criteria as specified in TSR 3.7.4.2.	Once for each discovery of an inoperable snubber
TSR 3.7.4.6	Verify replacement snubbers and snubbers having repairs, adjustments, or modifications which might affect the functional test results meet the test criteria of TSR 3.7.4.2. a. These snubbers shall have met the acceptance criteria subsequent to their most recent service; and b. The functional test must have been performed within the 12 months prior to being installed in the unit.	Once prior to installation in the unit for each replaced, repaired, adjusted, or modified snubber where functional test results might be affected

Table 3.7.4-1
Visual Examination Table
(Ref. Table ISTD-4252-1)

	NUMBER OF UNACCEPTABLE SNUBBERS		
Population or Category (Note 1)	Column A Extend Interval (Notes 2 and 3)	Column B Repeat Interval (Notes 2, 4, and 5)	Column C Reduce Interval 2/3 (Notes 2, 5, and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or more	29	56	109

Note 1: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. The next lower integer shall be used when interpolation results in a fraction.

Note 2: The basic interval shall be the normal fuel cycle up to 24 months. The examination interval may be as great as twice, the same, or as small as fractions of the previous interval as required by the following Notes. The examination interval may vary + / - 25% of the current interval.

Table 3.7.4-1
Visual Examination Table
(Ref. Table ISTD-4252-1)

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months. In that case, the next visual examination according to the previous interval may be skipped.

Note 4: If the number of unacceptable snubbers is equal or less than the number in Column B, but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is less than the number in Column C, but greater than the number is Column B, the next interval shall be reduced to two-thirds of the previous examination interval or, in accordance with the interpolation between Columns B and C, in proportion to the exact number of unacceptable snubbers.

Note 6: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval.

The provisions of TSR 3.0.1, 3.0.2, and 3.0.3 are applicable for all inspection intervals up to and including 48 months.

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QA Document
Pages Affected: All
BFE-3756, Revision 2

TVA NPG

Reactor Engineering and Fuels - BWRFE
1101 Market Street, Chattanooga, TN 37402

Browns Ferry Unit 1 Cycle 11
Core Operating Limits Report, (105% OLTP)

TVA-COLR-BF1C11 **Revision 2 (Final)**
(Revision Log, Page v)

August 2015

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0-R2		No Changes to Section 2; Sections 5 through 8. Other changes driven by CR 1014002.
1-R2	viii	Updated References 1 & 5.
2-R2	1	Added Section 1.1.1; identify new basis of revised reload safety analysis report.
3-R2	11	Updated Figure 3.3.
4-R2	15-16	Updated Figures 3.7 and 3.8.
5-R2	19	Edited Table number cross referencing.
6-R2	23-42	Updated Tables 4.2 through 4.17
1-R1	11	Implement QA Recommendation (PER 960548)
2-R1	17	First equation was fixed to include the $LHGR_{SLO}$ (which was defined below the equation). Constitutes a clarification of the generalized representation for LHGR. No impact on limits. (PER 951981)
3-R1	31	Section 4.2.5: Fixed table pointers to indicate correct tables. Values in tables are not altered. (PER 951981)
0-R0	All	New document.

Nomenclature

APLHGR	Average Planar LHGR
APRM	Average Power Range Monitor
AREVA NP	Vendor (Framatome, Siemens)
BOC	Beginning of Cycle
BSP	Backup Stability Protection
BWR	Boiling Water Reactor
CAVEX	Core Average Exposure
CD	Coast Down
CMSS	Core Monitoring System Software
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRWE	Control Rod Withdrawal Error
CSDM	Cold SDM
DIVOM	Delta CPR over Initial CPR vs. Oscillation Magnitude
EOC	End of Cycle
EOCLB	End-of-Cycle Licensing Basis
EOOS	Equipment OOS
FFTR	Final Feedwater Temperature Reduction
FFWTR	Final Feedwater Temperature Reduction
FHOOS	Feedwater Heaters OOS
ft	Foot: english unit of measure for length
GNF	Vendor (General Electric, Global Nuclear Fuels)
GWd	Giga Watt Day
HTSP	High TSP
ICA	Interim Corrective Action
ICF	Increased Core Flow (beyond rated)
IS	In-Service
kW	kilo watt: SI unit of measure for power.
LCO	License Condition of Operation
LFWH	Loss of Feedwater Heating
LHGRFAC	LHGR Multiplier (Power or Flow dependent)
LPRM	Low Power Range Monitor
LRNB	Generator Load Reject, No Bypass
MAPFAC	MAPLHGR multiplier (Power or Flow dependent)

MCPR	Minimum CPR
MSRV	Moisture Separator Reheater Valve
MSRVOOS	MSRV OOS
MTU	Metric Ton Uranium
MWd/MTU	Mega Watt Day per Metric Ton Uranium
NEOC	Near EOC
NRC	United States Nuclear Regulatory Commission
NSS	Nominal Scram Speed
NTSP	Nominal TSP
OLMCPR	MCPR Operating Limit
OOS	Out-Of-Service
OPRM	Oscillation Power Range Monitor
OSS	Optimum Scram Speed
PBDA	Period Based Detection Algorithm
Pbypass	Power, below which TSV Position and TCV Fast Closure Scrams are Bypassed
PLU	Power Load Unbalance
PLUOOS	PLU OOS
PRNM	Power Range Neutron Monitor
RBM	Rod Block Monitor
RCPOOS	Recirculation Pump OOS (*SLO*)
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPTOOS	RPT OOS
SDM	Shutdown Margin
SLMCPR	MCPR Safety Limit
SLO	Single Loop Operation
TBV	Turbine Bypass Valve
TBVIS	TBV IS
TBVOOS	Turbine Bypass Valves OOS
TIP	Transversing In-core Probe
TIPOOS	TIP OOS
TLO	Two Loop Operation
TSP	Trip Setpoint
TSSS	Technical Specification Scram Speed
TVA	Tennessee Valley Authority

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25. EMF-CC-074(P)(A), Volume 4, Revision 0, **BWR Stability Analysis: Assessment of STAIF with Input from MICROBURN-B2**, Siemens Power Corporation, August 2000.

26. BAW-10255(P)(A), Revision 2, **Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code**, AREVA NP Inc., Inc., May, 2008.

<u>PRNM Setpoint References</u>

27. Filtered Setpoints - EDE-28-0990 Rev. 3 Supplement E, "PRNM (APRM, RBM, and RFM) Setpoint Calculations [ARTS/MELLL (NUMAC) - Power-Uprate Condition] for Tennessee Valley Authority Browns Ferry Nuclear Plant", October 1997.

28. Unfiltered Setpoints - EDE-28-0990 Rev. 2 Supplement E, "PRNM (APRM, RBM, and RFM) Setpoint Calculations [ARTS/MELLL (NUMAC) - Power-Uprate Condition] for Tennessee Valley Authority Browns Ferry Nuclear Plant", October 1997.

29. GE Letter LB#: 262-97-133, Browns Ferry Nuclear Plant Rod Block Monitor Setpoint Clarification - GE Proprietary Information, September 12, 1997.

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

31. NEDO-32465-A, **Licensing Topical Report – Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications**, GE Nuclear Energy, August 1996.

TVA NPG

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Date: August 13, 2015

1 Introduction

In anticipation of cycle startup, it is necessary to describe the expected limits of operation.

1.1 Purpose

The primary purpose of this document is to satisfy requirements identified by unit technical specification section 5.6.5. This document may be provided, upon final approval, to the NRC.

1.1.1 Revision 2

The purpose of this revision is a consequence of the Reference 1 reload safety analyses revision. The Reference 1 revision is in response to condition report (CR) 1014002.

1.2 Scope

This document will discuss the following areas:

- Average Planar Linear Heat Generation Rate (APLHGR) Limit
  (Technical Specifications 3.2.1 and 3.7.5)
- Linear Heat Generation Rate (LHGR) Limit
  (Technical Specification 3.2.3, 3.3.4.1, and 3.7.5)
- Minimum Critical Power Ratio Operating Limit (OLMCPR)
  (Technical Specifications 3.2.2, 3.3.4.1, 3.7.5 and Table 3.3.2.1-1)
- Oscillation Power Range Monitor (OPRM) Setpoint
  (Technical Specification Table 3.3.1.1)
- Average Power Range Monitor (APRM) Flow Biased Rod Block Trip Setting
  (Technical Requirements Manual Section 5.3.1 and Table 3.3.4-1)
- Rod Block Monitor (RBM) Trip Setpoints and Operability
  (Technical Specification Table 3.3.2.1-1)
- Shutdown Margin (SDM) Limit
  (Technical Specification 3.1.1)

1.3 Fuel Loading

The core will contain previously exposed GNF GE14, and AREVA NP, Inc., ATRIUM-10 fuel, along with fresh ATRIUM-10 fuel. Nuclear fuel types used in the core loading are shown in Table 1.1. The core shuffle and final loading were explicitly evaluated for BOC cold shutdown margin performance as documented per Reference 6.

1.4 Acceptability

Limits discussed in this document were generated based on NRC approved methodologies per References 7 through 26.

Table 1.1 Nuclear Fuel Types*

Fuel Description	Original Cycle	Number of Assemblies	Nuclear Fuel Type (NFT)	Fuel Names (Range)
GE14-P10DNAB406-15GZ-100T-150-T6-3079	8	1	16	JYE295
GE14-P10DNAB408-16GZ-100T-150-T6-3363	9	111	1	JYP101-JYP280
GE14-P10DNAB412-16GZ-100T-150-T6-3364	9	40	2	JYP281-JYP320
GE14-P10DNAB404-15GZ-100T-150-T6-3365	9	9	8	JYP321-JYP336
GE14-P10DNAB408-17GZ-100T-150-T6-3366	9	16	19	JYP337-JYP352
ATRIUM-10 A10-3562B-14GV80-FAA	10	168	20	FAA001-FAA168
ATRIUM-10 A10-3676B-10GV80-FAA	10	24	21	FAA169-FAA192
ATRIUM-10 A10-4111B-15GV80-FAA	10	87	22	FAA193-FAA280
ATRIUM-10 A10-3840B-14GV80-FAB	11	200	23	FAB301-FAB500
ATRIUM-10 A10-4117B-13GV70-FAB	11	72	24	FAB501-FAB572
ATRIUM-10 A10-4112B-15GV70-FAB	11	36	25	FAB573-FAB608

* The table identifies the expected fuel type breakdown in anticipation of final core loading. The final composition of the core depends upon uncertainties during the outage such as discovering a failed fuel bundle, or other bundle damage. Minor core loading changes, due to unforeseen events, will conform to the safety and monitoring requirements identified in this document.

2 APLHGR Limits

(Technical Specifications 3.2.1 & 3.7.5)

The APLHGR limit is determined by adjusting the rated power APLHGR limit for off-rated power, off-rated flow, and SLO conditions. The most limiting of these is then used as follows:

$$\text{APLHGR limit} = \text{MIN} (\text{APLHGR}_P , \text{APLHGR}_F, \text{APLHGR}_{SLO})$$

where:

APLHGR_P	off-rated power APLHGR limit	[APLHGR_{RATED} * MAPFAC_P]
APLHGR_F	off-rated flow APLHGR limit	[APLHGR_{RATED} * MAPFAC_F]
APLHGR_{SLO}	SLO APLHGR limit	[APLHGR_{RATED} * SLO Multiplier]

2.1 Rated Power and Flow Limit: APLHGR_{RATED}

The rated conditions APLHGR for ATRIUM-10 fuel is identified in Reference 1 and shown in Figure 2.1; for GE 14 fuel is identified in Reference 2, and shown in Figure 2.2.

2.2 Off-Rated Power Dependent Limit: APLHGR_P

Reference 1, for both GE14 and ATRIUM-10 fuel, does not specify a power dependent APLHGR. Therefore, MAPFAC_P is set to a value of **1.0**.

2.2.1 Startup without Feedwater Heaters

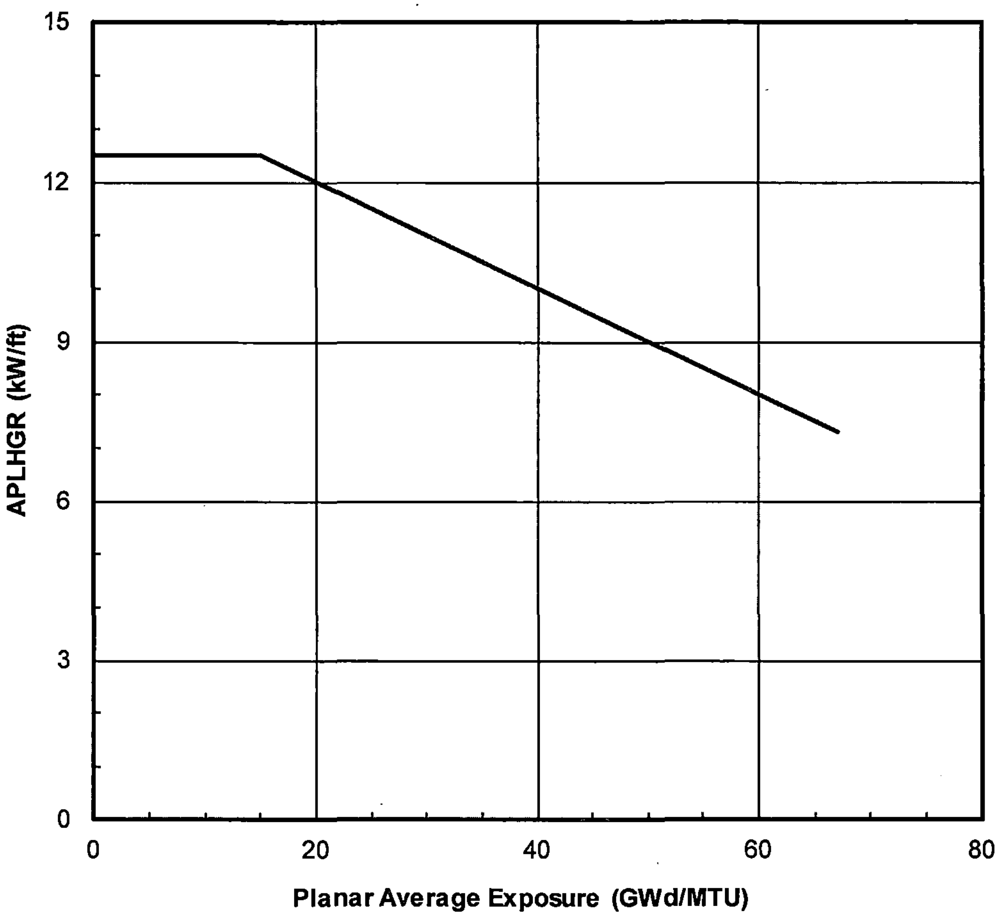
There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. No Additional power dependent limitation is required.

2.3 Off-Rated Flow Dependent Limit: APLHGR_F

Reference 1, for GE14 and ATRIUM-10 fuel, does not specify a flow dependent APLHGR. Therefore, MAPFAC_F is set to a value of **1.0**.

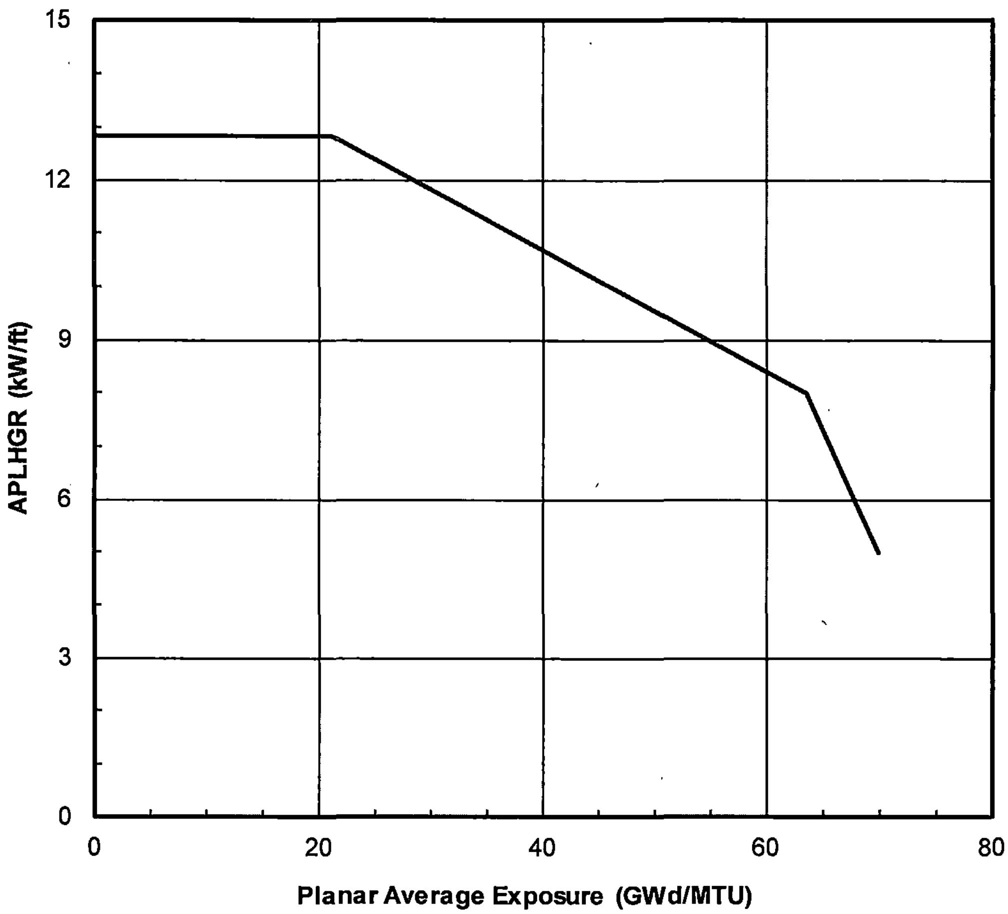
2.4 Single Loop Operation Limit: APLHGR_{SLO}

The single loop operation multiplier for ATRIUM-10 fuel is **0.85**, per Reference 1; for GE 14 fuel the multiplier is **0.93** per Reference 2.



Planar Avg. Exposure (GWd/MTU)	APLHGR Limit (kW/ft)
0.0	12.5
15.0	12.5
67.0	7.3

Figure 2.1 APLHGR$_{RATED}$ for ATRIUM-10 Fuel



Planar Avg. Exposure (GWd/MTU)	APLHGR Limit (kW/ft)
0.00	12.82
21.09	12.82
63.50	8.00
70.00	5.00

Figure 2.2 $APLHGR_{RATED}$ for GE 14 Fuel

2.5 Equipment Out-Of-Service Corrections

The limits shown in Figure 2.1 and Figure 2.2 are applicable for operation with all equipment In-Service as well as the following Equipment Out-Of-Service (EOOS) options; including combinations of the options.

In-Service	All equipment In-Service*
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
PLUOOS	Power Load Unbalance Out-Of-Service
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service or Final Feedwater Temperature Reduction
RCPOOS	One Recirculation Pump Out-Of-Service†

* All equipment service conditions assume 1 SRVOOS.

† Requires application of multipliers to rated APLHGR limits as described in Section 2.4.

3 LHGR Limits

(Technical Specification 3.2.3, 3.3.4.1, & 3.7.5)

The LHGR limit is determined by adjusting the rated power LHGR limit for off-rated power and off-rated flow conditions. The most limiting of these is then used as follows:

$$\text{LHGR limit} = \text{MIN} (\text{LHGR}_P, \text{LHGR}_F, \text{LHGR}_{SLO})$$

where:

$LHGR_P$	off-rated power LHGR limit	[$LHGR_{RATED}$ * $LHGRFAC_P$]
$LHGR_F$	off-rated flow LHGR limit	[$LHGR_{RATED}$ * $LHGRFAC_F$]
$LHGR_{SLO}$	SLO LHGR limit	[$LHGR_{RATED}$ * SLO Multiplier]

3.1 Rated Power and Flow Limit: $LHGR_{RATED}$

The rated conditions LHGR for ATRIUM-10 fuel is identified in Reference 1 and shown in Figure 3.1; for GE 14 fuel, is identified in Reference 3, and shown in Figure 3.2 for UO_2 fuel. Separate, concentration dependent limits apply for rods containing Gadolinium; LHGR limits are provided in Reference 3.

3.2 Off-Rated Power Dependent Limit: $LHGR_P$

LHGR limits are adjusted for off-rated power conditions using the $LHGRFAC_P$ multiplier provided in Reference 1, for both ATRIUM-10 and GE 14 fuel. The multiplier is split into two sub cases: turbine bypass valves in and out-of-service. The multipliers are shown in Figure 3.3 and Figure 3.4.

3.2.1 Startup without Feedwater Heaters

There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. Additional limits are shown in Figure 3.7 thru Figure 3.10, based on temperature conditions identified in Table 3.1.

Table 3.1 Startup Feedwater Temperature Basis

Power	Temperature	
	Range 1	Range 2
(% Rated)	(°F)	(°F)
25	160.0	155.0
30	165.0	160.0
40	175.0	170.0
50	185.0	180.0

3.3 Off-Rated Flow Dependent Limit: $LHGR_F$

LHGR limits are adjusted for off-rated flow conditions using the $LHGRFAC_F$ multiplier provided in Reference 1, for both ATRIUM-10 and GE 14 fuel. Multipliers are shown in Figure 3.5 and Figure 3.6.

3.4 Single Loop Operation Limit: $LHGR_{SLO}$

The single loop operation multiplier is **0.93**, per Reference 2 for GE 14 fuel. There is no single loop operation restriction for ATRIUM-10; therefore the multiplier is **1.0**.

3.5 Equipment Out-Of-Service Corrections

The limits shown in Figure 3.1 and Figure 3.2 are applicable for operation with all equipment In-Service as well as the following Equipment Out-Of-Service (EOOS) options; including combinations of the options.*

In-Service	All equipment In-Service
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
PLUOOS	Power Load Unbalance Out-Of-Service
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service or Final Feedwater Temperature Reduction
RCPOOS	One Recirculation Pump Out-Of-Service†

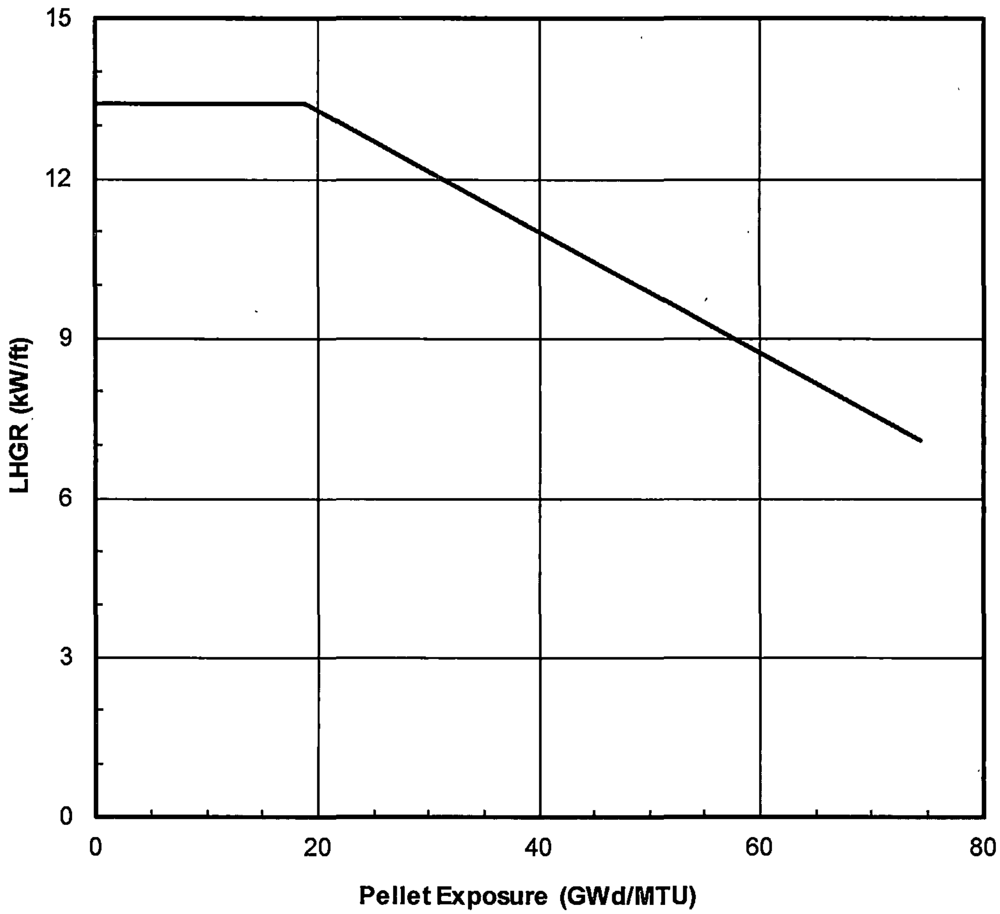
Off-rated power corrections shown in Figure 3.3 and Figure 3.4 are dependent on operation of the Turbine Bypass Valve system. For this reason, separate limits are to be applied for TBVIS or TBVOOS operation. The limits have no dependency on RPTOOS, PLUOOS, FHOOS/FFWTR, or RCPOOS.

Off-rated flow corrections shown in Figure 3.5 and Figure 3.6 are bounding for all EOOS conditions.

Off-rated power corrections shown in Figure 3.7 through Figure 3.10 are also dependent on operation of the Turbine Bypass Valve system. In this case, limits support FHOOS operation during startup. These limits have no dependency on RPTOOS, PLUOOS, or RCPOOS.

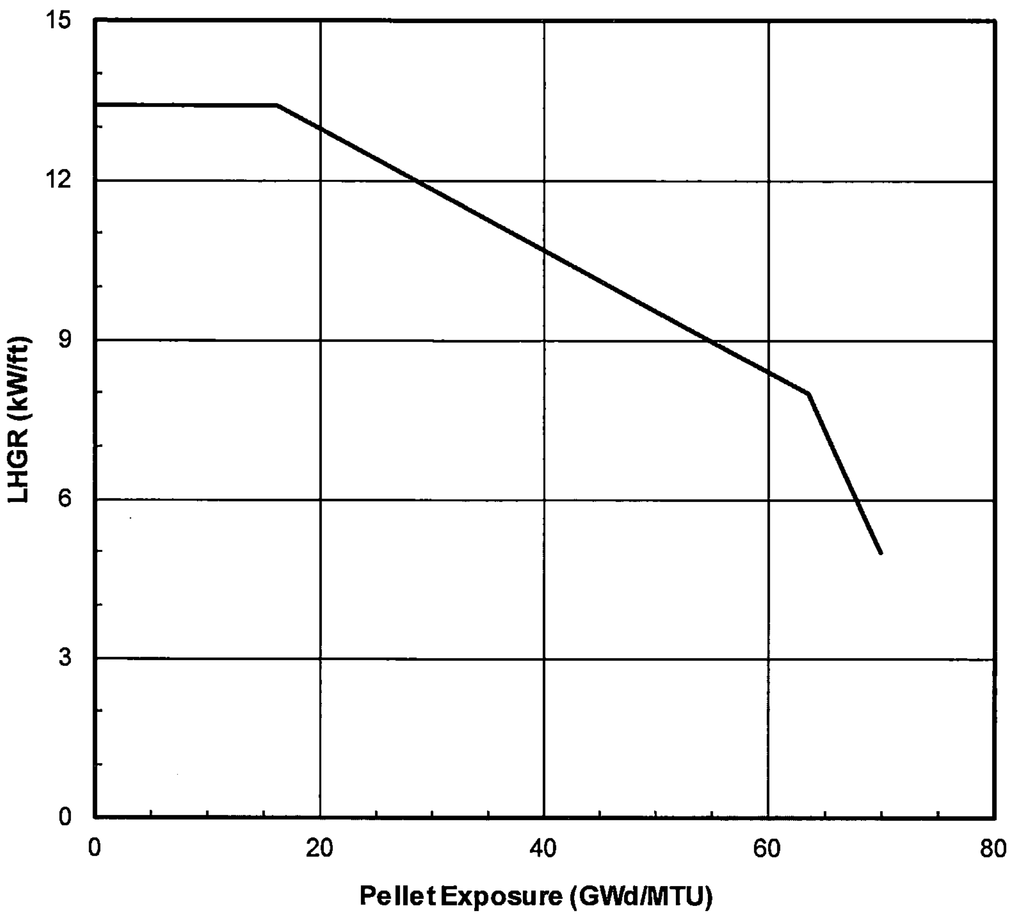
* All equipment service conditions assume 1 SRVOOS.

† Requires application of multipliers to rated LHGR limits as described in Section 3.4.



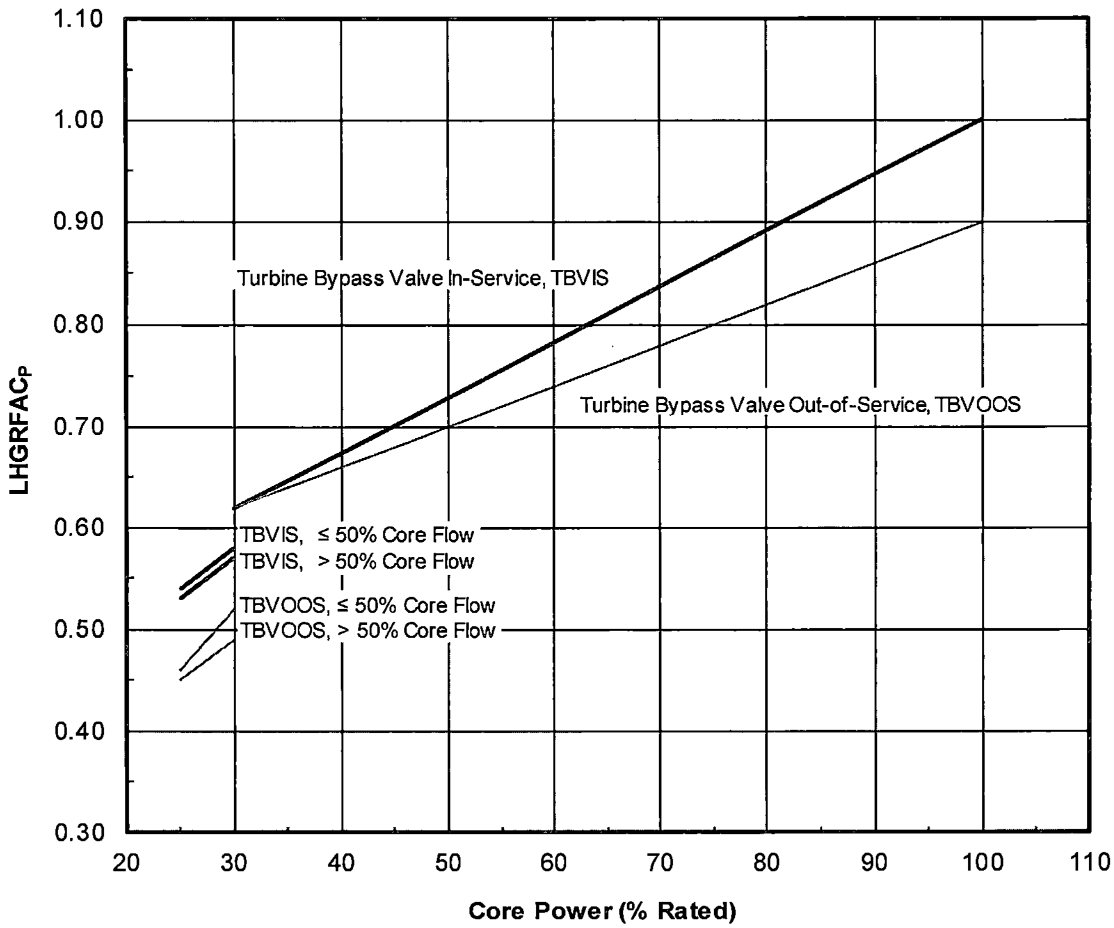
Pellet Exposure (GWd/MTU)	LHGR Limit (kW/ft)
0.0	13.4
18.9	13.4
74.4	7.1

Figure 3.1 $LHGR_{RATED}$ for ATRIUM-10 Fuel



Pellet Exposure (GWd/MTU)	LHGR Limit (kW/ft)
0.00	13.40
16.00	13.40
63.50	8.00
70.00	5.00

Figure 3.2 $LHGR_{RATED}$ for GE14 UO_2 Fuel

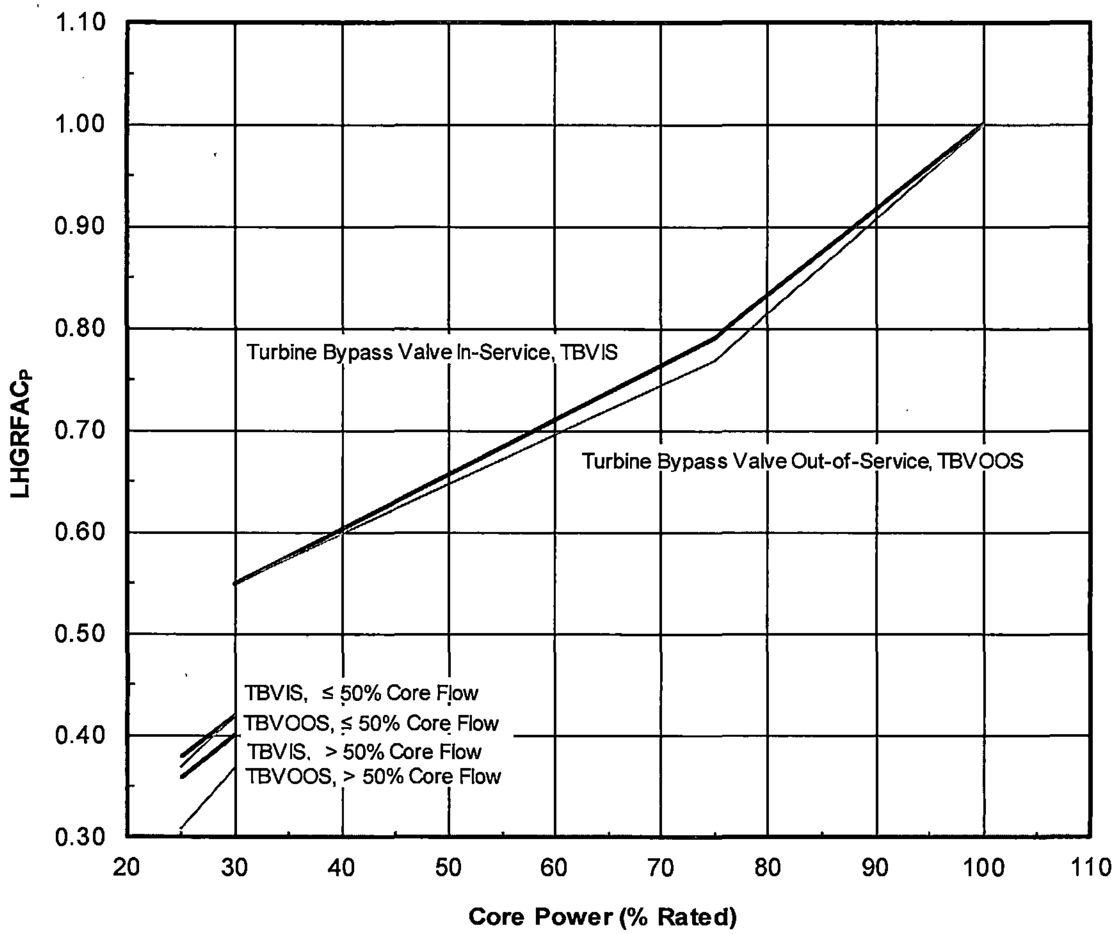


Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00
30.0	0.62
Core Flow > 50% Rated	
30.0	0.57
25.0	0.53
Core Flow ≤ 50% Rated	
30.0	0.58
25.0	0.54

Turbine Bypass Out-of-Service Core Power (% Rated)	LHGRFAC$_P$
100.0	0.90
30.0	0.62
Core Flow > 50% Rated	
30.0	0.49
25.0	0.45
Core Flow ≤ 50% Rated	
30.0	0.52
25.0	0.46

Figure 3.3 Base Operation LHGRFAC$_P$ for ATRIUM-10 Fuel

(*Independent of other EOOS conditions*)

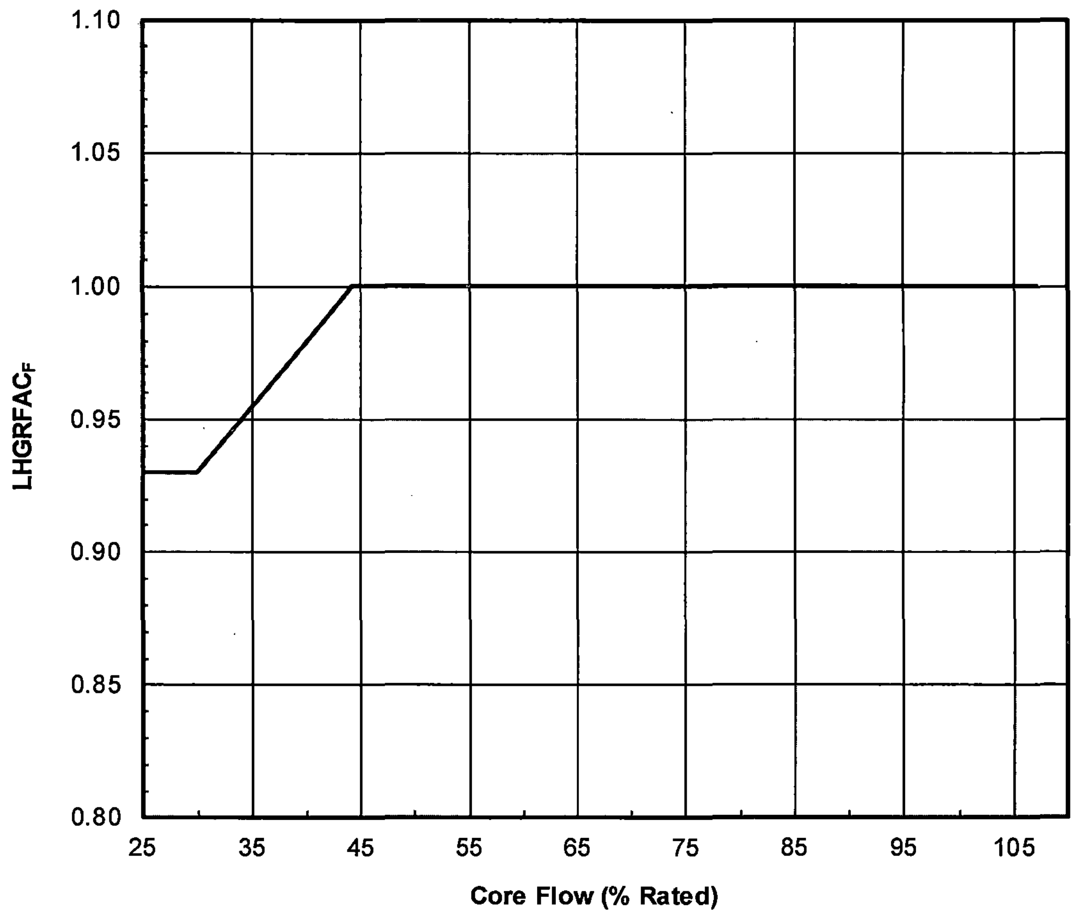


Turbine Bypass In-Service	
Core Power (% Rated)	**LHGRFAC$_P$**
100.0	1.00
75.0	0.79
30.0	0.55
Core Flow > 50% Rated	
30.0	0.40
25.0	0.36
Core Flow ≤ 50% Rated	
30.0	0.42
25.0	0.38

Turbine Bypass Out-of-Service	
Core Power (% Rated)	**LHGRFAC$_P$**
100.0	1.00
75.0	0.77
30.0	0.55
Core Flow > 50% Rated	
30.0	0.37
25.0	0.31
Core Flow ≤ 50% Rated	
30.0	0.42
25.0	0.37

Figure 3.4 Base Operation LHGRFAC$_P$ for GE 14 Fuel

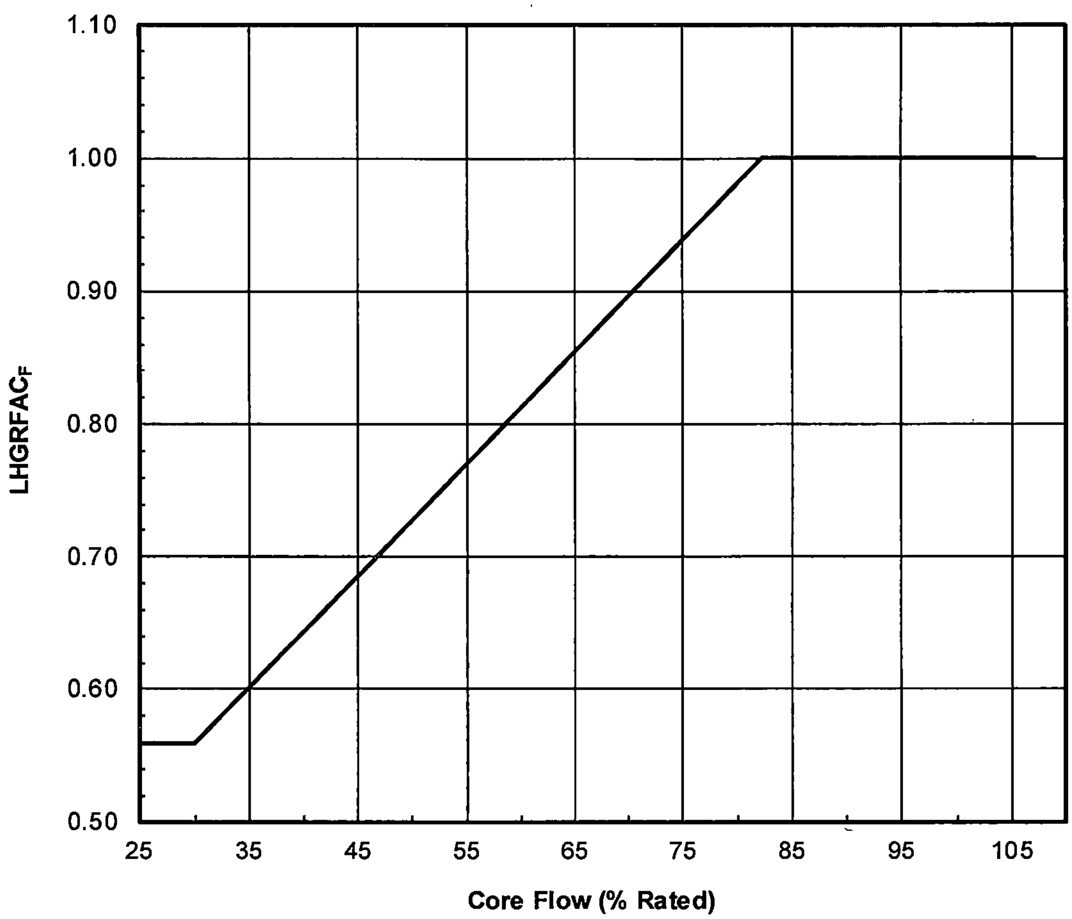
(*Independent of other EOOS conditions*)



Core Flow (% Rated)	$LHGRFAC_F$
0.0	0.93
30.0	0.93
44.2	1
107.0	1

Figure 3.5 $LHGRFAC_F$ for ATRIUM-10 Fuel
(*Values bound all EOOS conditions*)

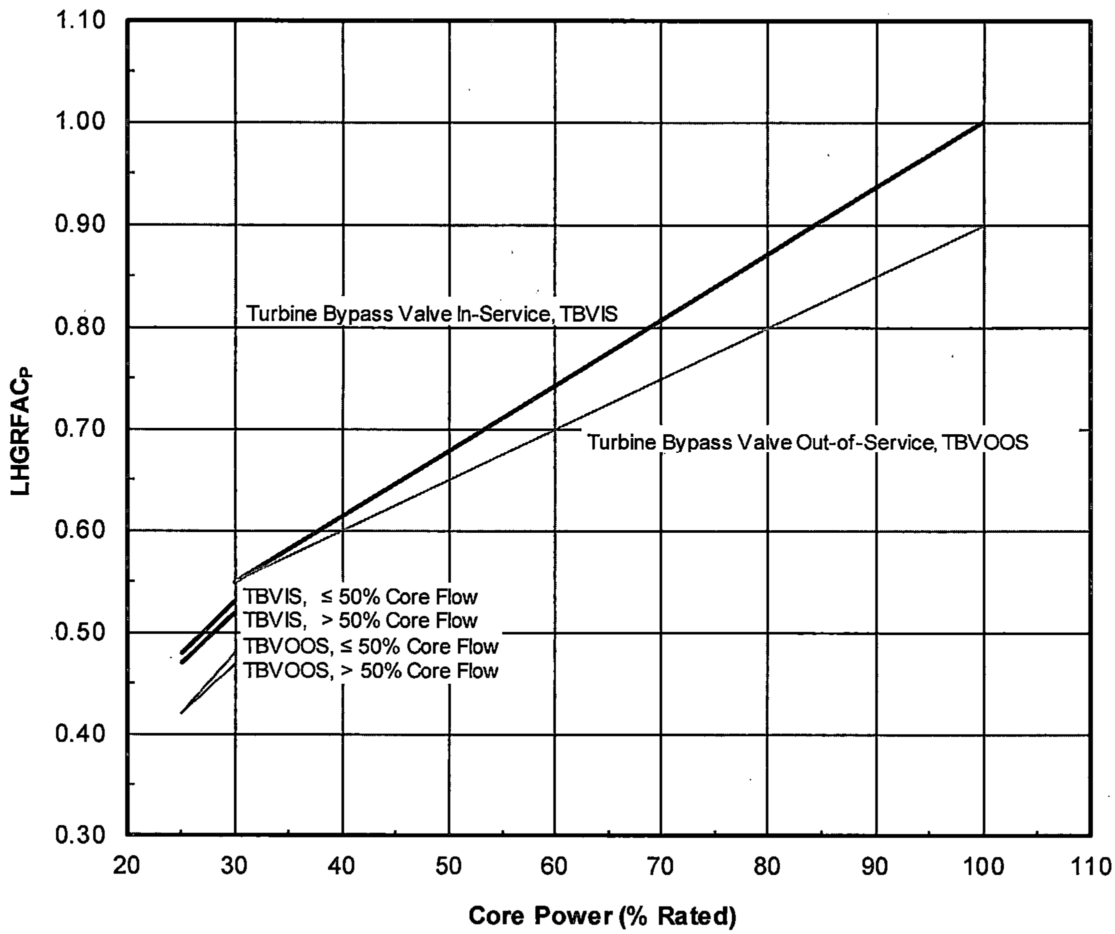
(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)



Core Flow (% Rated)	$LHGRFAC_F$
0.0	0.56
30.0	0.56
82.2	1
107.0	1

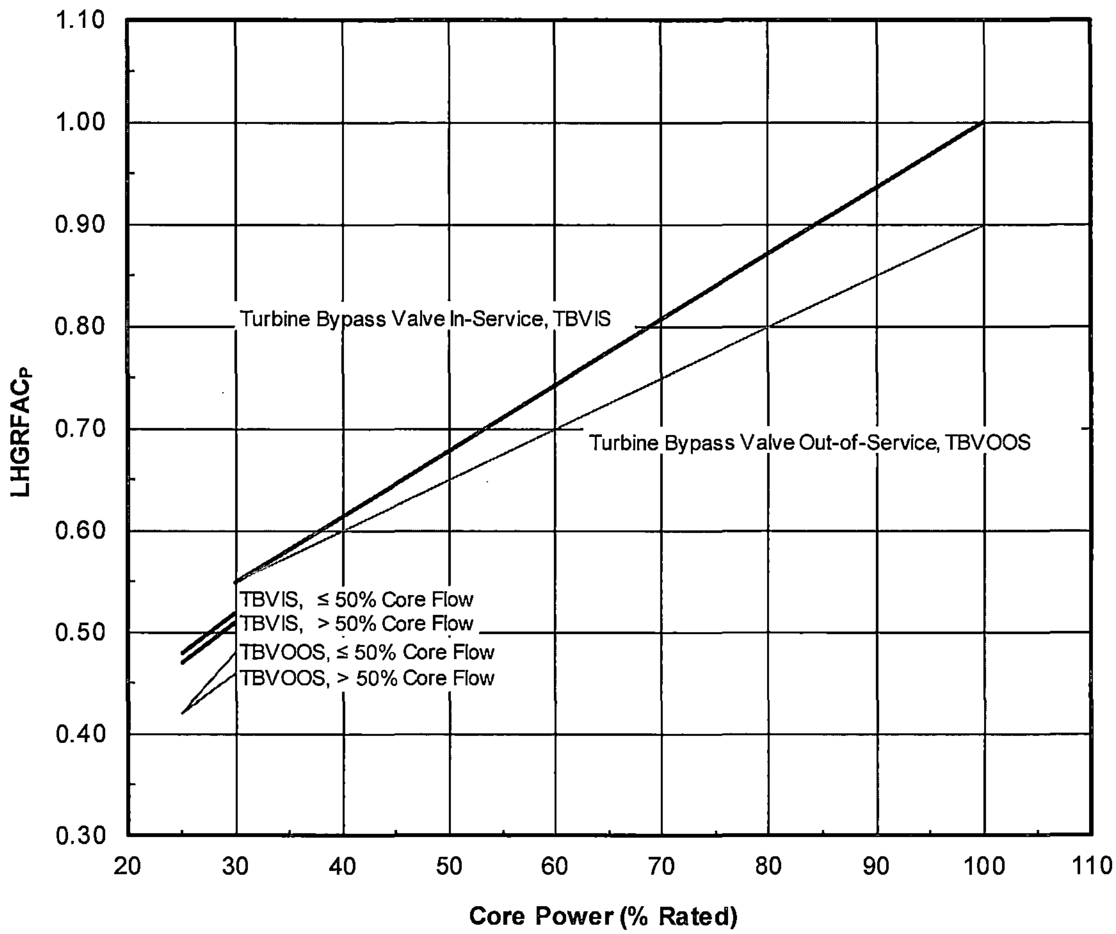
Figure 3.6 $LHGRFAC_F$ for GE 14 Fuel
(*Values bound all EOOS conditions*)

(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)



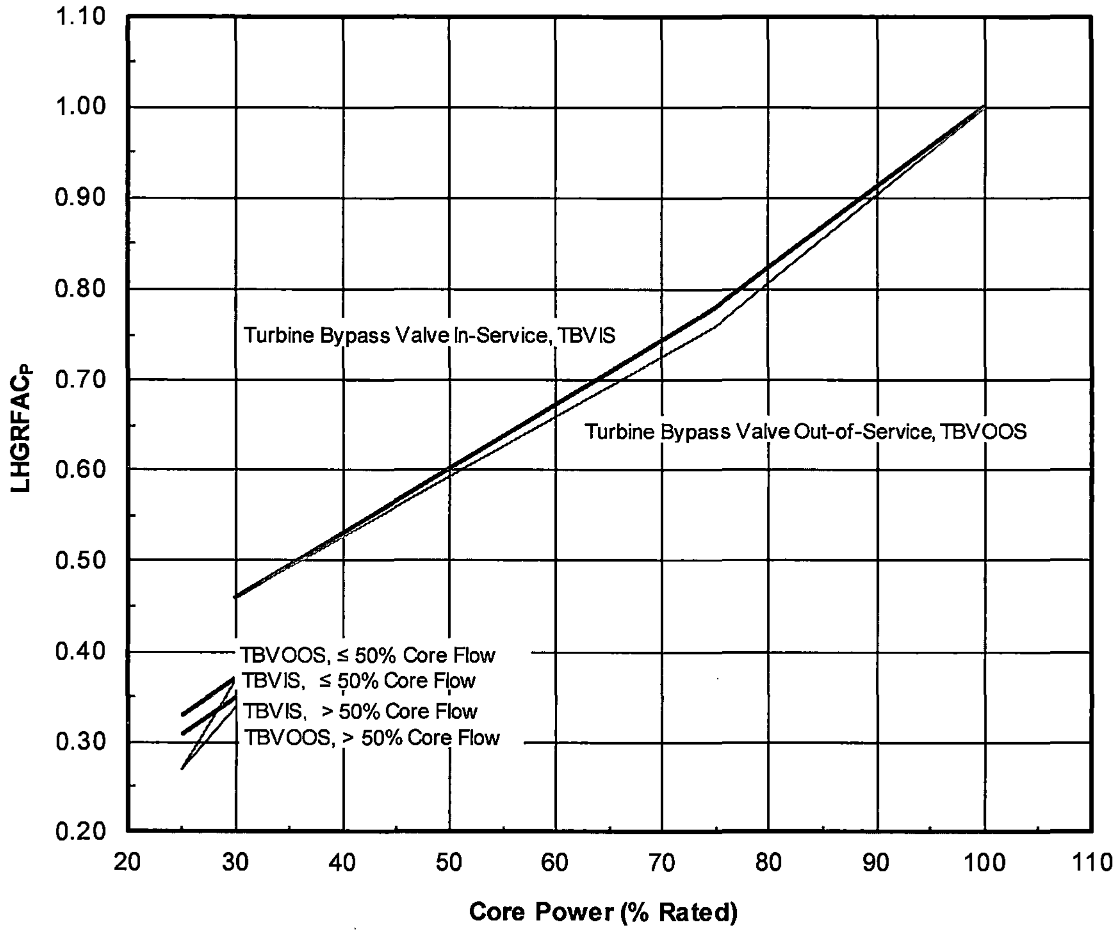
Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$	*Turbine Bypass Out-of-Service* Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00	100.0	0.90
30.0	0.55	30.0	0.55
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.52	30.0	0.47
25.0	0.47	25.0	0.42
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.53	30.0	0.48
25.0	0.48	25.0	0.42

Figure 3.7 Startup Operation LHGRFAC$_P$ for ATRIUM-10 Fuel:
Table 3.1 Temperature Range 1
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



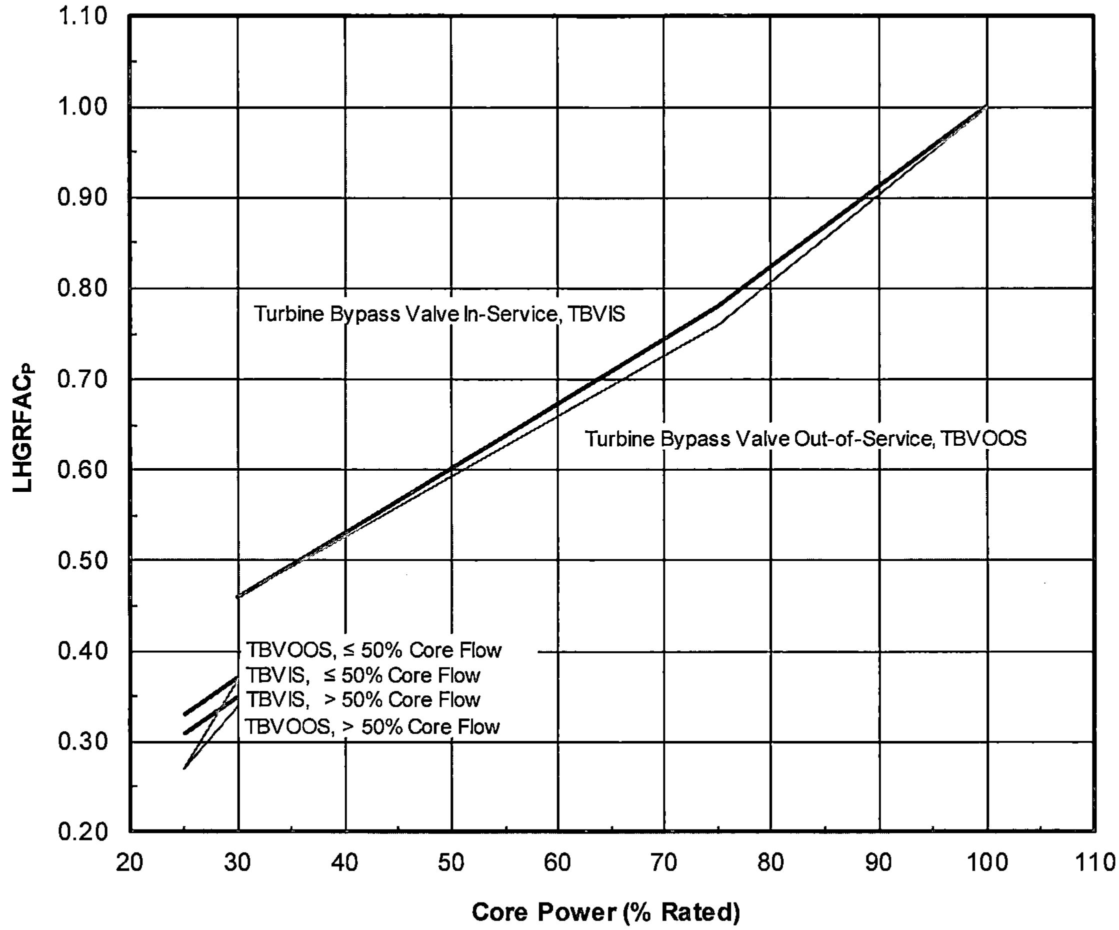
Turbine Bypass In-Service Core Power (% Rated)	$LHGRFAC_P$	*Turbine Bypass Out-of-Service* Core Power (% Rated)	$LHGRFAC_P$
100.0	1.00	100.0	0.90
30.0	0.55	30.0	0.55
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.51	30.0	0.46
25.0	0.47	25.0	0.42
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.52	30.0	0.48
25.0	0.48	25.0	0.42

Figure 3.8 Startup Operation $LHGRFAC_P$ for ATRIUM-10 Fuel:
Table 3.1 Temperature Range 2
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



Turbine Bypass In-Service Core Power	LHGRFAC$_P$	*Turbine Bypass Out-of-Service* Core Power	LHGRFAC$_P$
(% Rated)		**(% Rated)**	
100.0	1.00	100.0	1.00
75.0	0.78	75.0	0.76
30.0	0.46	30.0	0.46
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.35	30.0	0.34
25.0	0.31	25.0	0.27
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.37	30.0	0.37
25.0	0.33	25.0	0.27

Figure 3.9 Startup Operation LHGRFAC$_P$ for GE 14 Fuel:
Table 3.1 Temperature Range 1
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



Turbine Bypass In-Service		*Turbine Bypass Out-of-Service*	
Core Power (% Rated)	**$LHGRFAC_P$**	**Core Power (% Rated)**	**$LHGRFAC_P$**
100.0	1.00	100.0	1.00
75.0	0.78	75.0	0.76
30.0	0.46	30.0	0.46
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.35	30.0	0.34
25.0	0.31	25.0	0.27
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.37	30.0	0.37
25.0	0.33	25.0	0.27

Figure 3.10 Startup Operation $LHGRFAC_P$ for GE 14 Fuel:
Table 3.1 Temperature Range 2
(no Feedwater heating during startup)
(Limits valid at and below 50% power)

4 OLMCPR Limits

(Technical Specification 3.2.2, 3.3.4.1, & 3.7.5)

OLMCPR is calculated to be the most limiting of the flow or power dependent values

OLMCPR limit = MAX ($MCPR_F$, $MCPR_P$)

where:

$MCPR_F$ core flow-dependent MCPR limit
$MCPR_P$ power-dependent MCPR limit

4.1 Flow Dependent MCPR Limit: $MCPR_F$

$MCPR_F$ limits are dependent upon core flow (% of Rated), and the max core flow limit, (Rated or Increased Core Flow, ICF). $MCPR_F$ limits are shown in Figure 4.1, per Reference 1. Limits are valid for all EOOS combinations. No adjustment is required for SLO conditions.

4.2 Power Dependent MCPR Limit: $MCPR_P$

$MCPR_P$ limits are dependent upon:

- Core Power Level (% of Rated)
- Technical Specification Scram Speed (TSSS), Nominal Scram Speed (NSS), or Optimum Scram Speed (OSS)
- Cycle Operating Exposure (NEOC, EOC, and CD - as defined in this section)
- Equipment Out-Of-Service Options
- Two or Single recirculation Loop Operation (TLO vs. SLO)

The $MCPR_P$ limits are provided in the following tables, where each table contains the limits for all fuel types and EOOS options (for a specified scram speed and exposure range). The CMSS determines $MCPR_P$ limits, from these tables, based on linear interpolation between the specified powers.

4.2.1 <u>Startup without Feedwater Heaters</u>

There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. Additional power dependent limits are shown in Table 4.8, thru Table 4.15, based on temperature conditions identified in Table 3.1.

4.2.2 *Scram Speed Dependent Limits (TSSS vs. NSS vs. OSS)*

$MCPR_P$ limits are provided for three different sets of assumed scram speeds. The Technical Specification Scram Speed (TSSS) $MCPR_P$ limits are applicable at all times, as long as the scram time surveillance demonstrates the times in Technical Specification Table 3.1.4-1 are met. Both Nominal Scram Speeds (NSS) and/or Optimum Scram Speeds (OSS) may be used, as long as the scram time surveillance demonstrates Table 4.1 times are applicable.*†

Table 4.1 Nominal Scram Time Basis

Notch Position (index)	Nominal Scram Timing (seconds)	Optimum Scram Timing (seconds)
46	0.420	0.380
36	0.980	0.875
26	1.600	1.465
6	2.900	2.900

In demonstrating compliance with the NSS and/or OSS scram time basis, surveillance requirements from Technical Specification 3.1.4 apply; accepting the definition of SLOW rods should conform to scram speeds shown in Table 4.1. If conformance is not demonstrated, TSSS based $MCPR_P$ limits are applied.

On initial cycle startup, TSSS limits are used until the successful completion of scram timing confirms NSS and/or OSS based limits are applicable.

4.2.3 *Exposure Dependent Limits*

Exposures are tracked on a Core Average Exposure basis (CAVEX, not Cycle Exposure). Higher exposure $MCPR_P$ limits are always more limiting and may be used for any Core Average Exposure up to the ending exposure. Per Reference 1, $MCPR_P$ limits are provided for the following exposure ranges:

BOC to NEOC	NEOC corresponds to	**28,986.6 MWd / MTU**
BOC to EOCLB	EOCLB corresponds to	**32,248.5 MWd / MTU**
BOC to End of Coast	End of Coast	**33,643.5 MWd / MTU**

NEOC refers to a Near EOC exposure point.

* Reference 1 analysis results are based on information identified in Reference 5.

† Drop out times consistent with method used to perform actual timing measurements (i.e., including pickup/dropout effects).

The EOCLB exposure point is not the true End-Of-Cycle exposure. Instead it corresponds to a licensing exposure window exceeding expected end-of-full-power-life.

The End of Coast exposure point represents a licensing exposure point exceeding the expected end-of-cycle exposure including cycle extension options.

4.2.4 Equipment Out-Of-Service (EOOS) Options

EOOS options* covered by $MCPR_P$ limits are given by the following:

In-Service	All equipment In-Service
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
RPTOOS+TBVOOS	Combined RPTOOS and TBVOOS
PLUOOS	Power Load Unbalance Out-Of-Service
PLUOOS+RPTOOS	Combined PLUOOS and RPTOOS
PLUOOS+TBVOOS	Combined PLUOOS and TBVOOS
PLUOOS+TBVOOS+RPTOOS	Combined PLUOOS, RPTOOS, and TBVOOS
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service (or Final Feedwater Temperature Reduction)
RCPOOS	One Recirculation Pump Out-Of-Service

For exposure ranges up to NEOC and EOCLB, additional combinations of $MCPR_P$ limits are also provided including FHOOS. The coast down exposure range assumes application of FFWTR. FHOOS based $MCPR_P$ limits for the coast down exposure are redundant because the temperature setdown assumption is identical with FFWTR.

4.2.5 Single-Loop-Operation (SLO) Limits

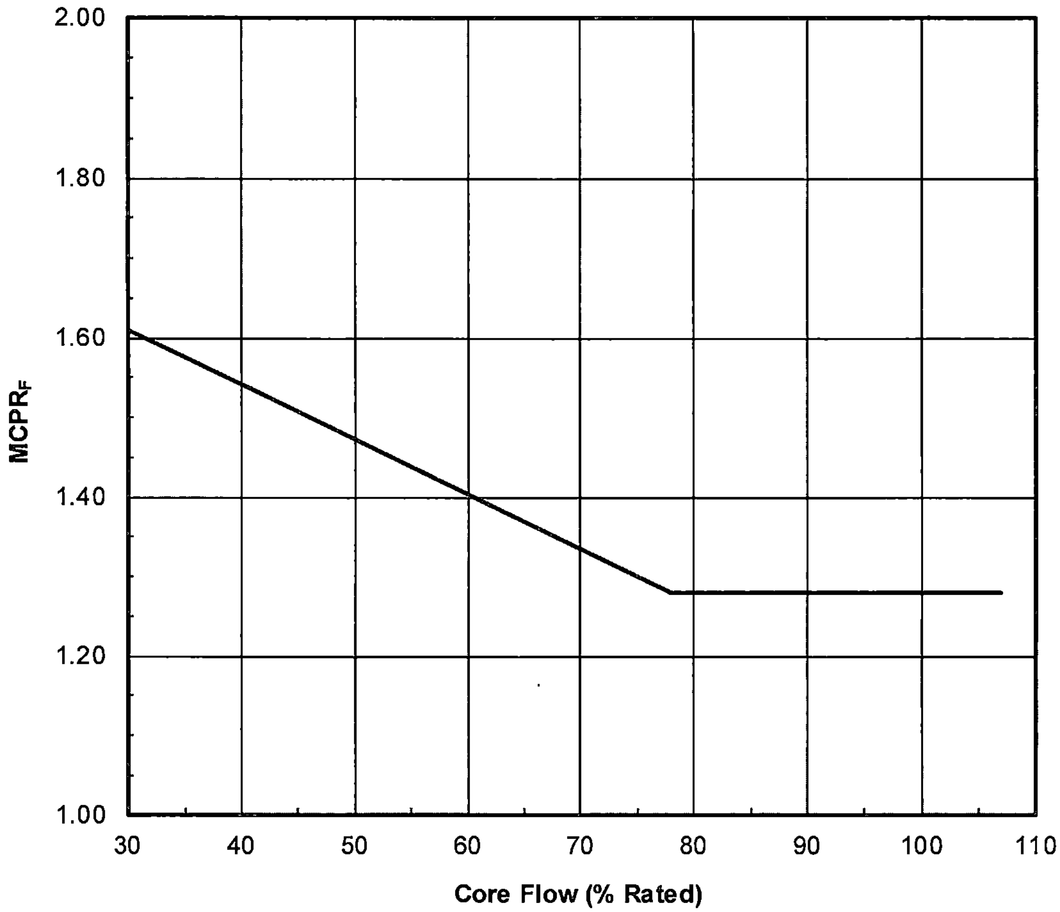
When operating in RCPOOS conditions, MCPRp limits are constructed differently from the normal operating RCP conditions. The limiting event for RCPOOS is a pump seizure scenario, which sets the upper bound for allowed core power and flow†. This event is not impacted by scram time assumptions. Specific $MCPR_P$ limits are shown in Table 4.16 and Table 4.17.

4.2.6 Below Pbypass Limits

Below Pbypass (30% rated power), $MCPR_P$ limits depend upon core flow. One set of $MCPR_P$ limits applies for core flow above 50% of rated; a second set applies if the core flow is less than or equal to 50% rated.

* All equipment service conditions assume 1 SRVOOS.

† RCPOOS limits are only valid up to 50% rated core power, 50% rated core flow, and an active recirculation drive flow of 17.73 Mlb_m/hr.



Core Flow	$MCPR_F$
(% Rated)	
30.0	1.61
78.0	1.28
107.0	1.28

Figure 4.1 $MCPR_F$ for GE 14 and ATRIUM-10 Fuel
(*Values bound all EOOS conditions*)

(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)

Table 4.2 $MCPR_P$ Limits for ATRIUM-10: Optimum Scram Time Basis*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
Base Case	100	1.45	1.46	1.48
	75	1.58	1.59	1.60
	65	1.66	1.66	1.70
	50	1.80	1.80	---
	50	1.83	1.83	1.86
	40	1.99	1.99	2.09
	30	2.29	2.29	2.40
	30 at > 50%F	2.34	2.34	2.43
	25 at > 50%F	2.56	2.56	2.66
	30 at ≤ 50%F	2.33	2.33	2.40
	25 at ≤ 50%F	2.50	2.50	2.60
FHOOS	100	1.47	1.48	---
	75	1.60	1.60	---
	65	1.70	1.70	---
	50	---	---	---
	50	1.86	1.86	---
	40	2.09	2.09	---
	30	2.40	2.40	---
	30 at > 50%F	2.43	2.43	---
	25 at > 50%F	2.66	2.66	---
	30 at ≤ 50%F	2.40	2.40	---
	25 at ≤ 50%F	2.60	2.60	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR/FHOOS is supported for the BOC to End of Coast limits.

Table 4.3 MCPR$_P$ Limits for GE-14: Optimum Scram Time Basis*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
Base Case	100	1.47	1.51	1.52
	75	1.60	1.64	1.65
	65	1.70	1.71	1.75
	50	1.85	1.85	---
	50	1.86	1.86	1.91
	40	2.04	2.04	2.14
	30	2.36	2.36	2.48
	30 at > 50%F	2.42	2.42	2.52
	25 at > 50%F	2.64	2.64	2.76
	30 at ≤ 50%F	2.41	2.41	2.48
	25 at ≤ 50%F	2.58	2.58	2.68
FHOOS	100	1.49	1.52	---
	75	1.63	1.65	---
	65	1.73	1.75	---
	50	---	---	---
	50	1.91	1.91	---
	40	2.14	2.14	---
	30	2.48	2.48	---
	30 at > 50%F	2.52	2.52	---
	25 at > 50%F	2.76	2.76	---
	30 at ≤ 50%F	2.48	2.48	---
	25 at ≤ 50%F	2.68	2.68	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.
FFWTR/FHOOS is supported for the BOC to End of Coast limits.

Table 4.4 MCPR$_P$ Limits for ATRIUM-10: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
Base Case	100	1.46	1.47	1.49
	75	1.59	1.60	1.62
	65	1.67	1.67	1.71
	50	1.82	1.82	---
	50	1.84	1.84	1.90
	40	2.03	2.03	2.12
	30	2.33	2.33	2.45
	30 at > 50%F	2.34	2.34	2.45
	25 at > 50%F	2.56	2.56	2.66
	30 at ≤ 50%F	2.33	2.33	2.45
	25 at ≤ 50%F	2.50	2.50	2.60
TBVOOS	100	1.49	1.51	1.52
	75	1.62	1.62	1.66
	65	1.71	1.71	1.74
	50	---	---	---
	50	1.84	1.84	1.90
	40	2.03	2.03	2.12
	30	2.33	2.33	2.45
	30 at > 50%F	2.90	2.90	3.00
	25 at > 50%F	3.23	3.23	3.32
	30 at ≤ 50%F	2.74	2.74	2.85
	25 at ≤ 50%F	3.10	3.10	3.24
FHOOS	100	1.49	1.49	---
	75	1.62	1.62	---
	65	1.71	1.71	---
	50	---	---	---
	50	1.90	1.90	---
	40	2.12	2.12	---
	30	2.45	2.45	---
	30 at > 50%F	2.45	2.45	---
	25 at > 50%F	2.66	2.66	---
	30 at ≤ 50%F	2.45	2.45	---
	25 at ≤ 50%F	2.60	2.60	---
PLUOOS	100	1.46	1.47	1.49
	75	1.59	1.60	1.62
	65	1.75	1.77	1.77
	50	---	---	---
	50	1.84	1.84	1.90
	40	2.03	2.03	2.12
	30	2.33	2.33	2.45
	30 at > 50%F	2.34	2.34	2.45
	25 at > 50%F	2.56	2.56	2.66
	30 at ≤ 50%F	2.33	2.33	2.45
	25 at ≤ 50%F	2.50	2.50	2.60

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.4 MCPR$_P$ Limits for ATRIUM-10: Nominal Scram Time Basis (*continued*)*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVOOS FHOOS	100	1.51	1.52	---
	75	1.66	1.66	---
	65	1.74	1.74	---
	50	---	---	---
	50	1.90	1.90	---
	40	2.12	2.12	---
	30	2.45	2.45	---
	30 at > 50%F	3.00	3.00	---
	25 at > 50%F	3.32	3.32	---
	30 at ≤ 50%F	2.85	2.85	---
	25 at ≤ 50%F	3.24	3.24	---
TBVOOS PLUOOS	100	1.49	1.51	1.52
	75	1.62	1.62	1.66
	65	1.75	1.77	1.77
	50	---	---	---
	50	1.84	1.84	1.90
	40	2.03	2.03	2.12
	30	2.33	2.33	2.45
	30 at > 50%F	2.90	2.90	3.00
	25 at > 50%F	3.23	3.23	3.32
	30 at ≤ 50%F	2.74	2.74	2.85
	25 at ≤ 50%F	3.10	3.10	3.24
FHOOS PLUOOS	100	1.49	1.49	---
	75	1.62	1.62	---
	65	1.75	1.77	---
	50	---	---	---
	50	1.90	1.90	---
	40	2.12	2.12	---
	30	2.45	2.45	---
	30 at > 50%F	2.45	2.45	---
	25 at > 50%F	2.66	2.66	---
	30 at ≤ 50%F	2.45	2.45	---
	25 at ≤ 50%F	2.60	2.60	---
TBVOOS FHOOS PLUOOS	100	1.51	1.52	---
	75	1.66	1.66	---
	65	1.75	1.77	---
	50	---	---	---
	50	1.90	1.90	---
	40	2.12	2.12	---
	30	2.45	2.45	---
	30 at > 50%F	3.00	3.00	---
	25 at > 50%F	3.32	3.32	---
	30 at ≤ 50%F	2.85	2.85	---
	25 at ≤ 50%F	3.24	3.24	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.5 MCPR$_P$ Limits for GE 14: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
Base Case	100	1.49	1.52	1.53
	75	1.62	1.65	1.66
	65	1.71	1.72	1.76
	50	---	---	---
	50	1.87	1.87	1.94
	40	2.08	2.08	2.18
	30	2.40	2.40	2.52
	30 at > 50%F	2.42	2.42	2.52
	25 at > 50%F	2.64	2.64	2.76
	30 at ≤ 50%F	2.41	2.41	2.52
	25 at ≤ 50%F	2.58	2.58	2.68
TBVOOS	100	1.52	1.56	1.57
	75	1.65	1.69	1.70
	65	1.75	1.77	1.80
	50	---	---	---
	50	1.90	1.90	1.96
	40	2.09	2.09	2.18
	30	2.40	2.40	2.52
	30 at > 50%F	2.96	2.96	3.06
	25 at > 50%F	3.28	3.28	3.36
	30 at ≤ 50%F	2.81	2.81	2.93
	25 at ≤ 50%F	3.19	3.19	3.33
FHOOS	100	1.51	1.53	---
	75	1.66	1.66	---
	65	1.75	1.76	---
	50	---	---	---
	50	1.94	1.94	---
	40	2.18	2.18	---
	30	2.52	2.52	---
	30 at > 50%F	2.52	2.52	---
	25 at > 50%F	2.76	2.76	---
	30 at ≤ 50%F	2.52	2.52	---
	25 at ≤ 50%F	2.68	2.68	---
PLUOOS	100	1.49	1.52	1.53
	75	1.62	1.65	1.66
	65	1.78	1.78	1.78
	50	---	---	---
	50	1.87	1.87	1.94
	40	2.08	2.08	2.18
	30	2.40	2.40	2.52
	30 at > 50%F	2.42	2.42	2.52
	25 at > 50%F	2.64	2.64	2.76
	30 at ≤ 50%F	2.41	2.41	2.52
	25 at ≤ 50%F	2.58	2.58	2.68

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.5 MCPR$_P$ Limits for GE 14: Nominal Scram Time Basis (*continued*)*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVOOS FHOOS	100	1.54	1.57	---
	75	1.69	1.70	---
	65	1.78	1.80	---
	50	---	---	---
	50	1.96	1.96	---
	40	2.18	2.18	---
	30	2.52	2.52	---
	30 at > 50%F	3.06	3.06	---
	25 at > 50%F	3.36	3.36	---
	30 at ≤ 50%F	2.93	2.93	---
	25 at ≤ 50%F	3.33	3.33	---
TBVOOS PLUOOS	100	1.52	1.56	1.57
	75	1.65	1.69	1.70
	65	1.78	1.78	1.80
	50	---	---	---
	50	1.90	1.90	1.96
	40	2.09	2.09	2.18
	30	2.40	2.40	2.52
	30 at > 50%F	2.96	2.96	3.06
	25 at > 50%F	3.28	3.28	3.36
	30 at ≤ 50%F	2.81	2.81	2.93
	25 at ≤ 50%F	3.19	3.19	3.33
FHOOS PLUOOS	100	1.51	1.53	---
	75	1.66	1.66	---
	65	1.78	1.78	---
	50	---	---	---
	50	1.94	1.94	---
	40	2.18	2.18	---
	30	2.52	2.52	---
	30 at > 50%F	2.52	2.52	---
	25 at > 50%F	2.76	2.76	---
	30 at ≤ 50%F	2.52	2.52	---
	25 at ≤ 50%F	2.68	2.68	---
TBVOOS FHOOS PLUOOS	100	1.54	1.57	---
	75	1.69	1.70	---
	65	1.78	1.80	---
	50	---	---	---
	50	1.96	1.96	---
	40	2.18	2.18	---
	30	2.52	2.52	---
	30 at > 50%F	3.06	3.06	---
	25 at > 50%F	3.36	3.36	---
	30 at ≤ 50%F	2.93	2.93	---
	25 at ≤ 50%F	3.33	3.33	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.6 MCPRP Limits for ATRIUM-10: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
Base Case	100	1.49	1.49	1.51
	75	1.62	1.62	1.64
	65	1.69	1.69	1.72
	50	---	---	---
	50	1.85	1.85	1.93
	40	2.06	2.06	2.16
	30	2.36	2.36	2.49
	30 at > 50%F	2.36	2.36	2.49
	25 at > 50%F	2.56	2.56	2.66
	30 at ≤ 50%F	2.36	2.36	2.49
	25 at ≤ 50%F	2.50	2.50	2.60
TBVOOS	100	1.52	1.52	1.54
	75	1.65	1.65	1.67
	65	1.72	1.72	1.75
	50	---	---	---
	50	1.86	1.86	1.94
	40	2.07	2.07	2.16
	30	2.36	2.36	2.49
	30 at > 50%F	2.90	2.90	3.00
	25 at > 50%F	3.23	3.23	3.32
	30 at ≤ 50%F	2.74	2.74	2.85
	25 at ≤ 50%F	3.10	3.10	3.24
FHOOS	100	1.51	1.51	---
	75	1.64	1.64	---
	65	1.72	1.72	---
	50	---	---	---
	50	1.93	1.93	---
	40	2.16	2.16	---
	30	2.49	2.49	---
	30 at > 50%F	2.49	2.49	---
	25 at > 50%F	2.66	2.66	---
	30 at ≤ 50%F	2.49	2.49	---
	25 at ≤ 50%F	2.60	2.60	---
PLUOOS	100	1.49	1.49	1.51
	75	1.62	1.62	1.64
	65	1.76	1.79	1.79
	50	---	---	---
	50	1.85	1.85	1.93
	40	2.06	2.06	2.16
	30	2.36	2.36	2.49
	30 at > 50%F	2.36	2.36	2.49
	25 at > 50%F	2.56	2.56	2.66
	30 at ≤ 50%F	2.36	2.36	2.49
	25 at ≤ 50%F	2.50	2.50	2.60

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.6 MCPRP Limits for ATRIUM-10: Technical Specification Scram Time Basis (*continued*)*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVOOS FHOOS	100	1.54	1.54	---
	75	1.67	1.67	---
	65	1.75	1.75	---
	50	---	---	---
	50	1.94	1.94	---
	40	2.16	2.16	---
	30	2.49	2.49	---
	30 at > 50%F	3.00	3.00	---
	25 at > 50%F	3.32	3.32	---
	30 at ≤ 50%F	2.85	2.85	---
	25 at ≤ 50%F	3.24	3.24	---
TBVOOS PLUOOS	100	1.52	1.52	1.54
	75	1.65	1.65	1.67
	65	1.76	1.79	1.79
	50	---	---	---
	50	1.86	1.86	1.94
	40	2.07	2.07	2.16
	30	2.36	2.36	2.49
	30 at > 50%F	2.90	2.90	3.00
	25 at > 50%F	3.23	3.23	3.32
	30 at ≤ 50%F	2.74	2.74	2.85
	25 at ≤ 50%F	3.10	3.10	3.24
FHOOS PLUOOS	100	1.51	1.51	---
	75	1.64	1.64	---
	65	1.76	1.79	---
	50	---	---	---
	50	1.93	1.93	---
	40	2.16	2.16	---
	30	2.49	2.49	---
	30 at > 50%F	2.49	2.49	---
	25 at > 50%F	2.66	2.66	---
	30 at ≤ 50%F	2.49	2.49	---
	25 at ≤ 50%F	2.60	2.60	---
TBVOOS FHOOS PLUOOS	100	1.54	1.54	---
	75	1.67	1.67	---
	65	1.76	1.79	---
	50	---	---	---
	50	1.94	1.94	---
	40	2.16	2.16	---
	30	2.49	2.49	---
	30 at > 50%F	3.00	3.00	---
	25 at > 50%F	3.32	3.32	---
	30 at ≤ 50%F	2.85	2.85	---
	25 at ≤ 50%F	3.24	3.24	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.7 MCPR$_P$ Limits for GE 14: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
Base Case	100	1.51	1.53	1.54
	75	1.66	1.66	1.67
	65	1.73	1.73	1.76
	50	---	---	---
	50	1.90	1.90	1.98
	40	2.12	2.12	2.22
	30	2.45	2.45	2.57
	30 at > 50%F	2.45	2.45	2.57
	25 at > 50%F	2.64	2.64	2.76
	30 at ≤ 50%F	2.45	2.45	2.57
	25 at ≤ 50%F	2.58	2.58	2.68
TBVOOS	100	1.54	1.57	1.57
	75	1.69	1.70	1.70
	65	1.77	1.77	1.81
	50	---	---	---
	50	1.92	1.92	1.99
	40	2.13	2.13	2.22
	30	2.45	2.45	2.57
	30 at > 50%F	2.96	2.96	3.06
	25 at > 50%F	3.28	3.28	3.36
	30 at ≤ 50%F	2.81	2.81	2.93
	25 at ≤ 50%F	3.19	3.19	3.33
FHOOS	100	1.54	1.54	---
	75	1.67	1.67	---
	65	1.76	1.76	---
	50	---	---	---
	50	1.98	1.98	---
	40	2.22	2.22	---
	30	2.57	2.57	---
	30 at > 50%F	2.57	2.57	---
	25 at > 50%F	2.76	2.76	---
	30 at ≤ 50%F	2.57	2.57	---
	25 at ≤ 50%F	2.68	2.68	---
PLUOOS	100	1.51	1.53	1.54
	75	1.66	1.66	1.67
	65	1.79	1.80	1.80
	50	---	---	---
	50	1.90	1.90	1.98
	40	2.12	2.12	2.22
	30	2.45	2.45	2.57
	30 at > 50%F	2.45	2.45	2.57
	25 at > 50%F	2.64	2.64	2.76
	30 at ≤ 50%F	2.45	2.45	2.57
	25 at ≤ 50%F	2.58	2.58	2.68

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.7 MCPR$_P$ Limits for GE 14: Technical Specification Scram Time Basis (*continued*)*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVOOS FHOOS	100	1.57	1.57	---
	75	1.70	1.70	---
	65	1.80	1.81	---
	50	---	---	---
	50	1.99	1.99	---
	40	2.22	2.22	---
	30	2.57	2.57	---
	30 at > 50%F	3.06	3.06	---
	25 at > 50%F	3.36	3.36	---
	30 at ≤ 50%F	2.93	2.93	---
	25 at ≤ 50%F	3.33	3.33	---
TBVOOS PLUOOS	100	1.54	1.57	1.57
	75	1.69	1.70	1.70
	65	1.79	1.80	1.81
	50	---	---	---
	50	1.92	1.92	1.99
	40	2.13	2.13	2.22
	30	2.45	2.45	2.57
	30 at > 50%F	2.96	2.96	3.06
	25 at > 50%F	3.28	3.28	3.36
	30 at ≤ 50%F	2.81	2.81	2.93
	25 at ≤ 50%F	3.19	3.19	3.33
FHOOS PLUOOS	100	1.54	1.54	---
	75	1.67	1.67	---
	65	1.79	1.80	---
	50	---	---	---
	50	1.98	1.98	---
	40	2.22	2.22	---
	30	2.57	2.57	---
	30 at > 50%F	2.57	2.57	---
	25 at > 50%F	2.76	2.76	---
	30 at ≤ 50%F	2.57	2.57	---
	25 at ≤ 50%F	2.68	2.68	---
TBVOOS FHOOS PLUOOS	100	1.57	1.57	---
	75	1.70	1.70	---
	65	1.80	1.81	---
	50	---	---	---
	50	1.99	1.99	---
	40	2.22	2.22	---
	30	2.57	2.57	---
	30 at > 50%F	3.06	3.06	---
	25 at > 50%F	3.36	3.36	---
	30 at ≤ 50%F	2.93	2.93	---
	25 at ≤ 50%F	3.33	3.33	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.8 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 1 for ATRIUM-10: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVIS	100	1.49	1.49	1.49
	75	1.62	1.62	1.62
	65	1.75	1.77	1.77
	50	---	---	---
	50	2.06	2.06	2.06
	40	2.35	2.35	2.35
	30	2.74	2.74	2.74
	30 at > 50%F	2.74	2.74	2.74
	25 at > 50%F	2.92	2.92	2.92
	30 at ≤ 50%F	2.74	2.74	2.74
	25 at ≤ 50%F	2.85	2.85	2.85
TBVOOS	100	1.51	1.52	1.52
	75	1.66	1.66	1.66
	65	1.75	1.77	1.77
	50	---	---	---
	50	2.06	2.06	2.06
	40	2.35	2.35	2.35
	30	2.74	2.74	2.74
	30 at > 50%F	3.15	3.15	3.15
	25 at > 50%F	3.48	3.48	3.48
	30 at ≤ 50%F	3.03	3.03	3.03
	25 at ≤ 50%F	3.46	3.46	3.46

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.9 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 1 for GE 14: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVIS	100	1.51	1.53	1.53
	75	1.66	1.66	1.66
	65	1.78	1.78	1.78
	50	---	---	---
	50	2.13	2.13	2.13
	40	2.42	2.42	2.42
	30	2.83	2.83	2.83
	30 at > 50%F	2.83	2.83	2.83
	25 at > 50%F	3.03	3.03	3.03
	30 at ≤ 50%F	2.83	2.83	2.83
	25 at ≤ 50%F	2.95	2.95	2.95
TBVOOS	100	1.54	1.57	1.57
	75	1.69	1.70	1.70
	65	1.78	1.80	1.80
	50	---	---	---
	50	2.13	2.13	2.13
	40	2.42	2.42	2.42
	30	2.83	2.83	2.83
	30 at > 50%F	3.21	3.21	3.21
	25 at > 50%F	3.55	3.55	3.55
	30 at ≤ 50%F	3.11	3.11	3.11
	25 at ≤ 50%F	3.55	3.55	3.55

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.10 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 2 for ATRIUM-10: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVIS	100	1.49	1.49	1.49
	75	1.62	1.62	1.62
	65	1.75	1.77	1.77
	50	---	---	---
	50	2.07	2.07	2.07
	40	2.36	2.36	2.36
	30	2.76	2.76	2.76
	30 at > 50%F	2.76	2.76	2.76
	25 at > 50%F	2.98	2.98	2.98
	30 at ≤ 50%F	2.76	2.76	2.76
	25 at ≤ 50%F	2.87	2.87	2.87
TBVOOS	100	1.51	1.52	1.52
	75	1.66	1.66	1.66
	65	1.75	1.77	1.77
	50	---	---	---
	50	2.07	2.07	2.07
	40	2.36	2.36	2.36
	30	2.76	2.76	2.76
	30 at > 50%F	3.16	3.16	3.16
	25 at > 50%F	3.49	3.49	3.49
	30 at ≤ 50%F	3.04	3.04	3.04
	25 at ≤ 50%F	3.47	3.47	3.47

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.11 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 2 for GE 14: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVIS	100	1.51	1.53	1.53
	75	1.66	1.66	1.66
	65	1.78	1.78	1.78
	50	---	---	---
	50	2.14	2.14	2.14
	40	2.43	2.43	2.43
	30	2.84	2.84	2.84
	30 at > 50%F	2.84	2.84	2.84
	25 at > 50%F	3.05	3.05	3.05
	30 at ≤ 50%F	2.84	2.84	2.84
	25 at ≤ 50%F	2.96	2.96	2.96
TBVOOS	100	1.54	1.57	1.57
	75	1.69	1.70	1.70
	65	1.78	1.80	1.80
	50	---	---	---
	50	2.14	2.14	2.14
	40	2.43	2.43	2.43
	30	2.84	2.84	2.84
	30 at > 50%F	3.22	3.22	3.22
	25 at > 50%F	3.57	3.57	3.57
	30 at ≤ 50%F	3.13	3.13	3.13
	25 at ≤ 50%F	3.57	3.57	3.57

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.12 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 1 for ATRIUM-10: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVIS	100	1.51	1.51	1.51
	75	1.64	1.64	1.64
	65	1.76	1.79	1.79
	50	---	---	---
	50	2.09	2.09	2.09
	40	2.39	2.39	2.39
	30	2.79	2.79	2.79
	30 at > 50%F	2.79	2.79	2.79
	25 at > 50%F	2.92	2.92	2.92
	30 at ≤ 50%F	2.79	2.79	2.79
	25 at ≤ 50%F	2.85	2.85	2.85
TBVOOS	100	1.54	1.54	1.54
	75	1.67	1.67	1.67
	65	1.76	1.79	1.79
	50	---	---	---
	50	2.09	2.09	2.09
	40	2.39	2.39	2.39
	30	2.79	2.79	2.79
	30 at > 50%F	3.15	3.15	3.15
	25 at > 50%F	3.48	3.48	3.48
	30 at ≤ 50%F	3.03	3.03	3.03
	25 at ≤ 50%F	3.46	3.46	3.46

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.13 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 1 for GE 14: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVIS	100	1.54	1.54	1.54
	75	1.67	1.67	1.67
	65	1.79	1.80	1.80
	50	---	---	---
	50	2.17	2.17	2.17
	40	2.46	2.46	2.46
	30	2.87	2.87	2.87
	30 at > 50%F	2.87	2.87	2.87
	25 at > 50%F	3.03	3.03	3.03
	30 at ≤ 50%F	2.87	2.87	2.87
	25 at ≤ 50%F	2.95	2.95	2.95
TBVOOS	100	1.57	1.57	1.57
	75	1.70	1.70	1.70
	65	1.80	1.81	1.81
	50	---	---	---
	50	2.17	2.17	2.17
	40	2.46	2.46	2.46
	30	2.87	2.87	2.87
	30 at > 50%F	3.21	3.21	3.21
	25 at > 50%F	3.55	3.55	3.55
	30 at ≤ 50%F	3.11	3.11	3.11
	25 at ≤ 50%F	3.55	3.55	3.55

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.14 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 2 for ATRIUM-10: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVIS	100	1.51	1.51	1.51
	75	1.64	1.64	1.64
	65	1.76	1.79	1.79
	50	---	---	---
	50	2.10	2.10	2.10
	40	2.40	2.40	2.40
	30	2.80	2.80	2.80
	30 at > 50%F	2.80	2.80	2.80
	25 at > 50%F	2.98	2.98	2.98
	30 at ≤ 50%F	2.80	2.80	2.80
	25 at ≤ 50%F	2.87	2.87	2.87
TBVOOS	100	1.54	1.54	1.54
	75	1.67	1.67	1.67
	65	1.76	1.79	1.79
	50	---	---	---
	50	2.10	2.10	2.10
	40	2.40	2.40	2.40
	30	2.80	2.80	2.80
	30 at > 50%F	3.16	3.16	3.16
	25 at > 50%F	3.49	3.49	3.49
	30 at ≤ 50%F	3.04	3.04	3.04
	25 at ≤ 50%F	3.47	3.47	3.47

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.15 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 2 for GE 14: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVIS	100	1.54	1.54	1.54
	75	1.67	1.67	1.67
	65	1.79	1.80	1.80
	50	---	---	---
	50	2.18	2.18	2.18
	40	2.47	2.47	2.47
	30	2.89	2.89	2.89
	30 at > 50%F	2.89	2.89	2.89
	25 at > 50%F	3.05	3.05	3.05
	30 at ≤ 50%F	2.89	2.89	2.89
	25 at ≤ 50%F	2.96	2.96	2.96
TBVOOS	100	1.57	1.57	1.57
	75	1.70	1.70	1.70
	65	1.80	1.81	1.81
	50	---	---	---
	50	2.18	2.18	2.18
	40	2.47	2.47	2.47
	30	2.89	2.89	2.89
	30 at > 50%F	3.22	3.22	3.22
	25 at > 50%F	3.57	3.57	3.57
	30 at ≤ 50%F	3.13	3.13	3.13
	25 at ≤ 50%F	3.57	3.57	3.57

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.16 MCPR$_P$ Limits for ATRIUM-10: Single Loop Operation for All Scram Times*

Operating Condition	Power (% of rated)	BOC to End of Coast
RCPOOS FHOOS	100	2.04
	50	2.04
	40	2.18
	30	2.51
	30 at > 50%F	2.51
	25 at > 50%F	2.68
	30 at ≤ 50%F	2.51
	25 at ≤ 50%F	2.62
RCPOOS TBVOOS PLUOOS FHOOS	100	2.04
	50	2.04
	40	2.18
	30	2.51
	30 at > 50%F	3.02
	25 at > 50%F	3.34
	30 at ≤ 50%F	2.87
	25 at ≤ 50%F	3.26
RCPOOS TBVOOS FHOOS1	100	2.11
	50	2.11
	40	2.41
	30	2.81
	30 at > 50%F	3.17
	25 at > 50%F	3.50
	30 at ≤ 50%F	3.05
	25 at ≤ 50%F	3.48
RCPOOS TBVOOS FHOOS2	100	2.12
	50	2.12
	40	2.42
	30	2.82
	30 at > 50%F	3.18
	25 at > 50%F	3.51
	30 at ≤ 50%F	3.06
	25 at ≤ 50%F	3.49

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop.

RCPOOS limits are only valid up to 50% rated core power, 50% rated core flow, and an active recirculation drive flow of 17.73 Mlbm/hr.

Table 4.17 MCPR$_P$ Limits for GE-14: Single Loop Operation for All Scram Times*

Operating Condition	Power (% of rated)	BOC to End of Coast
RCPOOS FHOOS	100	2.00
	50	2.00
	40	2.24
	30	2.59
	30 at > 50%F	2.59
	25 at > 50%F	2.78
	30 at ≤ 50%F	2.59
	25 at ≤ 50%F	2.70
RCPOOS TBVOOS PLUOOS FHOOS	100	2.01
	50	2.01
	40	2.24
	30	2.59
	30 at > 50%F	3.08
	25 at > 50%F	3.38
	30 at ≤ 50%F	2.95
	25 at ≤ 50%F	3.35
RCPOOS TBVOOS FHOOS1	100	2.19
	50	2.19
	40	2.48
	30	2.89
	30 at > 50%F	3.23
	25 at > 50%F	3.57
	30 at ≤ 50%F	3.13
	25 at ≤ 50%F	3.57
RCPOOS TBVOOS FHOOS2	100	2.20
	50	2.20
	40	2.49
	30	2.91
	30 at > 50%F	3.24
	25 at > 50%F	3.59
	30 at ≤ 50%F	3.15
	25 at ≤ 50%F	3.59

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop.

RCPOOS limits are only valid up to 50% rated core power, 50% rated core flow, and an active recirculation drive flow of 17.73 Mlbm/hr.

5 Oscillation Power Range Monitor (OPRM) Setpoint

(Technical Specification 3.3.1.1)

Technical Specification Table 3.3.1.1-1, Function 2f, identifies the OPRM upscale function.

Instrument setpoints are established, such that the reactor will be tripped before an oscillation can grow to the point where the SLMCPR is exceeded. An Option III stability analysis is performed for each reload core to determine allowable OLMCPR's as a function of OPRM setpoint. Analyses consider both steady state startup operation, and the case of a two recirculation pump trip from rated power.

The resulting stability based OLMCPR's are reported in Reference 1. The OPRM setpoint (*sometimes referred to as the Amplitude Trip,* S_p) is selected, such that required margin to the SLMCPR is provided without stability being a limiting event. Analyses are based on cycle specific DIVOM analyses performed per Reference 26. The calculated OLMCPR's are shown in Table 5.1. Review of results shown in COLR Table 4.2 and Table 4.3 indicates an OPRM setpoint of **1.14** may be used. The successive confirmation count (*sometimes referred to as* N_p) is provided in Table 5.2, per Reference 31.

Table 5.1 OPRM Setpoint Range*

OPRM Setpoint	OLMCPR (SS)	OLMCPR (2PT)
1.05	1.18	1.23
1.06	1.20	1.24
1.07	1.22	1.26
1.08	1.24	1.28
1.09	1.26	1.30
1.10	1.28	1.33
1.11	1.30	1.35
1.12	1.32	1.37
1.13	1.34	1.39
1.14	1.36	1.41
1.15	1.39	1.44

Table 5.2 OPRM Successive Confirmation Count Setpoint

Count	OPRM Setpoint
6	≥ 1.04
8	≥ 1.05
10	≥ 1.07
12	≥ 1.09
14	≥ 1.11
16	≥ 1.14
18	≥ 1.18
20	≥ 1.24

* Extrapolation beyond a setpoint of 1.15 is not allowed

6 APRM Flow Biased Rod Block Trip Settings

(Technical Requirements Manual Section 5.3.1 and Table 3.3.4-1)

The APRM rod block trip setting is based upon References 27 & 28, and is defined by the following:

SRB $\leq$ (0.66(W-ΔW) + 61%)	Allowable Value
SRB $\leq$ (0.66(W-ΔW) + 59%)	Nominal Trip Setpoint (NTSP)

where:

SRB	=	Rod Block setting in percent of rated thermal power (3458 MW_t)
W	=	Loop recirculation flow rate in percent of rated
ΔW	=	Difference between two-loop and single-loop effective recirculation flow at the same core flow (ΔW=0.0 for two-loop operation)

The APRM rod block trip setting is clamped at a maximum allowable value of 115% (corresponding to a NTSP of 113%).

7 Rod Block Monitor (RBM) Trip Setpoints and Operability

(Technical Specification Table 3.3.2.1-1)

The RBM trip setpoints and applicable power ranges, based on References 27 & 28, are shown in Table 7.1. Setpoints are based on an HTSP, unfiltered analytical limit of 114%. Unfiltered setpoints are consistent with a nominal RBM filter setting of 0.0 seconds; filtered setpoints are consistent with a nominal RBM filter setting less than 0.5 seconds. Cycle specific CRWE analyses of OLMCPR are documented in Reference 1, superseding values reported in References 27, 28, and 30.

Table 7.1 Analytical RBM Trip Setpoints*

RBM Trip Setpoint		Allowable Value (AV)	Nominal Trip Setpoint (NTSP)
LPSP		27%	25%
IPSP		62%	60%
HPSP		82%	80%
LTSP	- unfiltered	121.7%	120.0%
	- filtered	120.7%	119.0%
ITSP	- unfiltered	116.7%	115.0%
	- filtered	115.7%	114.0%
HTSP	- unfiltered	111.7%	110.0%
	- filtered	110.9%	109.2%
DTSP		90%	92%

As a result of cycle specific CRWE analyses, RBM setpoints in Technical Specification Table 3.3.2.1-1 are applicable as shown in Table 7.2. Cycle specific setpoint analysis results are shown in Table 7.3, per Reference 1.

Table 7.2 RBM Setpoint Applicability

Thermal Power (% Rated)	Applicable MCPR†	Notes from Table 3.3.2.1-1	Comment
> 27% and < 90%	< 1.70	(a), (b), (f), (h)	two loop operation
	< 1.74	(a), (b), (f), (h)	single loop operation
≥ 90%	< 1.43	(g)	two loop operation‡

* Values are considered maximums. Using lower values, due to RBM system hardware/software limitations, is conservative, and acceptable.

† MCPR values shown correspond with, (support), SLMPCR values identified in Reference 1.

‡ Greater than 90% rated power is not attainable in single loop operation.

Table 7.3 Control Rod Withdrawal Error Results

RBM HTSP Analytical Limit	CRWE OLMCPR
Unfiltered	
107	1.28
111	1.31
114	1.36
117	1.40

Results, compared against the base case OLMCPR results of Table 4.2, indicate SLMCPR remains protected for RBM inoperable conditions (i.e., 114% unblocked).

8 Shutdown Margin Limit

(Technical Specification 3.1.1)

Assuming the strongest OPERABLE control blade is fully withdrawn, and all other OPERABLE control blades are fully inserted, the core shall be sub-critical and meet the following minimum shutdown margin:

SDM > 0.38% dk/k

EDMS L32 160909 801
QA Document
BFE-4086, Revision 0

TVA NPG

Reactor Engineering and Fuels - BWRFE
1101 Market Street, Chattanooga, TN 37402

Browns Ferry Unit 1 Cycle 12
Core Operating Limits Report, (105% OLTP)

TVA-COLR-BF1C12 **Revision 0 (Final)**
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Revision Log

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0-R0	All	New document.

Nomenclature

APLHGR	Average Planar LHGR
APRM	Average Power Range Monitor
AREVA NP	Vendor (Framatome, Siemens)
BOC	Beginning of Cycle
BSP	Backup Stability Protection
BWR	Boiling Water Reactor
CAVEX	Core Average Exposure
CD	Coast Down
CMSS	Core Monitoring System Software
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRWE	Control Rod Withdrawal Error
CSDM	Cold SDM
DIVOM	Delta CPR over Initial CPR vs. Oscillation Magnitude
EOC	End of Cycle
EOCLB	End-of-Cycle Licensing Basis
EOOS	Equipment OOS
FFTR	Final Feedwater Temperature Reduction
FFWTR	Final Feedwater Temperature Reduction
FHOOS	Feedwater Heaters OOS
ft	Foot: english unit of measure for length
GNF	Vendor (General Electric, Global Nuclear Fuels)
GWd	Giga Watt Day
HTSP	High TSP
ICA	Interim Corrective Action
ICF	Increased Core Flow (beyond rated)
IS	In-Service
kW	kilo watt: SI unit of measure for power.
LCO	License Condition of Operation
LFWH	Loss of Feedwater Heating
LHGRFAC	LHGR Multiplier (Power or Flow dependent)
LPRM	Low Power Range Monitor
LRNB	Generator Load Reject, No Bypass
MAPFAC	MAPLHGR multiplier (Power or Flow dependent)
MCPR	Minimum CPR

MSRV	Moisture Separator Reheater Valve
MSRVOOS	MSRV OOS
MTU	Metric Ton Uranium
MWd/MTU	Mega Watt Day per Metric Ton Uranium
NEOC	Near EOC
NRC	United States Nuclear Regulatory Commission
NSS	Nominal Scram Speed
NTSP	Nominal TSP
OLMCPR	MCPR Operating Limit
OOS	Out-Of-Service
OPRM	Oscillation Power Range Monitor
OSS	Optimum Scram Speed
PBDA	Period Based Detection Algorithm
Pbypass	Power, below which TSV Position and TCV Fast Closure Scrams are Bypassed
PLU	Power Load Unbalance
PLUOOS	PLU OOS
PRNM	Power Range Neutron Monitor
RBM	Rod Block Monitor
RCPOOS	Recirculation Pump OOS (*SLO*)
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPTOOS	RPT OOS
SDM	Shutdown Margin
SLMCPR	MCPR Safety Limit
SLO	Single Loop Operation
TBV	Turbine Bypass Valve
TBVIS	TBV IS
TBVOOS	Turbine Bypass Valves OOS
TIP	Transversing In-core Probe
TIPOOS	TIP OOS
TLO	Two Loop Operation
TSP	Trip Setpoint
TSSS	Technical Specification Scram Speed
TVA	Tennessee Valley Authority

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1 Introduction

In anticipation of cycle startup, it is necessary to describe the expected limits of operation.

1.1 Purpose

The primary purpose of this document is to satisfy requirements identified by unit technical specification section 5.6.5. This document may be provided, upon final approval, to the NRC.

1.2 Scope

This document will discuss the following areas:

- Average Planar Linear Heat Generation Rate (APLHGR) Limit
  (Technical Specifications 3.2.1 and 3.7.5)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Linear Heat Generation Rate (LHGR) Limit
  (Technical Specification 3.2.3, 3.3.4.1, and 3.7.5)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Minimum Critical Power Ratio Operating Limit (OLMCPR)
  (Technical Specifications 3.2.2, 3.3.4.1, 3.7.5 and Table 3.3.2.1-1)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Oscillation Power Range Monitor (OPRM) Setpoint
  (Technical Specification Table 3.3.1.1)
  Applicability: Mode 1, ≥ (as specified in Technical Specifications Table 3.3.1.1-1)
- Average Power Range Monitor (APRM) Flow Biased Rod Block Trip Setting
  (Technical Requirements Manual Section 5.3.1 and Table 3.3.4-1)
  Applicability: Mode 1, ≥ (as specified in Technical Requirements Manuals Table 3.3.4-1)
- Rod Block Monitor (RBM) Trip Setpoints and Operability
  (Technical Specification Table 3.3.2.1-1)
  Applicability: Mode 1, ≥ % RTP as specified in Table 3.3.2.1-1 (TS definition of RTP)
- Shutdown Margin (SDM) Limit
  (Technical Specification 3.1.1)
  Applicability: All Modes

1.3 Fuel Loading

The core will contain previously exposed AREVA Inc., ATRIUM-10 fuel, along with fresh ATRIUM-10XM. Nuclear fuel types used in the core loading are shown in Table 1.1. The planned outage will consist of a shuffle for maintenance. The final core loading was evaluated for BOC cold shutdown margin performance as documented per Reference 5.

Table 1.1 Nuclear Fuel Types*

Fuel Description	Original Cycle	Number of Assemblies	Nuclear Fuel Type (NFT)	Fuel Names (Range)
ATRIUM-10 A10-3562B-14GV80-FAA	10	65	20	FAA002-FAA168
ATRIUM-10 A10-3676B-10GV80-FAA	10	24	21	FAA169-FAA192
ATRIUM-10 A10-4111B-15GV80-FAA	10	87	22	FAA193-FAA280
ATRIUM-10 A10-3840B-14GV80-FAB	11	200	23	FAB301-FAB500
ATRIUM-10 A10-4117B-13GV70-FAB	11	72	24	FAB501-FAB572
ATRIUM-10 A10-4112B-15GV70-FAB	11	36	25	FAB573-FAB608
ATRIUM-10XM XMLC-4102B-11GV70-FAC-B	12	40	26	FAC701-FAC740
ATRIUM-10XM XMLC-3969B-13GV80-FAC-C	12	128	27	FAC741-FAC868
ATRIUM-10XM XMLC-3948B-13GV70-FAC-B	12	112	28	FAC869-FAC980

1.4 Acceptability

Limits discussed in this document were generated based on NRC approved methodologies per References 6 through 25.

* The table identifies the expected fuel type breakdown in anticipation of final core loading. The final composition of the core depends upon uncertainties during the outage such as discovering a failed fuel bundle, or other bundle damage. Minor core loading changes, due to unforeseen events, will conform to the safety and monitoring requirements identified in this document.

2 APLHGR Limits

(Technical Specifications 3.2.1 & 3.7.5)

The APLHGR limit is determined by adjusting the rated power APLHGR limit for off-rated power, off-rated flow, and SLO conditions. The most limiting of these is then used as follows:

$$\text{APLHGR limit} = \text{MIN}\ (\text{APLHGR}_P\ ,\ \text{APLHGR}_F,\ \text{APLHGR}_{SLO}\ )$$

where:

APLHGR_P	off-rated power APLHGR limit	[APLHGR_{RATED} * MAPFAC_P]
APLHGR_F	off-rated flow APLHGR limit	[APLHGR_{RATED} * MAPFAC_F]
APLHGR_{SLO}	SLO APLHGR limit	[APLHGR_{RATED} * SLO Multiplier]

2.1 Rated Power and Flow Limit: APLHGR_{RATED}

The rated conditions APLHGR for ATRIUM-10 fuel is identified in Reference 1 and shown in Figure 2.1. The rated conditions APLHGR for ATRIUM-10XM are shown in Figure 2.2.

2.2 Off-Rated Power Dependent Limit: APLHGR_P

Reference 1 does not specify a power dependent APLHGR. Therefore, MAPFAC_P is set to a value of **1.0**.

2.2.1 Startup without Feedwater Heaters

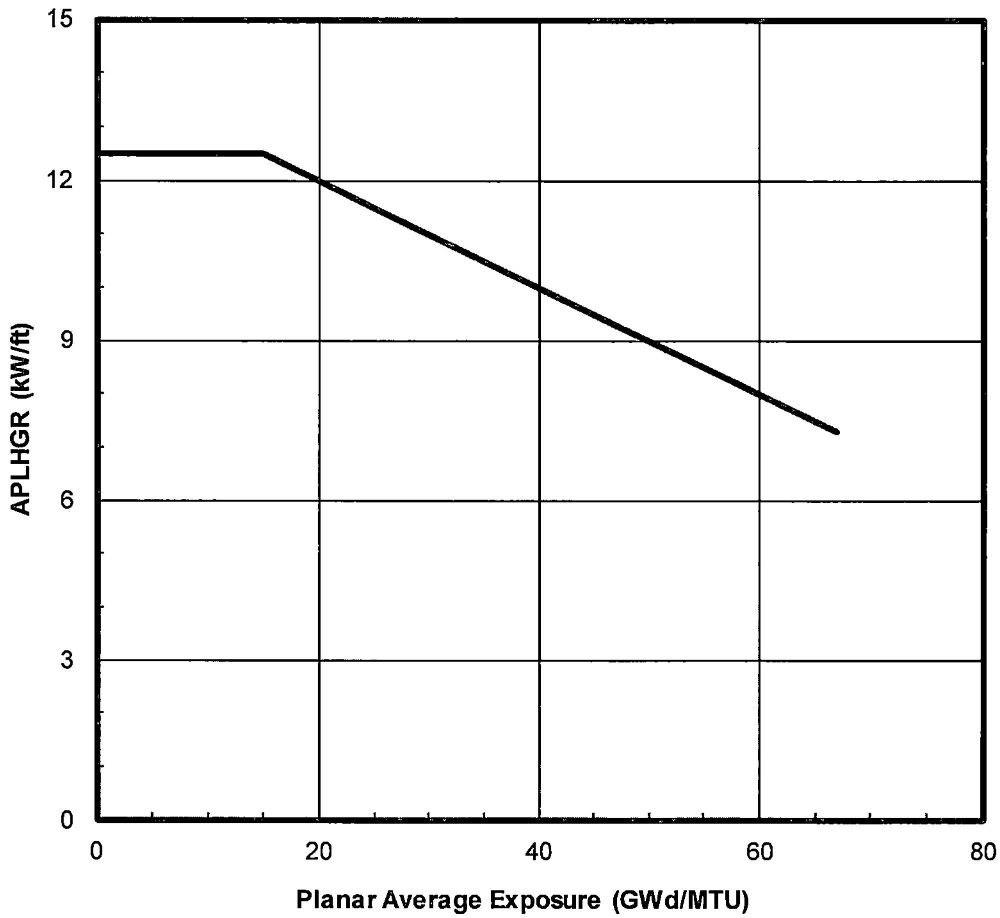
There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. No additional power dependent limitation is required.

2.3 Off-Rated Flow Dependent Limit: APLHGR_F

Reference 1 does not specify a flow dependent APLHGR. Therefore, MAPFAC_F is set to a value of **1.0**.

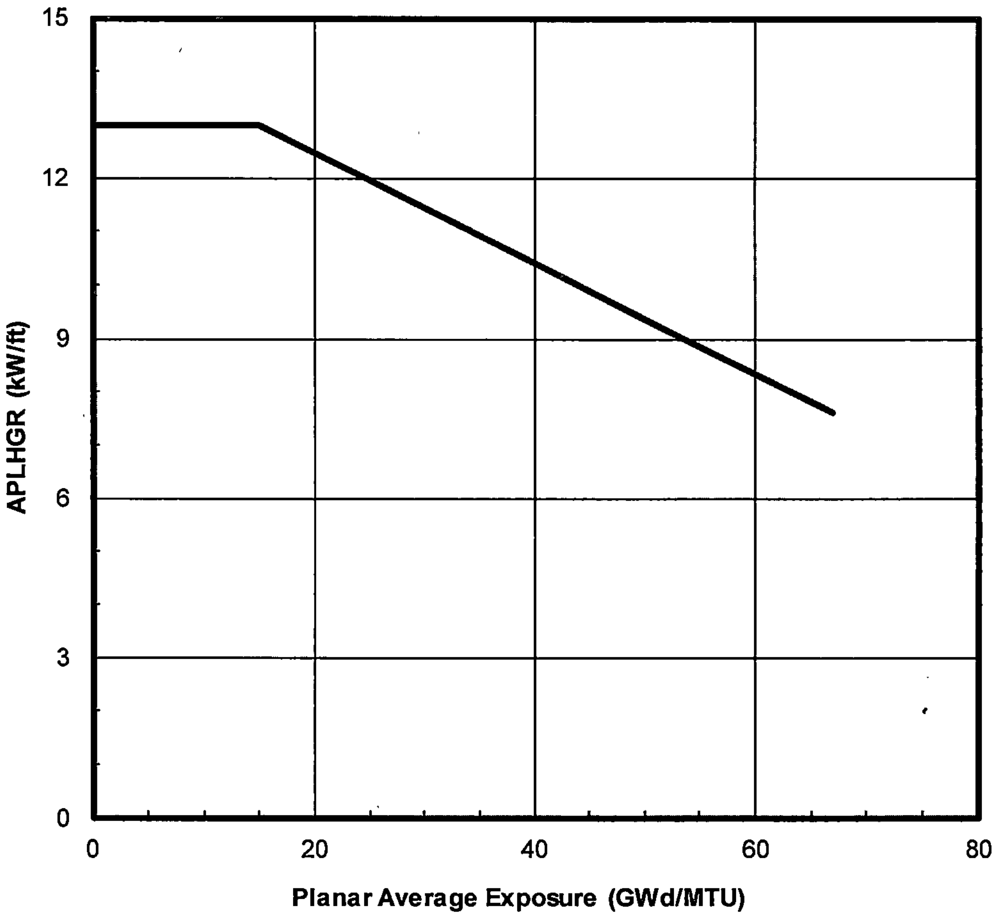
2.4 Single Loop Operation Limit: APLHGR_{SLO}

The single loop operation multiplier for ATRIUM-10, and ATRIUM-10XM fuel is **0.85**, per Reference 1.



Planar Avg. Exposure (GWd/MTU)	APLHGR Limit (kW/ft)
0.0	12.5
15.0	12.5
67.0	7.3

Figure 2.1 APLHGR$_{RATED}$ for ATRIUM-10 Fuel



Planar Avg. Exposure (GWd/MTU)	APLHGR Limit (kW/ft)
0.0	13.0
15.0	13.0
67.0	7.6

Figure 2.2 APLHGR$_{RATED}$ for ATRIUM-10XM Fuel

2.5 Equipment Out-Of-Service Corrections

The limits shown in Figure 2.1 and Figure 2.2 are applicable for operation with all equipment In-Service as well as the following Equipment Out-Of-Service (EOOS) options; including combinations of the options.

In-Service	All equipment In-Service*
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
PLUOOS	Power Load Unbalance Out-Of-Service
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service or Final Feedwater Temperature Reduction
RCPOOS	One Recirculation Pump Out-Of-Service

* All equipment service conditions assume 1 SRVOOS.

3 LHGR Limits

(Technical Specification 3.2.3, 3.3.4.1, & 3.7.5)

The LHGR limit is determined by adjusting the rated power LHGR limit for off-rated power and off-rated flow conditions. The most limiting of these is then used as follows:

LHGR limit = MIN ($LHGR_P$, $LHGR_F$)

where:

$LHGR_P$	off-rated power LHGR limit	[$LHGR_{RATED}$ * $LHGRFAC_P$]
$LHGR_F$	off-rated flow LHGR limit	[$LHGR_{RATED}$ * $LHGRFAC_F$]

3.1 Rated Power and Flow Limit: $LHGR_{RATED}$

The rated conditions LHGR for ATRIUM-10 fuel is identified in Reference 1 and shown in Figure 3.1. The rated conditions LHGR for ATRIUM-10XM fuel is shown in Figure 3.2. The LHGR limit is consistent with References 2 and 3.

3.2 Off-Rated Power Dependent Limit: $LHGR_P$

LHGR limits are adjusted for off-rated power conditions using the $LHGRFAC_P$ multiplier provided in Reference 1. The multiplier is split into two sub cases: turbine bypass valves in and out-of-service. The base case multipliers are shown in Figure 3.3 and Figure 3.4.

3.2.1 Startup without Feedwater Heaters

There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. Additional limits are shown in Figure 3.7 through Figure 3.10, based on temperature conditions identified in Table 3.1.

Table 3.1 Startup Feedwater Temperature Basis

Power	Temperature	
	Range 1	Range 2
(% Rated)	(°F)	(°F)
25	160.0	155.0
30	165.0	160.0
40	175.0	170.0
50	185.0	180.0

3.3 Off-Rated Flow Dependent Limit: $LHGR_F$

LHGR limits are adjusted for off-rated flow conditions using the $LHGRFAC_F$ multiplier provided in Reference 1. Multipliers are shown in Figure 3.5 and Figure 3.6.

3.4 Equipment Out-Of-Service Corrections

The limits shown in Figure 3.1 and Figure 3.2 are applicable for operation with all equipment In-Service as well as the following Equipment Out-Of-Service (EOOS) options; including combinations of the options.*

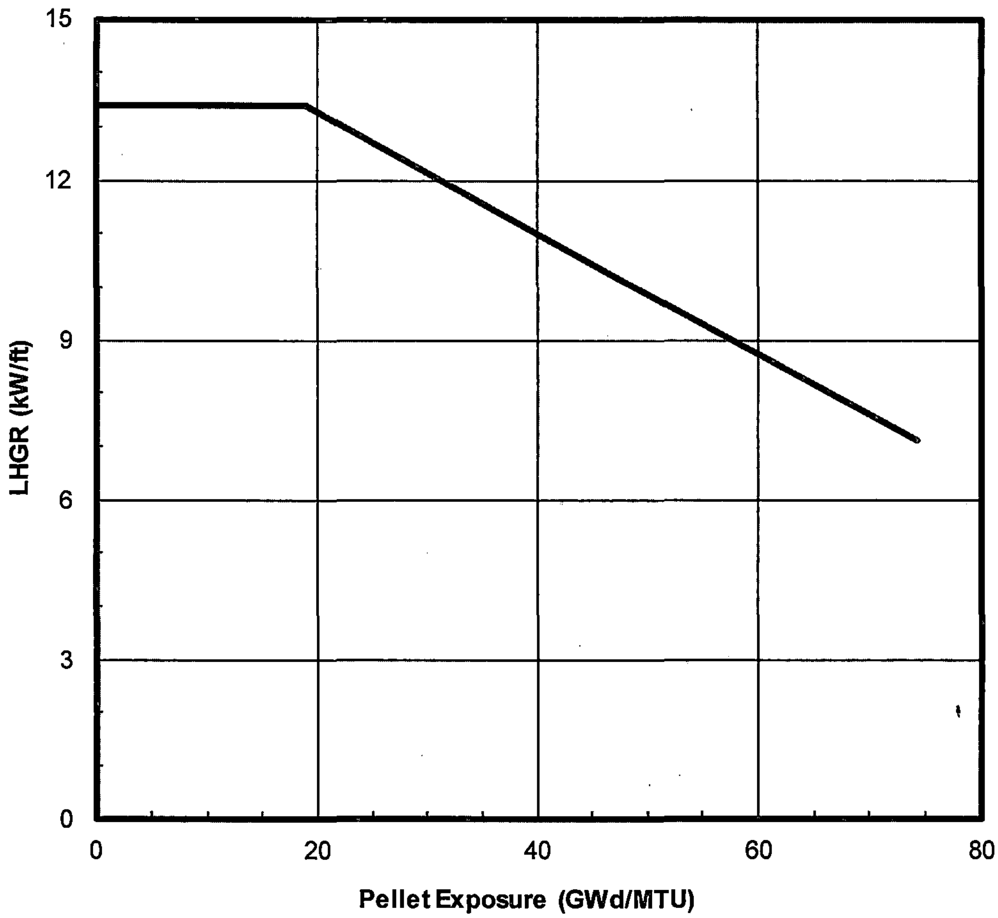
In-Service	All equipment In-Service
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
PLUOOS	Power Load Unbalance Out-Of-Service
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service or Final Feedwater Temperature Reduction
RCPOOS	One Recirculation Pump Out-Of-Service

Off-rated power corrections shown in Figure 3.3 and Figure 3.4 are dependent on operation of the Turbine Bypass Valve system. For this reason, separate limits are to be applied for TBVIS or TBVOOS operation. The limits have no dependency on RPTOOS, PLUOOS, FHOOS/FFWTR, or SLO.

Off-rated flow corrections shown in Figure 3.5 and Figure 3.6 are bounding for all EOOS conditions.

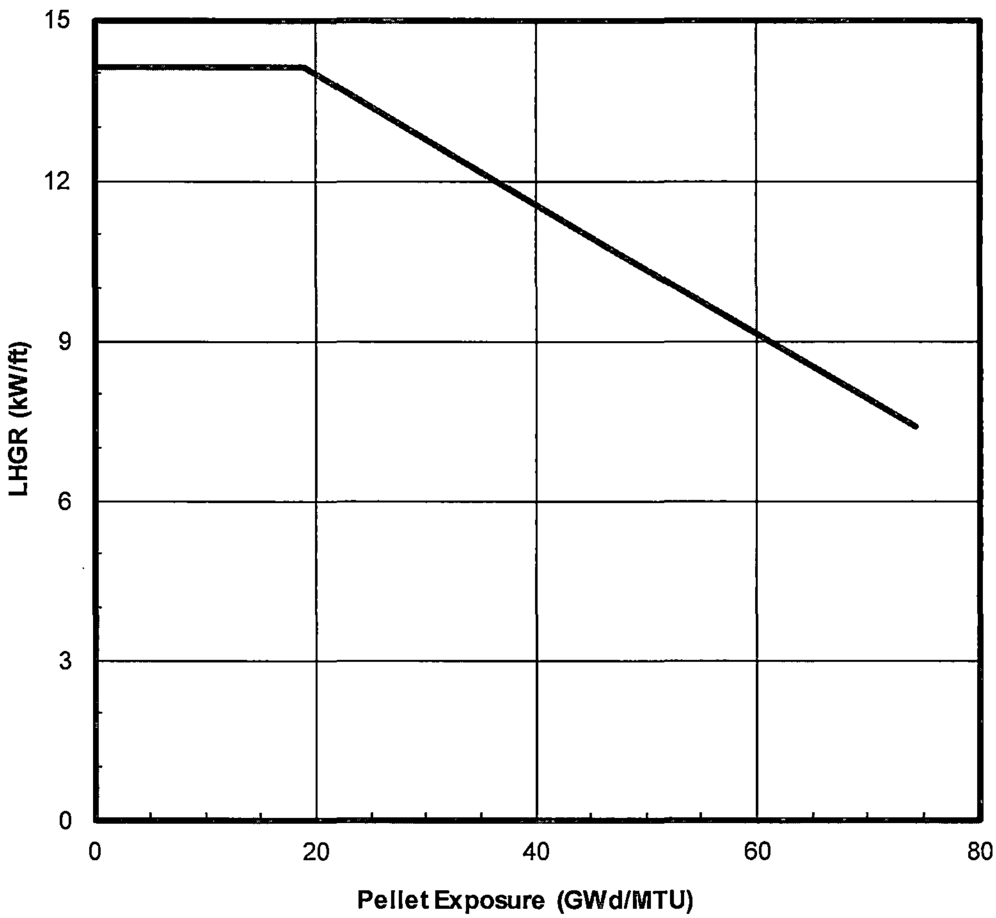
Off-rated power corrections shown in Figure 3.7 through Figure 3.10 are also dependent on operation of the Turbine Bypass Valve system. In this case, limits support FHOOS operation during startup. These limits have no dependency on RPTOOS, PLUOOS, or SLO.

* All equipment service conditions assume 1 SRVOOS.



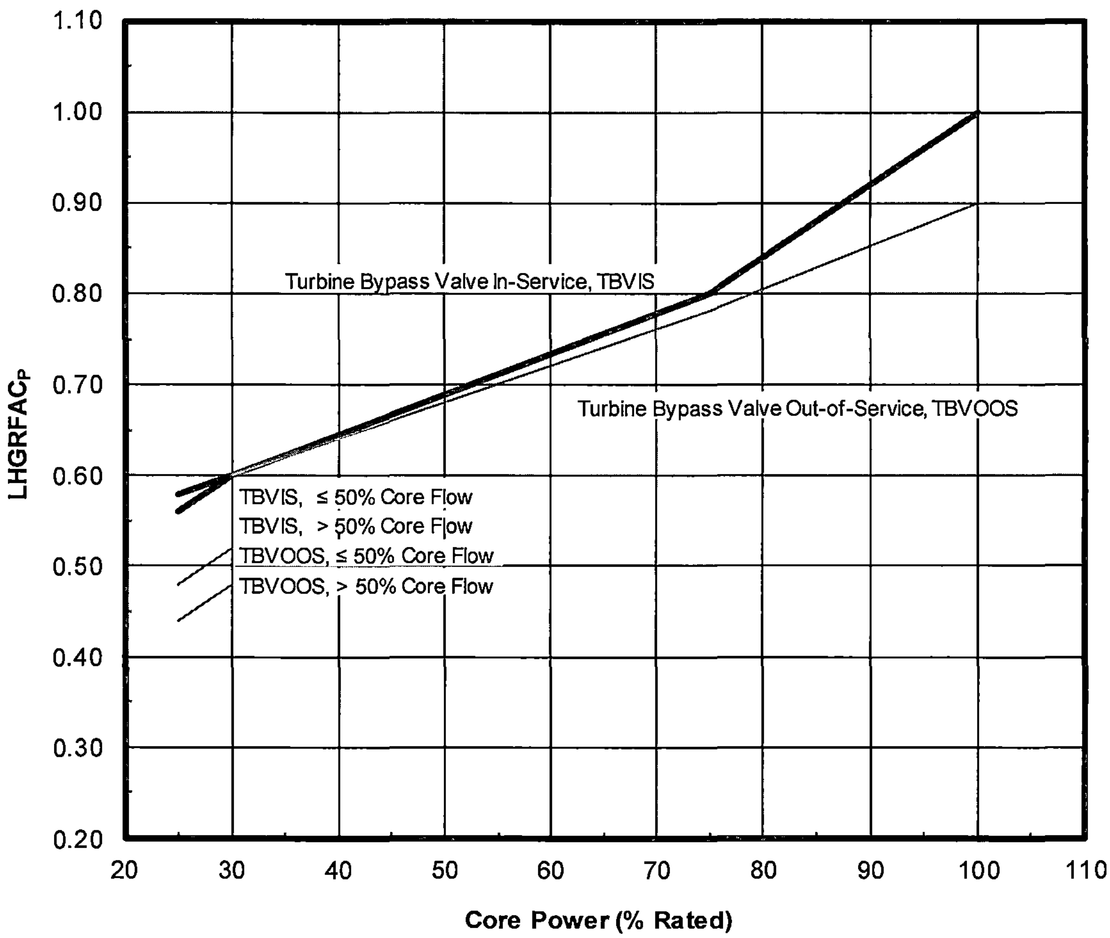
Pellet Exposure (GWd/MTU)	LHGR Limit (kW/ft)
0.0	13.4
18.9	13.4
74.4	7.1

Figure 3.1 $LHGR_{RATED}$ for ATRIUM-10 Fuel



Pellet Exposure (GWd/MTU)	LHGR Limit (kW/ft)
0.0	14.1
18.9	14.1
74.4	7.4

Figure 3.2 $LHGR_{RATED}$ for ATRIUM-10XM Fuel

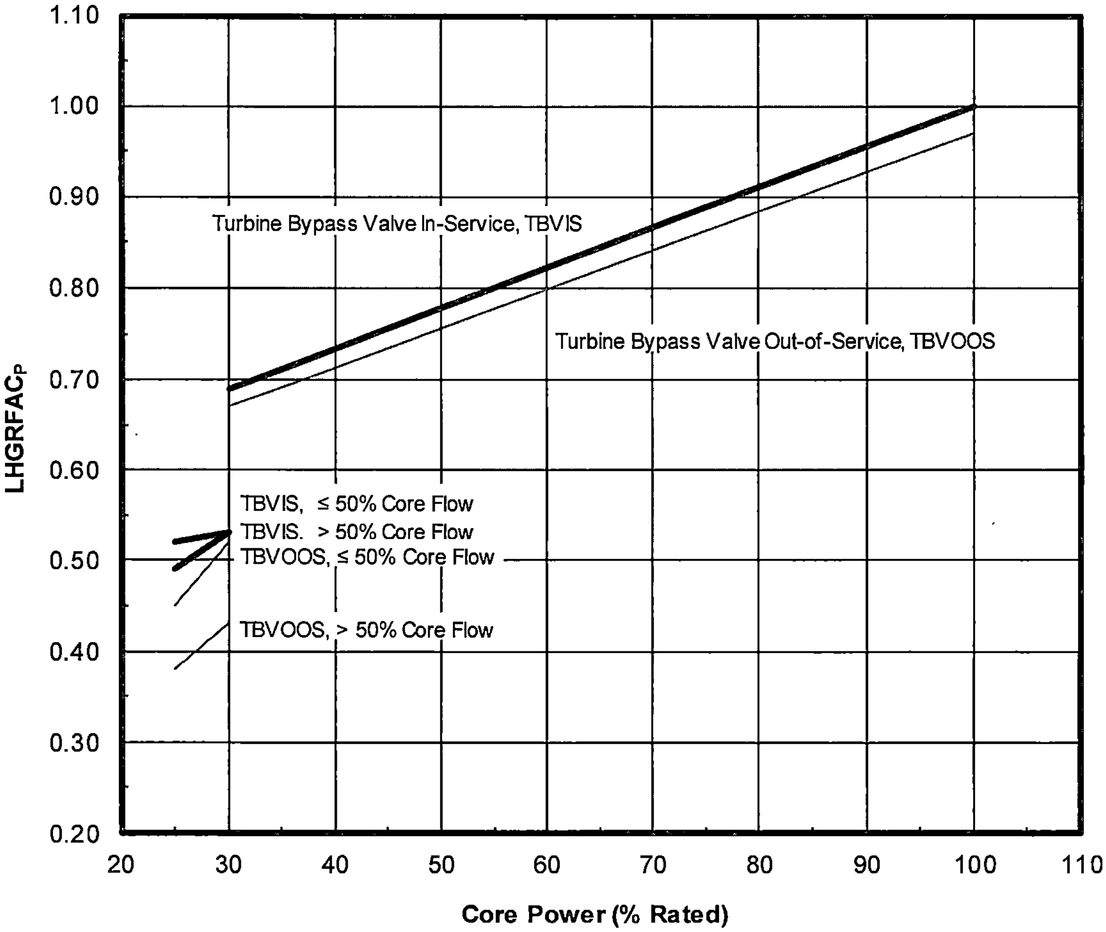


Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00
75.0	0.80
30.0	0.60
Core Flow > 50% Rated	
30.0	0.60
25.0	0.56
Core Flow ≤ 50% Rated	
30.0	0.60
25.0	0.58

Turbine Bypass Out-of-Service Core Power (% Rated)	LHGRFAC$_P$
100.0	0.90
75.0	0.78
30.0	0.60
Core Flow > 50% Rated	
30.0	0.48
25.0	0.44
Core Flow ≤ 50% Rated	
30.0	0.52
25.0	0.48

Figure 3.3 Base Operation LHGRFAC$_P$ for ATRIUM-10 Fuel

(*Independent of other EOOS conditions*)

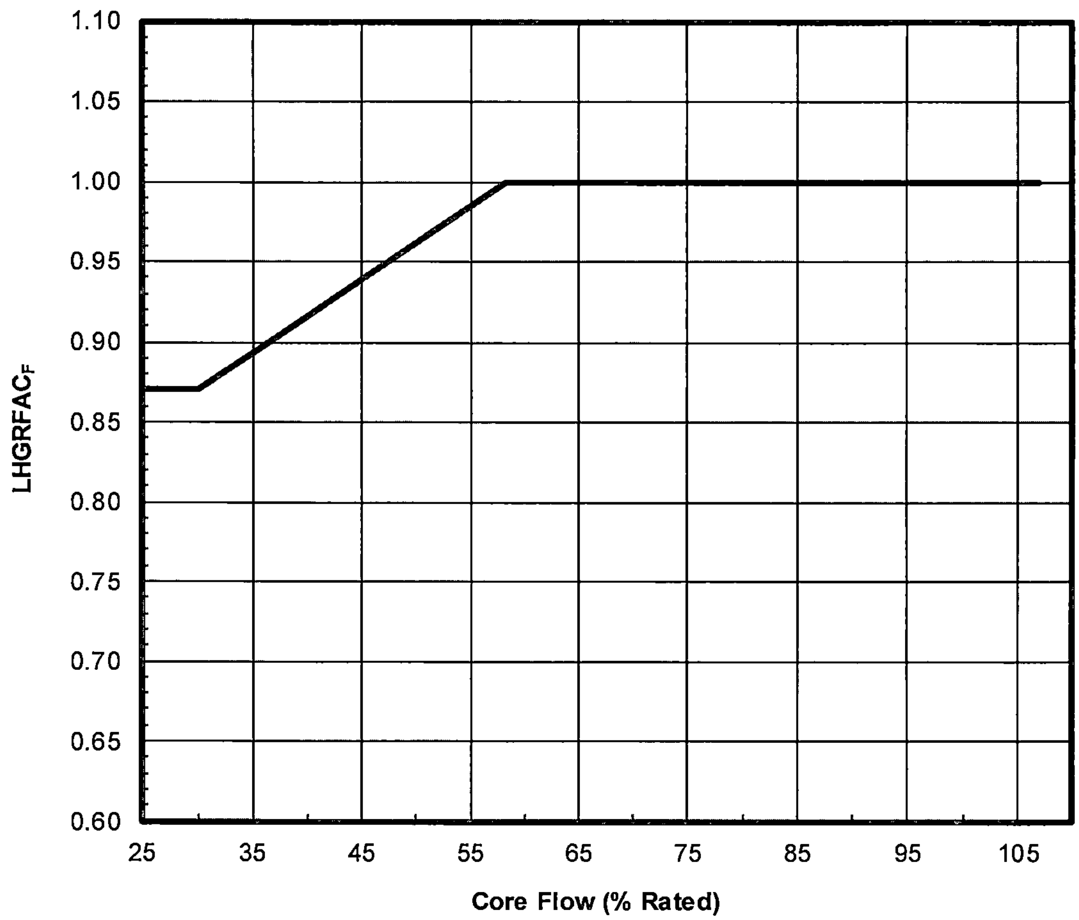


Turbine Bypass In-Service	
Core Power (% Rated)	**$LHGRFAC_P$**
100.0	1.00
30.0	0.69
Core Flow > 50% Rated	
30.0	0.53
25.0	0.49
Core Flow ≤ 50% Rated	
30.0	0.53
25.0	0.52

Turbine Bypass Out-of-Service	
Core Power (% Rated)	**$LHGRFAC_P$**
100.0	0.97
30.0	0.67
Core Flow > 50% Rated	
30.0	0.43
25.0	0.38
Core Flow ≤ 50% Rated	
30.0	0.52
25.0	0.45

Figure 3.4 Base Operation $LHGRFAC_P$ for ATRIUM-10XM Fuel

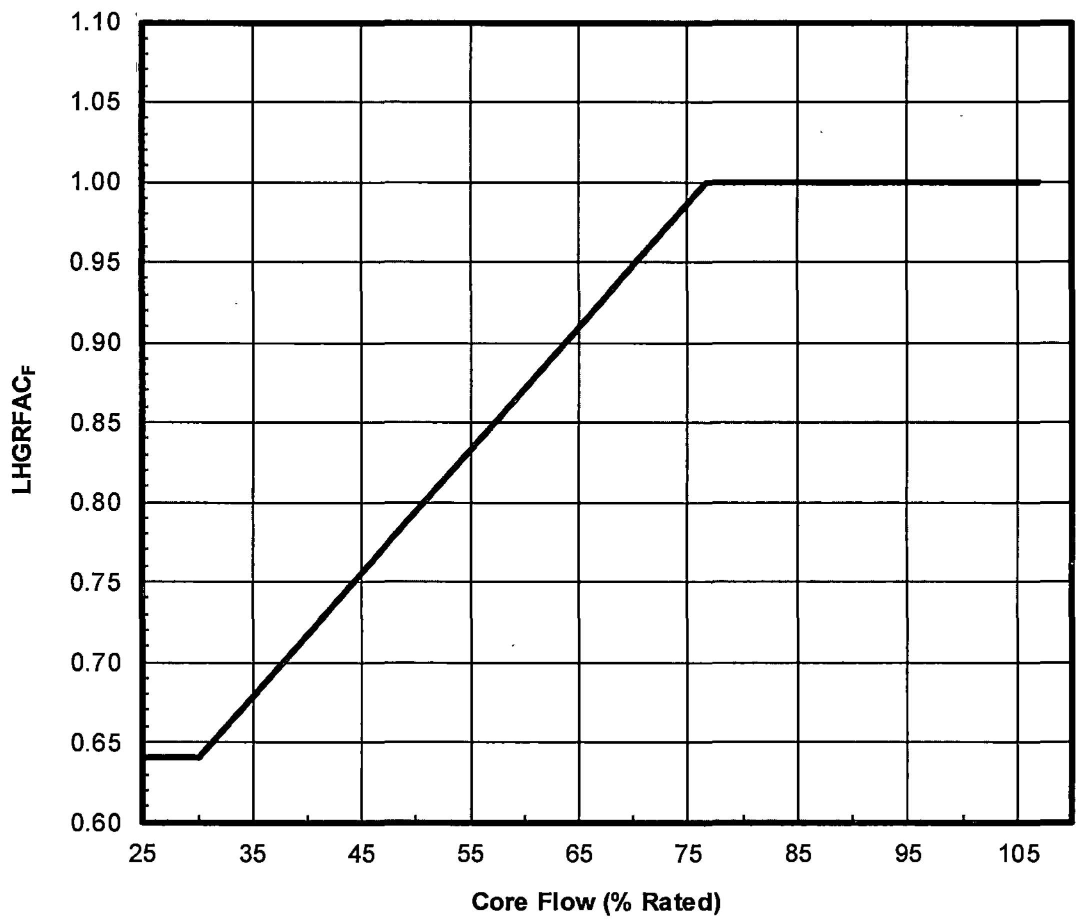
(*Independent of other EOOS conditions*)



Core Flow (% Rated)	LHGRFAC$_F$
0.0	0.87
30.0	0.87
58.3	1.00
107.0	1.00

Figure 3.5 LHGRFAC$_F$ for ATRIUM-10 Fuel
(*Values bound all EOOS conditions*)

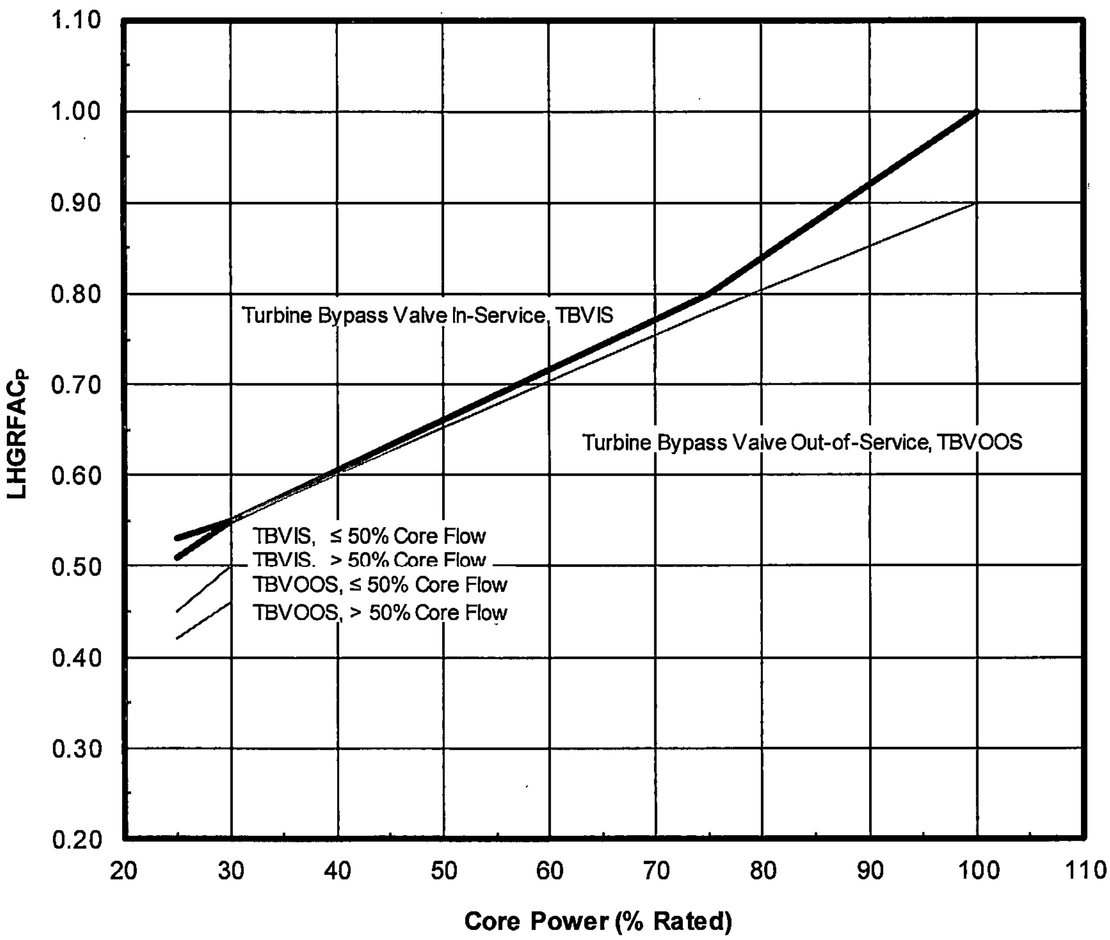
(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)



Core Flow (% Rated)	LHGRFAC$_F$
0.0	0.64
30.0	0.64
76.7	1
107.0	1

Figure 3.6 LHGRFAC$_F$ for ATRIUM-10XM Fuel
(*Values bound all EOOS conditions*)

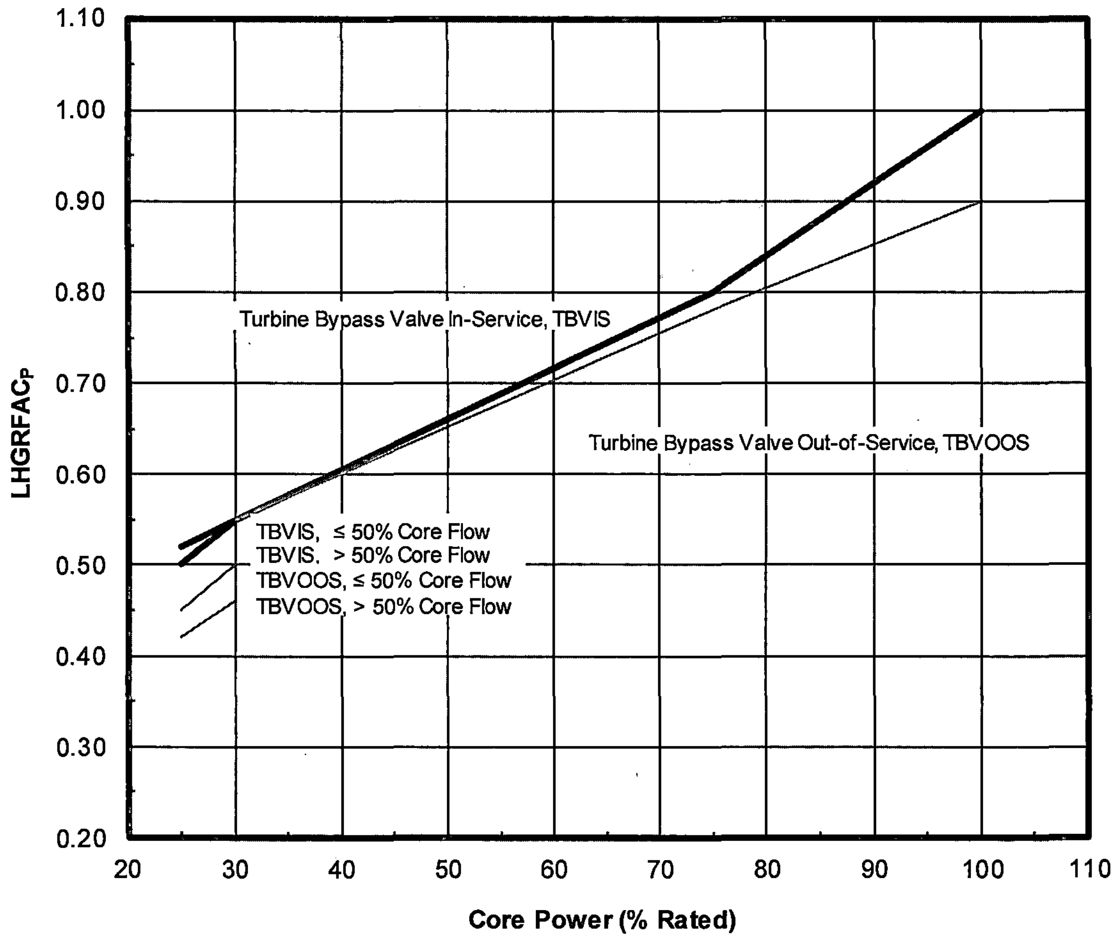
(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)



Turbine Bypass In-Service Core Power	$LHGRFAC_P$
(% Rated)	
100.0	1.00
75.0	0.80
30.0	0.55
Core Flow > 50% Rated	
30.0	0.55
25.0	0.51
Core Flow ≤ 50% Rated	
30.0	0.55
25.0	0.53

Turbine Bypass Out-of-Service Core Power	$LHGRFAC_P$
(% Rated)	
100.0	0.90
75.0	0.78
30.0	0.55
Core Flow > 50% Rated	
30.0	0.46
25.0	0.42
Core Flow ≤ 50% Rated	
30.0	0.50
25.0	0.45

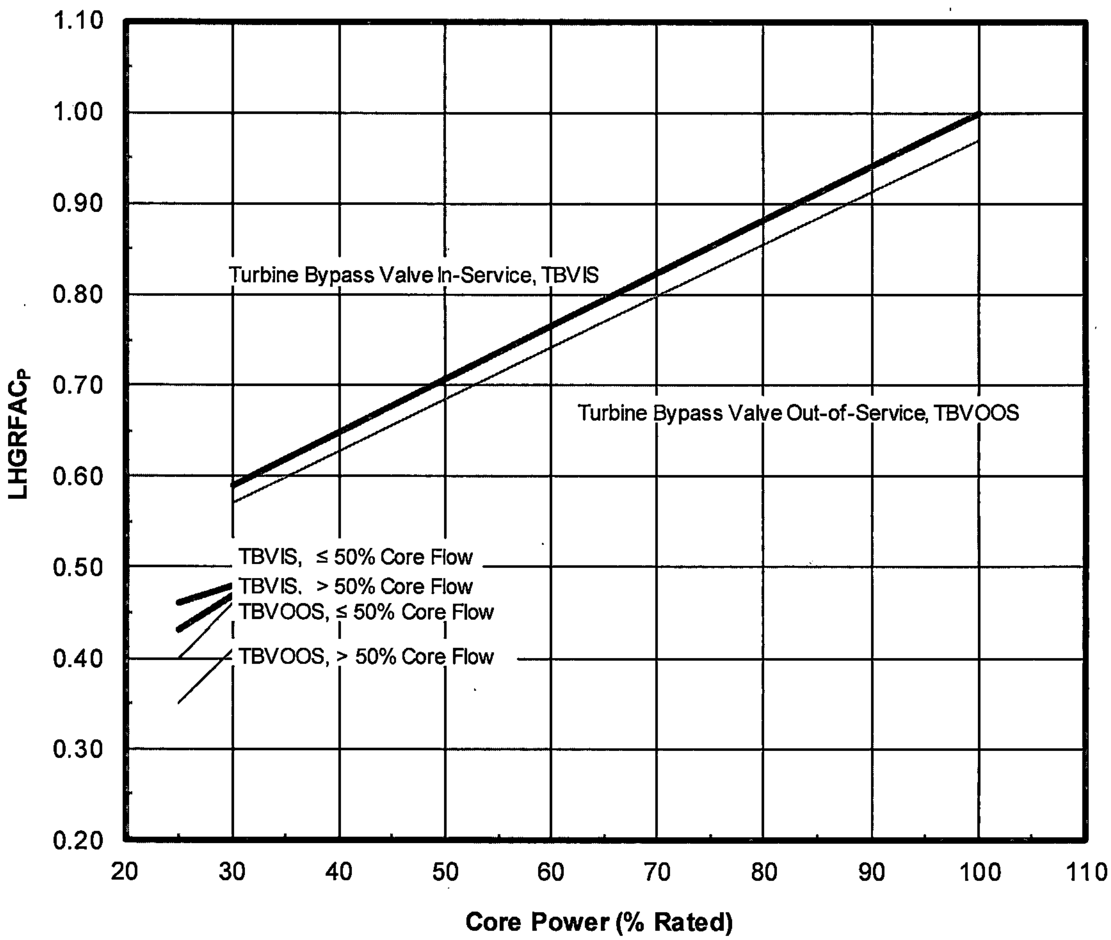
Figure 3.7 Startup Operation $LHGRFAC_P$ for ATRIUM-10 Fuel:
Table 3.1 Temperature Range 1
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00
75.0	0.80
30.0	0.55
Core Flow > 50% Rated	
30.0	0.55
25.0	0.50
Core Flow ≤ 50% Rated	
30.0	0.55
25.0	0.52

Turbine Bypass Out-of-Service Core Power (% Rated)	LHGRFAC$_P$
100.0	0.90
75.0	0.78
30.0	0.55
Core Flow > 50% Rated	
30.0	0.46
25.0	0.42
Core Flow ≤ 50% Rated	
30.0	0.50
25.0	0.45

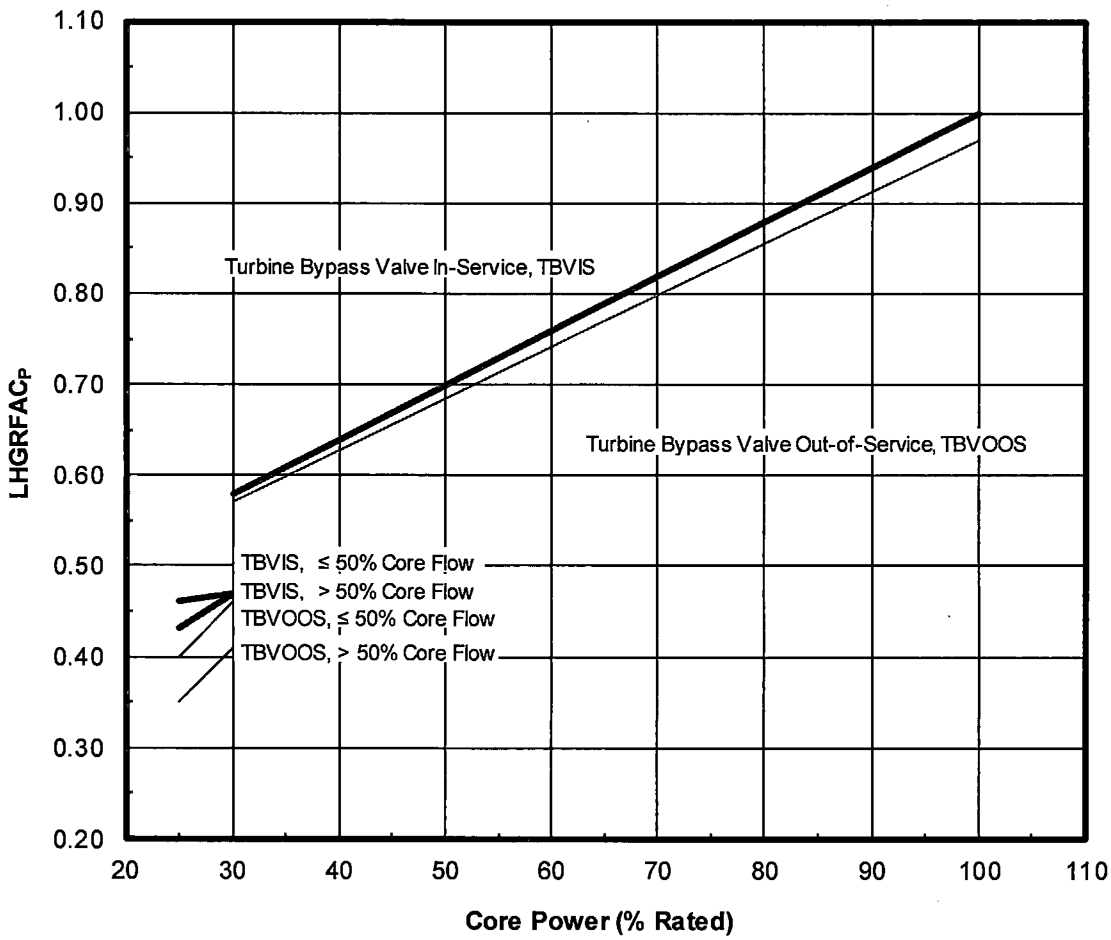
Figure 3.8 Startup Operation LHGRFAC$_P$ for ATRIUM-10 Fuel:
Table 3.1 Temperature Range 2
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



Turbine Bypass In-Service	
Core Power (% Rated)	**LHGRFAC$_P$**
100.0	1.00
30.0	0.59
Core Flow > 50% Rated	
30.0	0.47
25.0	0.43
Core Flow ≤ 50% Rated	
30.0	0.48
25.0	0.46

Turbine Bypass Out-of-Service	
Core Power (% Rated)	**LHGRFAC$_P$**
100.0	0.97
30.0	0.57
Core Flow > 50% Rated	
30.0	0.41
25.0	0.35
Core Flow ≤ 50% Rated	
30.0	0.46
25.0	0.40

Figure 3.9 Startup Operation LHGRFAC$_P$ for ATRIUM-10XM Fuel:
Table 3.1 Temperature Range 1
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



Turbine Bypass In-Service	
Core Power (% Rated)	**LHGRFAC$_P$**
100.0	1.00
30.0	0.58
Core Flow > 50% Rated	
30.0	0.47
25.0	0.43
Core Flow ≤ 50% Rated	
30.0	0.47
25.0	0.46

Turbine Bypass Out-of-Service	
Core Power (% Rated)	**LHGRFAC$_P$**
100.0	0.97
30.0	0.57
Core Flow > 50% Rated	
30.0	0.41
25.0	0.35
Core Flow ≤ 50% Rated	
30.0	0.46
25.0	0.40

Figure 3.10 Startup Operation LHGRFAC$_P$ for ATRIUM-10XM Fuel: Table 3.1 Temperature Range 2
(no Feedwater heating during startup)
(Limits valid at and below 50% power)

4 OLMCPR Limits

(Technical Specification 3.2.2, 3.3.4.1, & 3.7.5)

OLMCPR is calculated to be the most limiting of the flow or power dependent values

$$\text{OLMCPR limit} = \text{MAX} (\text{MCPR}_F , \text{MCPR}_P)$$

where:

$MCPR_F$	core flow-dependent MCPR limit
$MCPR_P$	power-dependent MCPR limit

4.1 Flow Dependent MCPR Limit: $MCPR_F$

$MCPR_F$ limits are dependent upon core flow (% of Rated), and the max core flow limit, (Rated or Increased Core Flow, ICF). $MCPR_F$ limits are shown in Figure 4.1, per Reference 1. Limits are valid for all EOOS combinations. No adjustment is required for SLO conditions.

4.2 Power Dependent MCPR Limit: $MCPR_P$

$MCPR_P$ limits are dependent upon:

- Core Power Level (% of Rated)
- Technical Specification Scram Speed (TSSS), Nominal Scram Speed (NSS), or Optimum Scram Speed (OSS)
- Cycle Operating Exposure (NEOC, EOC, and CD - as defined in this section)
- Equipment Out-Of-Service Options
- Two or Single recirculation Loop Operation (TLO vs. SLO)

The $MCPR_P$ limits are provided in Table 4.2 through Table 4.9, where each table contains the limits for all fuel types and EOOS options (for a specified scram speed and exposure range). The CMSS determines $MCPR_P$ limits, from these tables, based on linear interpolation between the specified powers.

4.2.1 Startup without Feedwater Heaters

There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. Additional power dependent limits are shown in Table 4.5 through Table 4.8 based on temperature conditions identified in Table 3.1.

4.2.2 Scram Speed Dependent Limits (TSSS vs. NSS vs. OSS)

$MCPR_P$ limits are provided for three different sets of assumed scram speeds. The Technical Specification Scram Speed (TSSS) $MCPR_P$ limits are applicable at all times, as long as the scram time surveillance demonstrates the times in Technical Specification Table 3.1.4-1 are met. Both Nominal Scram Speeds (NSS) and/or Optimum Scram Speeds (OSS) may be used, as long as the scram time surveillance demonstrates Table 4.1 times are applicable.*†

Table 4.1 Nominal Scram Time Basis

Notch Position (index)	Nominal Scram Timing (seconds)	Optimum Scram Timing (seconds)
46	0.420	0.380
36	0.980	0.875
26	1.600	1.465
6	2.900	2.900

In demonstrating compliance with the NSS and/or OSS scram time basis, surveillance requirements from Technical Specification 3.1.4 apply; accepting the definition of SLOW rods should conform to scram speeds shown in Table 4.1. If conformance is not demonstrated, TSSS based $MCPR_P$ limits are applied.

On initial cycle startup, TSSS limits are used until the successful completion of scram timing confirms NSS and/or OSS based limits are applicable.

4.2.3 Exposure Dependent Limits

Exposures are tracked on a Core Average Exposure basis (CAVEX, not Cycle Exposure). Higher exposure $MCPR_P$ limits are always more limiting and may be used for any Core Average Exposure up to the ending exposure. Per Reference 1, $MCPR_P$ limits are provided for the following exposure ranges:

BOC to NEOC	NEOC corresponds to	**29,949.1 MWd / MTU**
BOC to EOCLB	EOCLB corresponds to	**33,699.9 MWd / MTU**
BOC to End of Coast	End of Coast	**35,231.8 MWd / MTU**

* Reference 1 analysis results are based on information identified in Reference 4.

† Drop out times consistent with method used to perform actual timing measurements (i.e., including pickup/dropout effects).

NEOC refers to a Near EOC exposure point.

The EOCLB exposure point is not the true End-Of-Cycle exposure. Instead it corresponds to a licensing exposure window exceeding expected end-of-full-power-life.

The End of Coast exposure point represents a licensing exposure point exceeding the expected end-of-cycle exposure including cycle extension options.

4.2.4 Equipment Out-Of-Service (EOOS) Options

EOOS options* covered by $MCPR_P$ limits are given by the following:

In-Service	All equipment In-Service
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
RPTOOS+TBVOOS	Combined RPTOOS and TBVOOS
PLUOOS	Power Load Unbalance Out-Of-Service
PLUOOS+RPTOOS	Combined PLUOOS and RPTOOS
PLUOOS+TBVOOS	Combined PLUOOS and TBVOOS
PLUOOS+TBVOOS+RPTOOS	Combined PLUOOS, RPTOOS, and TBVOOS
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service (or Final Feedwater Temperature Reduction)
RCPOOS	One Recirculation Pump Out-Of-Service

For exposure ranges up to NEOC and EOCLB, additional combinations of $MCPR_P$ limits are also provided including FHOOS. The coast down exposure range assumes application of FFWTR. FHOOS based $MCPR_P$ limits for the coast down exposure are redundant because the temperature setdown assumption is identical with FFWTR.

4.2.5 Single-Loop-Operation (SLO) Limits

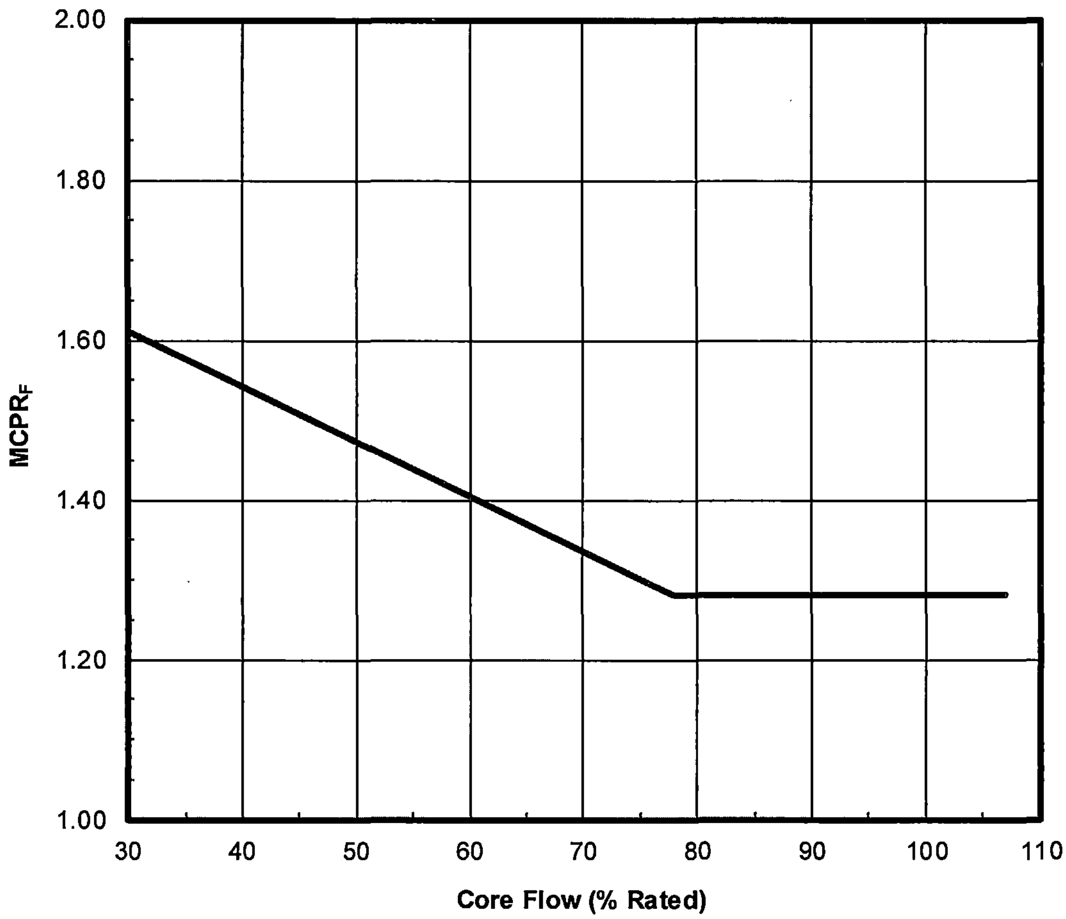
When operating in RCPOOS conditions, $MCPR_p$ limits are constructed differently from the normal operating RCP conditions. The limiting event for RCPOOS is a pump seizure scenario, which sets the upper bound for allowed core power and flow†. This event is not impacted by scram time assumptions. Specific $MCPR_P$ limits are shown in Table 4.9.

4.2.6 Below Pbypass Limits

Below Pbypass (30% rated power), $MCPR_P$ limits depend upon core flow. One set of $MCPR_P$ limits applies for core flow above 50% of rated; a second set applies if the core flow is less than or equal to 50% rated.

* All equipment service conditions assume 1 SRVOOS.

† RCPOOS limits are only valid up to 50% rated core power, 50% rated core flow, and an active recirculation drive flow of 17.73 Mlb_m/hr.



Core Flow (% Rated)	$MCPR_F$
30.0	1.61
78.0	1.28
107.0	1.28

Figure 4.1 $MCPR_F$ for All Fuel Types
(*Values bound all EOOS conditions*)

(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)

Table 4.2 MCPRp Limits for All Fuel Types: Optimum Scram Time Basis*

		ATRIUM-10			ATRIUM-10XM		
Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
Base Case	100	1.47	1.49	1.50	1.41	1.42	1.44
	75	1.59	1.59	1.62	1.52	1.52	1.57
	65	1.65	1.65	1.69	1.58	1.58	1.62
	50	1.84	1.84	1.92	1.72	1.72	1.79
	50	1.95	1.95	1.95	1.79	1.79	1.80
	40	2.07	2.07	2.17	1.93	1.93	2.02
	30	2.39	2.39	2.51	2.21	2.21	2.33
	30 at > 50%F	2.41	2.41	2.51	2.46	2.46	2.55
	25 at > 50%F	2.63	2.63	2.74	2.48	2.48	2.72
	30 at ≤ 50%F	2.39	2.39	2.51	2.21	2.21	2.33
	25 at ≤ 50%F	2.57	2.57	2.66	2.39	2.39	2.50
FHOOS	100	1.49	1.50	---	1.43	1.44	---
	75	1.62	1.62	---	1.56	1.57	---
	65	1.69	1.69	---	1.62	1.62	---
	50	1.92	1.92	---	1.79	1.79	---
	50	1.95	1.95	---	1.80	1.80	---
	40	2.17	2.17	---	2.02	2.02	---
	30	2.51	2.51	---	2.33	2.33	---
	30 at > 50%F	2.51	2.51	---	2.55	2.55	---
	25 at > 50%F	2.74	2.74	---	2.72	2.72	---
	30 at ≤ 50%F	2.51	2.51	---	2.33	2.33	---
	25 at ≤ 50%F	2.66	2.66	---	2.50	2.50	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR/FHOOS is supported for the BOC to End of Coast limits.

Table 4.3 MCPR$_P$ Limits for All Fuel Types: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
Base Case	100	1.49	1.50	1.52	1.42	1.43	1.45
	75	1.60	1.60	1.64	1.53	1.53	1.57
	65	1.66	1.66	1.73	1.59	1.59	1.63
	50	1.87	1.87	---	1.74	1.74	---
	50	1.95	1.95	1.96	1.79	1.80	1.82
	40	2.10	2.10	2.21	1.95	1.95	2.05
	30	2.43	2.43	2.55	2.23	2.23	2.35
	30 at > 50%F	2.43	2.43	2.55	2.46	2.46	2.55
	25 at > 50%F	2.63	2.63	2.74	2.48	2.48	2.72
	30 at ≤ 50%F	2.43	2.43	2.55	2.23	2.23	2.35
	25 at ≤ 50%F	2.57	2.57	2.66	2.39	2.39	2.50
TBVOOS	100	1.55	1.56	1.57	1.46	1.47	1.48
	75	1.66	1.66	1.68	1.57	1.57	1.61
	65	1.72	1.72	1.76	1.63	1.63	1.67
	50	1.88	1.88	---	1.76	1.76	---
	50	1.95	1.95	1.96	1.79	1.80	1.82
	40	2.11	2.11	2.21	1.95	1.95	2.05
	30	2.43	2.43	2.55	2.23	2.23	2.35
	30 at > 50%F	3.01	3.01	3.10	2.91	2.91	3.01
	25 at > 50%F	3.33	3.33	3.42	3.15	3.15	3.24
	30 at ≤ 50%F	2.82	2.82	2.93	2.60	2.60	2.72
	25 at ≤ 50%F	3.19	3.19	3.33	2.97	2.97	3.11
FHOOS	100	1.51	1.52	---	1.45	1.45	---
	75	1.64	1.64	---	1.56	1.57	---
	65	1.73	1.73	---	1.63	1.63	---
	50	---	---	---	---	---	---
	50	1.96	1.96	---	1.82	1.82	---
	40	2.21	2.21	---	2.05	2.05	---
	30	2.55	2.55	---	2.35	2.35	---
	30 at > 50%F	2.55	2.55	---	2.55	2.55	---
	25 at > 50%F	2.74	2.74	---	2.72	2.72	---
	30 at ≤ 50%F	2.55	2.55	---	2.35	2.35	---
	25 at ≤ 50%F	2.66	2.66	---	2.50	2.50	---
PLUOOS	100	1.49	1.50	1.52	1.42	1.43	1.45
	75	1.60	1.60	1.64	1.53	1.53	1.57
	65	1.86	1.86	1.86	1.71	1.72	1.72
	50	---	---	---	---	---	---
	50	1.95	1.95	1.96	1.79	1.81	1.82
	40	2.10	2.10	2.21	1.95	1.95	2.05
	30	2.43	2.43	2.55	2.23	2.23	2.35
	30 at > 50%F	2.43	2.43	2.55	2.46	2.46	2.55
	25 at > 50%F	2.63	2.63	2.74	2.48	2.48	2.72
	30 at ≤ 50%F	2.43	2.43	2.55	2.23	2.23	2.35
	25 at ≤ 50%F	2.57	2.57	2.66	2.39	2.39	2.50

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.3 MCPR$_P$ Limits for All Fuel Types: Nominal Scram Time Basis (*continued*)*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
TBVOOS FHOOS	100	1.57	1.57	---	1.48	1.48	---
	75	1.68	1.68	---	1.60	1.61	---
	65	1.76	1.76	---	1.67	1.67	---
	50	---	---	---	---	---	---
	50	1.96	1.96	---	1.82	1.82	---
	40	2.21	2.21	---	2.05	2.05	---
	30	2.55	2.55	---	2.35	2.35	---
	30 at > 50%F	3.10	3.10	---	3.01	3.01	---
	25 at > 50%F	3.42	3.42	---	3.24	3.24	---
	30 at ≤ 50%F	2.93	2.93	---	2.72	2.72	---
	25 at ≤ 50%F	3.33	3.33	---	3.11	3.11	---
TBVOOS PLUOOS	100	1.55	1.56	1.57	1.46	1.47	1.48
	75	1.66	1.66	1.68	1.57	1.57	1.61
	65	1.86	1.86	1.86	1.71	1.72	1.72
	50	---	---	---	---	---	---
	50	1.95	1.95	1.96	1.79	1.81	1.82
	40	2.11	2.11	2.21	1.95	1.95	2.05
	30	2.43	2.43	2.55	2.23	2.23	2.35
	30 at > 50%F	3.01	3.01	3.10	2.91	2.91	3.01
	25 at > 50%F	3.33	3.33	3.42	3.15	3.15	3.24
	30 at ≤ 50%F	2.82	2.82	2.93	2.60	2.60	2.72
	25 at ≤ 50%F	3.19	3.19	3.33	2.97	2.97	3.11
FHOOS PLUOOS	100	1.51	1.52	---	1.45	1.45	---
	75	1.64	1.64	---	1.56	1.57	---
	65	1.86	1.86	---	1.71	1.72	---
	50	---	---	---	---	---	---
	50	1.96	1.96	---	1.82	1.82	---
	40	2.21	2.21	---	2.05	2.05	---
	30	2.55	2.55	---	2.35	2.35	---
	30 at > 50%F	2.55	2.55	---	2.55	2.55	---
	25 at > 50%F	2.74	2.74	---	2.72	2.72	---
	30 at ≤ 50%F	2.55	2.55	---	2.35	2.35	---
	25 at ≤ 50%F	2.66	2.66	---	2.50	2.50	---
TBVOOS FHOOS PLUOOS	100	1.57	1.57	---	1.48	1.48	---
	75	1.68	1.68	---	1.60	1.61	---
	65	1.86	1.86	---	1.71	1.72	---
	50	---	---	---	---	---	---
	50	1.96	1.96	---	1.82	1.82	---
	40	2.21	2.21	---	2.05	2.05	---
	30	2.55	2.55	---	2.35	2.35	---
	30 at > 50%F	3.10	3.10	---	3.01	3.01	---
	25 at > 50%F	3.42	3.42	---	3.24	3.24	---
	30 at ≤ 50%F	2.93	2.93	---	2.72	2.72	---
	25 at ≤ 50%F	3.33	3.33	---	3.11	3.11	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.4 MCPR$_P$ Limits for All Fuel Types: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
Base Case	100	1.51	1.51	1.54	1.44	1.44	1.46
	75	1.61	1.61	1.66	1.54	1.54	1.57
	65	1.70	1.70	1.77	1.60	1.60	1.64
	50	1.91	1.91	---	1.77	1.77	---
	50	1.96	1.96	1.99	1.80	1.82	1.84
	40	2.14	2.14	2.24	1.98	1.98	2.07
	30	2.47	2.47	2.59	2.25	2.25	2.37
	30 at > 50%F	2.47	2.47	2.59	2.46	2.46	2.55
	25 at > 50%F	2.63	2.63	2.74	2.48	2.48	2.72
	30 at ≤ 50%F	2.47	2.47	2.59	2.25	2.25	2.37
	25 at ≤ 50%F	2.57	2.57	2.66	2.39	2.39	2.50
TBVOOS	100	1.57	1.57	1.61	1.47	1.48	1.49
	75	1.67	1.67	1.70	1.58	1.58	1.61
	65	1.73	1.73	1.78	1.64	1.64	1.68
	50	1.92	1.92	---	1.78	1.78	---
	50	1.96	1.96	2.00	1.80	1.82	1.85
	40	2.15	2.15	2.24	1.98	1.98	2.08
	30	2.47	2.47	2.59	2.26	2.26	2.37
	30 at > 50%F	3.01	3.01	3.10	2.91	2.91	3.01
	25 at > 50%F	3.33	3.33	3.42	3.15	3.15	3.24
	30 at ≤ 50%F	2.82	2.82	2.93	2.60	2.60	2.72
	25 at ≤ 50%F	3.19	3.19	3.33	2.97	2.97	3.11
FHOOS	100	1.53	1.53	---	1.46	1.46	---
	75	1.66	1.66	---	1.57	1.57	---
	65	1.77	1.77	---	1.64	1.64	---
	50	---	---	---	---	---	---
	50	1.99	1.99	---	1.84	1.84	---
	40	2.24	2.24	---	2.07	2.07	---
	30	2.59	2.59	---	2.37	2.37	---
	30 at > 50%F	2.59	2.59	---	2.55	2.55	---
	25 at > 50%F	2.74	2.74	---	2.72	2.72	---
	30 at ≤ 50%F	2.59	2.59	---	2.37	2.37	---
	25 at ≤ 50%F	2.66	2.66	---	2.50	2.50	---
PLUOOS	100	1.51	1.51	1.54	1.44	1.44	1.46
	75	1.61	1.61	1.66	1.54	1.54	1.57
	65	1.87	1.87	1.88	1.72	1.75	1.75
	50	---	---	---	---	---	---
	50	1.96	1.96	1.99	1.81	1.82	1.84
	40	2.14	2.14	2.24	1.98	1.98	2.07
	30	2.47	2.47	2.59	2.25	2.25	2.37
	30 at > 50%F	2.47	2.47	2.59	2.46	2.46	2.55
	25 at > 50%F	2.63	2.63	2.74	2.48	2.48	2.72
	30 at ≤ 50%F	2.47	2.47	2.59	2.25	2.25	2.37
	25 at ≤ 50%F	2.57	2.57	2.66	2.39	2.39	2.50

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.4 MCPRP Limits for All Fuel Types: Technical Specification Scram Time Basis (*continued*)*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
TBVOOS FHOOS	100	1.58	1.58	---	1.49	1.49	---
	75	1.70	1.70	---	1.61	1.61	---
	65	1.78	1.78	---	1.68	1.68	---
	50	---	---	---	---	---	---
	50	2.00	2.00	---	1.85	1.85	---
	40	2.24	2.24	---	2.08	2.08	---
	30	2.59	2.59	---	2.37	2.37	---
	30 at > 50%F	3.10	3.10	---	3.01	3.01	---
	25 at > 50%F	3.42	3.42	---	3.24	3.24	---
	30 at ≤ 50%F	2.93	2.93	---	2.72	2.72	---
	25 at ≤ 50%F	3.33	3.33	---	3.11	3.11	---
TBVOOS PLUOOS	100	1.57	1.57	1.61	1.47	1.48	1.49
	75	1.67	1.67	1.70	1.58	1.58	1.61
	65	1.87	1.87	1.88	1.72	1.75	1.75
	50	---	---	---	---	---	---
	50	1.96	1.96	2.00	1.81	1.82	1.85
	40	2.15	2.15	2.24	1.98	1.98	2.08
	30	2.47	2.47	2.59	2.26	2.26	2.37
	30 at > 50%F	3.01	3.01	3.10	2.91	2.91	3.01
	25 at > 50%F	3.33	3.33	3.42	3.15	3.15	3.24
	30 at ≤ 50%F	2.82	2.82	2.93	2.60	2.60	2.72
	25 at ≤ 50%F	3.19	3.19	3.33	2.97	2.97	3.11
FHOOS PLUOOS	100	1.53	1.53	---	1.46	1.46	---
	75	1.66	1.66	---	1.57	1.57	---
	65	1.87	1.87	---	1.72	1.75	---
	50	---	---	---	---	---	---
	50	1.99	1.99	---	1.84	1.84	---
	40	2.24	2.24	---	2.07	2.07	---
	30	2.59	2.59	---	2.37	2.37	---
	30 at > 50%F	2.59	2.59	---	2.55	2.55	---
	25 at > 50%F	2.74	2.74	---	2.72	2.72	---
	30 at ≤ 50%F	2.59	2.59	---	2.37	2.37	---
	25 at ≤ 50%F	2.66	2.66	---	2.50	2.50	---
TBVOOS FHOOS PLUOOS	100	1.58	1.58	---	1.49	1.49	---
	75	1.70	1.70	---	1.61	1.61	---
	65	1.87	1.87	---	1.72	1.75	---
	50	---	---	---	---	---	---
	50	2.00	2.00	---	1.85	1.85	---
	40	2.24	2.24	---	2.08	2.08	---
	30	2.59	2.59	---	2.37	2.37	---
	30 at > 50%F	3.10	3.10	---	3.01	3.01	---
	25 at > 50%F	3.42	3.42	---	3.24	3.24	---
	30 at ≤ 50%F	2.93	2.93	---	2.72	2.72	---
	25 at ≤ 50%F	3.33	3.33	---	3.11	3.11	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.5 Startup Operation MCPRp Limits for Table 3.1 Temperature Range 1 for All Fuel Types: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
TBVIS	100	1.51	1.52	1.52	1.45	1.45	1.45
	75	1.64	1.64	1.64	1.56	1.57	1.57
	65	1.86	1.86	1.86	1.71	1.72	1.72
	50	---	---	---	---	---	---
	50	2.15	2.15	2.15	1.99	1.99	1.99
	40	2.45	2.45	2.45	2.28	2.28	2.28
	30	2.86	2.86	2.86	2.65	2.65	2.65
	30 at > 50%F	2.86	2.86	2.86	2.79	2.79	2.79
	25 at > 50%F	3.02	3.02	3.02	3.02	3.02	3.02
	30 at ≤ 50%F	2.86	2.86	2.86	2.65	2.65	2.65
	25 at ≤ 50%F	2.93	2.93	2.93	2.77	2.77	2.77
TBVOOS	100	1.57	1.57	1.57	1.48	1.48	1.48
	75	1.68	1.68	1.68	1.60	1.61	1.61
	65	1.86	1.86	1.86	1.71	1.72	1.72
	50	---	---	---	---	---	---
	50	2.15	2.15	2.15	1.99	1.99	1.99
	40	2.45	2.45	2.45	2.28	2.28	2.28
	30	2.86	2.86	2.86	2.65	2.65	2.65
	30 at > 50%F	3.25	3.25	3.25	3.18	3.18	3.18
	25 at > 50%F	3.58	3.58	3.58	3.40	3.40	3.40
	30 at ≤ 50%F	3.10	3.10	3.10	2.91	2.91	2.91
	25 at ≤ 50%F	3.55	3.55	3.55	3.32	3.32	3.32

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.6 Startup Operation MCPRP Limits for Table 3.1 Temperature Range 2 for All Fuel Types: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
TBVIS	100	1.51	1.52	1.52	1.45	1.45	1.45
	75	1.64	1.64	1.64	1.56	1.57	1.57
	65	1.86	1.86	1.86	1.71	1.72	1.72
	50	---	---	---	---	---	---
	50	2.16	2.16	2.16	2.00	2.00	2.00
	40	2.46	2.46	2.46	2.29	2.29	2.29
	30	2.88	2.88	2.88	2.67	2.67	2.67
	30 at > 50%F	2.88	2.88	2.88	2.80	2.80	2.80
	25 at > 50%F	3.04	3.04	3.04	3.05	3.05	3.05
	30 at ≤ 50%F	2.88	2.88	2.88	2.67	2.67	2.67
	25 at ≤ 50%F	2.95	2.95	2.95	2.79	2.79	2.79
TBVOOS	100	1.57	1.57	1.57	1.48	1.48	1.48
	75	1.68	1.68	1.68	1.60	1.61	1.61
	65	1.86	1.86	1.86	1.71	1.72	1.72
	50	---	---	---	---	---	---
	50	2.16	2.16	2.16	2.00	2.00	2.00
	40	2.46	2.46	2.46	2.29	2.29	2.29
	30	2.88	2.88	2.88	2.67	2.67	2.67
	30 at > 50%F	3.26	3.26	3.26	3.19	3.19	3.19
	25 at > 50%F	3.59	3.59	3.59	3.41	3.41	3.41
	30 at ≤ 50%F	3.12	3.12	3.12	2.92	2.92	2.92
	25 at ≤ 50%F	3.56	3.56	3.56	3.33	3.33	3.33

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.7 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 1 for All Fuel Types: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
TBVIS	100	1.53	1.53	1.54	1.46	1.46	1.46
	75	1.66	1.66	1.66	1.57	1.57	1.57
	65	1.87	1.87	1.88	1.72	1.75	1.75
	50	---	---	---	---	---	---
	50	2.18	2.18	2.18	2.02	2.02	2.02
	40	2.48	2.48	2.48	2.31	2.31	2.31
	30	2.90	2.90	2.90	2.67	2.67	2.67
	30 at > 50%F	2.90	2.90	2.90	2.79	2.79	2.79
	25 at > 50%F	3.02	3.02	3.02	3.02	3.02	3.02
	30 at ≤ 50%F	2.90	2.90	2.90	2.67	2.67	2.67
	25 at ≤ 50%F	2.93	2.93	2.93	2.77	2.77	2.77
TBVOOS	100	1.58	1.58	1.61	1.49	1.49	1.49
	75	1.70	1.70	1.70	1.61	1.61	1.61
	65	1.87	1.87	1.88	1.72	1.75	1.75
	50	---	---	---	---	---	---
	50	2.18	2.18	2.18	2.02	2.02	2.02
	40	2.48	2.48	2.48	2.31	2.31	2.31
	30	2.90	2.90	2.90	2.67	2.67	2.67
	30 at > 50%F	3.25	3.25	3.25	3.18	3.18	3.18
	25 at > 50%F	3.58	3.58	3.58	3.40	3.40	3.40
	30 at ≤ 50%F	3.10	3.10	3.10	2.91	2.91	2.91
	25 at ≤ 50%F	3.55	3.55	3.55	3.32	3.32	3.32

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.8 Startup Operation MCPRp Limits for Table 3.1 Temperature Range 2 for All Fuel Types: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
TBVIS	100	1.53	1.53	1.54	1.46	1.46	1.46
	75	1.66	1.66	1.66	1.57	1.57	1.57
	65	1.87	1.87	1.88	1.72	1.75	1.75
	50	---	---	---	---	---	---
	50	2.19	2.19	2.19	2.03	2.03	2.03
	40	2.50	2.50	2.50	2.32	2.32	2.32
	30	2.92	2.92	2.92	2.69	2.69	2.69
	30 at > 50%F	2.92	2.92	2.92	2.80	2.80	2.80
	25 at > 50%F	3.04	3.04	3.04	3.05	3.05	3.05
	30 at ≤ 50%F	2.92	2.92	2.92	2.69	2.69	2.69
	25 at ≤ 50%F	2.95	2.95	2.95	2.79	2.79	2.79
TBVOOS	100	1.58	1.58	1.61	1.49	1.49	1.49
	75	1.70	1.70	1.70	1.61	1.61	1.61
	65	1.87	1.87	1.88	1.72	1.75	1.75
	50	---	---	---	---	---	---
	50	2.19	2.19	2.19	2.03	2.03	2.03
	40	2.50	2.50	2.50	2.32	2.32	2.32
	30	2.92	2.92	2.92	2.69	2.69	2.69
	30 at > 50%F	3.26	3.26	3.26	3.19	3.19	3.19
	25 at > 50%F	3.59	3.59	3.59	3.41	3.41	3.41
	30 at ≤ 50%F	3.12	3.12	3.12	2.92	2.92	2.92
	25 at ≤ 50%F	3.56	3.56	3.56	3.33	3.33	3.33

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.9 MCPR$_P$ Limits for All Fuel Types: Single Loop Operation for All Scram Times*

Operating Condition	Power (% of rated)	BOC to End of COAST	
		ATRIUM-10	ATRIUM-10XM
RCPOOS FHOOS	100	2.01	2.02
	50	2.01	2.02
	40	2.26	2.09
	30	2.61	2.39
	30 at > 50%F	2.61	2.57
	25 at > 50%F	2.76	2.74
	30 at ≤ 50%F	2.61	2.39
	25 at ≤ 50%F	2.68	2.52
RCPOOS TBVOOS PLUOOS FHOOS	100	2.02	2.02
	50	2.02	2.02
	40	2.26	2.10
	30	2.61	2.39
	30 at > 50%F	3.12	3.03
	25 at > 50%F	3.44	3.26
	30 at ≤ 50%F	2.95	2.74
	25 at ≤ 50%F	3.35	3.13
RCPOOS TBVOOS FHOOS1	100	2.20	2.04
	50	2.20	2.04
	40	2.50	2.33
	30	2.92	2.69
	30 at > 50%F	3.27	3.20
	25 at > 50%F	3.60	3.42
	30 at ≤ 50%F	3.12	2.93
	25 at ≤ 50%F	3.57	3.34
RCPOOS TBVOOS FHOOS2	100	2.21	2.05
	50	2.21	2.05
	40	2.52	2.34
	30	2.94	2.71
	30 at > 50%F	3.28	3.21
	25 at > 50%F	3.61	3.43
	30 at ≤ 50%F	3.14	2.94
	25 at ≤ 50%F	3.58	3.35

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop.

RCPOOS limits are only valid up to 50% rated core power, 50% rated core flow, and an active recirculation drive flow of 17.73 Mlbm/hr.

5 Oscillation Power Range Monitor (OPRM) Setpoint

(Technical Specification 3.3.1.1)

Technical Specification Table 3.3.1.1-1, Function 2f, identifies the OPRM upscale function.

Instrument setpoints are established, such that the reactor will be tripped before an oscillation can grow to the point where the SLMCPR is exceeded. An Option III stability analysis is performed for each reload core to determine allowable OLMCPR's as a function of OPRM setpoint. Analyses consider both steady state startup operation, and the case of a two recirculation pump trip from rated power.

The resulting stability based OLMCPR's are reported in Reference 1. The OPRM setpoint (*sometimes referred to as the Amplitude Trip, S_p*) is selected, such that required margin to the SLMCPR is provided without stability being a limiting event. Analyses are based on cycle specific DIVOM analyses performed per Reference 22. The calculated OLMCPR's are shown in Table 5.1. Review of results shown in COLR Table 4.2 indicates an OPRM setpoint of **1.14** may be used. The successive confirmation count (*sometimes referred to as N_p*) is provided in Table 5.2, per Reference 30.

Table 5.1 OPRM Setpoint Range*

OPRM Setpoint	OLMCPR (SS)	OLMCPR (2PT)
1.05	1.15	1.17
1.06	1.17	1.19
1.07	1.19	1.21
1.08	1.20	1.23
1.09	1.22	1.25
1.10	1.24	1.27
1.11	1.26	1.29
1.12	1.28	1.31
1.13	1.30	1.33
1.14	1.33	1.35
1.15	1.35	1.38

Table 5.2 OPRM Successive Confirmation Count Setpoint

Count	OPRM Setpoint
6	≥ 1.04
8	≥ 1.05
10	≥ 1.07
12	≥ 1.09
14	≥ 1.11
16	≥ 1.14
18	≥ 1.18
20	≥ 1.24

* Extrapolation beyond a setpoint of 1.15 is not allowed

6 APRM Flow Biased Rod Block Trip Settings

(Technical Requirements Manual Section 5.3.1 and Table 3.3.4-1)

The APRM rod block trip setting is based upon References 26 & 27, and is defined by the following:

$SRB \leq (0.66(W-\Delta W) + 61\%)$	Allowable Value
$SRB \leq (0.66(W-\Delta W) + 59\%)$	Nominal Trip Setpoint (NTSP)

where:

SRB	=	Rod Block setting in percent of rated thermal power (3458 MW_t)
W	=	Loop recirculation flow rate in percent of rated
ΔW	=	Difference between two-loop and single-loop effective recirculation flow at the same core flow (ΔW=0.0 for two-loop operation)

The APRM rod block trip setting is clamped at a maximum allowable value of 115% (corresponding to a NTSP of 113%).

7 Rod Block Monitor (RBM) Trip Setpoints and Operability

(Technical Specification Table 3.3.2.1-1)

The RBM trip setpoints and applicable power ranges, based on References 26 & 27, are shown in Table 7.1. Setpoints are based on an HTSP, unfiltered analytical limit of 114%. Unfiltered setpoints are consistent with a nominal RBM filter setting of 0.0 seconds; filtered setpoints are consistent with a nominal RBM filter setting less than 0.5 seconds. Cycle specific CRWE analyses of OLMCPR are documented in Reference 1, superseding values reported in References 26, 27, and 29.

Table 7.1 Analytical RBM Trip Setpoints*

RBM Trip Setpoint	Allowable Value (AV)	Nominal Trip Setpoint (NTSP)
LPSP	27%	25%
IPSP	62%	60%
HPSP	82%	80%
LTSP - unfiltered - filtered	121.7% 120.7%	120.0% 119.0%
ITSP - unfiltered - filtered	116.7% 115.7%	115.0% 114.0%
HTSP - unfiltered - filtered	111.7% 110.9%	110.0% 109.2%
DTSP	90%	92%

As a result of cycle specific CRWE analyses, RBM setpoints in Technical Specification Table 3.3.2.1-1 are applicable as shown in Table 7.2. Cycle specific setpoint analysis results are shown in Table 7.3, per Reference 1.

Table 7.2 RBM Setpoint Applicability

Thermal Power (% Rated)	Applicable MCPR†	Notes from Table 3.3.2.1-1	Comment
> 27% and < 90%	< 1.75	(a), (b), (f), (h)	two loop operation
	< 1.79	(a), (b), (f), (h)	single loop operation
≥ 90%	< 1.43	(g)	two loop operation‡

* Values are considered maximums. Using lower values, due to RBM system hardware/software limitations, is conservative, and acceptable.

† MCPR values shown correspond with, (support), SLMPCR values identified in Reference 1.

‡ Greater than 90% rated power is not attainable in single loop operation.

Table 7.3 Control Rod Withdrawal Error Results

RBM HTSP Analytical Limit	CRWE OLMCPR
Unfiltered	
107	1.35
111	1.36
114	1.36
117	1.41

Results, compared against the base case OLMCPR results of Table 4.2, indicate SLMCPR remains protected for RBM inoperable conditions (i.e., 114% unblocked).

8 Shutdown Margin Limit

(Technical Specification 3.1.1)

Assuming the strongest OPERABLE control blade is fully withdrawn, and all other OPERABLE control blades are fully inserted, the core shall be sub-critical and meet the following minimum shutdown margin:

SDM > 0.38% dk/k

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B 3.5-5	0	Initial
B 3.5-6	Revision 70	04-24-2009
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BASES

ACTIONS (continued)

C.1 and C.2

When required control room indication channels are inoperable but the redundant channels for the parameters are still OPERABLE, the required control room indication channels must be returned to OPERABLE status in 30 days (Required Action C.2). However, if both redundant channels for one or more of the associated parameters are not indicating in the control room, the 30 day allowed out of service time is not acceptable and one of the required control room indication channels for each associated parameter must be returned to OPERABLE status in 7 days (Required Action C.1).

D.1, D.2, and D.3

When the Tailpipe Thermocouple Temperature and Acoustic Monitor is inoperable for one or more Main Steam Relief Valves (MSRVs), the torus temperature must be observed once per 12 hours to observe any unexplained temperature increase which might be indicative of an open relief valve (Required Action D.1) and control room indication by either the Tailpipe Thermocouple Temperature or Acoustic Monitor must be returned to OPERABLE status for each relief valve in 30 days (Required Action D.2). The condition must be entered into the Corrective Action Program within 24 hours if control room indication is not restored in 30 days (Required Action D.3).

E.1.1 and E.1.2

When the Wide Range Gaseous Effluent Radiation Monitor and Recorder instrument channel is inoperable, either the inoperable channel must be returned to OPERABLE status in 72 hours (Required Action E.1.1), or the preplanned alternate method of monitoring the parameter must be initiated (Required Action E.1.2). A note is provided to indicate that Required Actions E.1.1 and E.1.2 are not applicable when in MODES 4 and 5.

E.2

The condition must be entered into the Corrective Action Program within 24 hours after the Wide Range Gaseous Effluent Radiation Monitor and Recorder instrument channel has been inoperable for 7 days.

BASES

LCO 3.3.7

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation dose to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public.

APPLICABILITY

At all times.

ACTIONS

A.1

Seven days to obtain replacement parts and repair is reasonable for these instruments given the requirements to be available for radiological emergencies.

B.1

If the instruments cannot be repaired in the allowed time frame, the condition must be entered into the Corrective Action Program within 24 hours.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.3.7.1

Daily checks of these parameters assures prompt replacement/repair of inoperable or questionable instruments.

TSR 3.3.7.2

Surveillance requirement times are based on equipment reliability and engineering judgment and conservatively set to provide adequate assurance of instrument performance.

BASES

ACTIONS

B.1 (continued)

conditions at the locations of these instruments to determine the impact of the vibratory ground motion on structures and equipment in these locations following any required unit shutdown after a seismic event.

C.1

If any Required Action and associated Completion Time of Condition A or B is not met, the failure to restore the inoperable seismic monitoring instrumentation within the required Completion Time, must be entered into the Corrective Action Program within 24 hours.

BASES

LCO 3.3.11

The primary containment hydrogen monitoring instrumentation allows the operators to detect trends in hydrogen concentration to diagnose the course of beyond design basis accidents. High hydrogen concentration is measured, continuously recorded, and displayed in the control room by a single instrument channel. The analyzer has the capability for sampling both the drywell and the suppression chamber. LCO 3.3.11 requires the primary containment hydrogen analyzer be OPERABLE.

APPLICABILITY

The primary containment hydrogen analyzer is required to be OPERABLE when primary containment is inerted, except as allowed by the relaxations during startup and shutdown addressed below. The primary containment must be inert in MODE 1, since this is the condition with the highest probability of an event that could produce hydrogen.

Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and de-inerted as soon as possible in the plant shutdown. As long as reactor power is < 15% RTP, the potential for an event that generates significant hydrogen is low and the primary containment need not be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a startup, or within the last 24 hours before a shutdown, is low enough that these "windows," when the primary containment is not inerted, are also justified. The 24 hour time period is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.

ACTIONS

A.1

Seven days to restore the primary containment hydrogen analyzer capability is reasonable given the requirement to be available for use in diagnosing beyond design basis events.

TR 3.7 PLANT SYSTEMS

TR 3.7.4 Snubbers

BASES

BACKGROUND	Snubbers are designed to prevent unrestrained component or system motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to the component or system as a result of a seismic or other event initiating dynamic loads. An inoperable snubber (ex: failed by locked in place) may cause damage to the supported component or system from normal operating modes such as thermal operation. It is, therefore, required that all snubbers required to protect the primary coolant system or any other safety related component or system be OPERABLE during MODES 1, 2, 3, 4, and 5. The Technical Requirements Manual (TRM) action statements establish allowable outage times for components or systems addressed by the Limiting Conditions of Operation (LCO) for snubbers. These time limits are applicable when a snubber must be removed from service to perform required surveillance tests. For snubbers, the allowable outage time is 72 hours. Table 3.7.4-1, "Visual Examination Table" is published in ASME OM Code Subsection ISTD, Table ISTD-4252-1, and is based on previous table issued to all nuclear plant license holders by the Nuclear Regulatory Commission (NRC) under Generic Letter (GL) 90-09, which was added to the old Technical Specification and approved by the NRC under Technical Specification Amendment 210.
APPLICABLE SAFETY ANALYSIS	During MODES 1, 2, 3, 4, and 5 snubbers may be removed from service for functional surveillance testing to satisfy the required testing interval. When a snubber is removed from a component or system, the snubber is declared inoperable since it cannot perform its intended function while removed. This type of inoperability is not a failure. Examples of snubber failures include locked in place, high drag force, does not activate, no lockup, high lockup, low lockup, high bleed, no bleed, and damage to the snubber hardware. If a snubber is determined to be inoperable based on failure to meet the functional test acceptance criteria, an engineering evaluation is performed to establish whether, during

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

A note is provided to define the term "code snubber," as it is used in this Technical Requirements, and it shall mean snubbers that are identified by BFN ASME Code IST program as ASME Class 1, 2, or 3 equivalent snubbers. It shall also mean those BFN safety-related snubbers that are not identified as ASME Class 1, 2, or 3 equivalent, but are treated as such.

A note is provided to indicate that each code snubber, identified by those snubbers listed in plant procedures, shall be demonstrated OPERABLE by performance of the following inservice examination and test program requirements, which are derived from ASME OM Code Subsection ISTD.

An additional note is provided to indicate that in this Technical Requirement, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

The augmented inservice inspection program includes the following.

All code snubbers are visually inspected, pin to pin, inclusive, for overall integrity and OPERABILITY. The visual inspection will include verification that no visible indications of damage, leakage, corrosion, degradation, binding, misalignment, deformation, or other external characteristics that may indicate impaired OPERABILITY are present, verification that proper attachment of the snubber to the component or system and structures exist, and no loose or missing fasteners exist. In addition, hydraulic fluid level is verified. The removal of insulation or the verification of torque values for threaded fasteners is not required for visual inspections.

The visual inspection frequency is based upon maintaining a constant level of snubber protection. In accordance with Table 3.7.4-1, the number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25 percent) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)	When the cause of the rejection of a snubber in a visual inspection is clearly established and remedied for that snubber and for any other snubber(s) that may be generically susceptible and OPERABILITY verified by inservice functional testing, if applicable, that snubber(s) may be reclassified as OPERABLE. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to the rejected snubber, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

vibration. The inspection population or category may be established based on design features, or installed conditions which may be expected to be generic. Each of these inspection populations or categories may be inspected and tested separately unless an engineering analysis indicates the inspection population or category is improperly constituted. All suspect snubbers are subject to inspection and testing regardless of inspection population or category.

To verify snubber OPERABILITY, a functional test shall be performed once per fuel cycle.

These tests will include stroking of the snubbers to verify proper movement, activation, and bleed or release. Ten percent represents an adequate sample for such tests. Observed failures on these samples will require a failure analysis and testing of additional units. If the failure analysis results in the determination that the failure of a snubber to activate or to stroke (i.e., seized components) is the result of a manufacture or design deficiency, all snubbers subject to the same defect shall be functionally tested. Also, an engineering evaluation shall be performed to determine the effects on the supported component or system during the previous unit operating cycle with the snubber inoperable, and to ensure it remains capable of meeting its designed service. A thorough visual inspection of the snubber threaded attachments to the component or system and the anchorage will be made in conjunction with all required functional tests. The stroke setting of the snubbers selected for functional testing also will be verified.

Exemption from Visual Inspection or Functional Tests:

Permanent or other exemptions from visual inspections and/or functional testing for individual snubbers may be granted by the Nuclear Regulatory Commission if a justifiable basis for exemption is presented and if applicable snubber life destructive testing was performed to qualify the snubber OPERABILITY for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall continue to be listed in the plant instructions with footnotes indicating the extent of the exemptions.

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

Snubber Service Life Program:

The service life of snubbers may be extended based on an evaluation of the records of functional tests, maintenance history, and environmental conditions to which the snubbers have been exposed.

The following will be implemented by the augmented inservice inspection program:

TSR 3.7.4.1

Visual Inspections:

At BFN, code snubbers are visually examined as one population or category, regardless of accessibility, in accordance with the schedule and interval determined by Table 3.7.4-1, which is Table ISTD-4252-1. The first inspection interval determined using Table 3.7.4-1 criteria shall be based on the previous inspection interval as established by the requirements in effect before Revision 007 of these Technical Requirements was issued.

Visual Inspection Acceptance Criteria:

Visual inspections shall verify that:

a) The snubber has no visible indications of damage, leakage, corrosion, degradation, or other external characteristics that may indicate impaired OPERABILITY, pin to pin, inclusive.

b) Fasteners for the attachment of the snubber functional.

c) No indications of binding, misalignment, or deformation of the snubber.

d) Hydraulic snubber fluid is at the recommended level and vented reservoir is oriented such that fluid can gravitate to snubber body.

e) The absence of weld arc strikes, paint, weld slag, adhesive, or other deposits on piston rod or support cylinder that could result in unacceptable snubber performance.

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.7.4.1 (continued)

f) Snubber spherical bearing is fully engaged in attachment lug.

g) Snubber position setting is adequate.

Snubbers which appear inoperable as a result of visual inspection shall be classified unacceptable. Snubbers confirmed as unacceptable snubbers are adjusted, repaired, modified, or replaced, and counted in the determination of the subsequent examination interval in accordance with Table 3.7.4-1, regardless if the affected snubber is functionally tested in the as-found condition and determined OPERABLE per the functional test acceptance criteria of TSR 3.7.4.2.

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.4.1 (continued)

A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable.

Additionally, snubbers attached to sections of safety-related systems that have experienced unexpected potentially damaging transients since the last inspection period shall be evaluated for the possibility of concealed damage and functionally tested, if applicable, to confirm OPERABILITY. Snubbers which have been made inoperable as the result of unexpected transients, isolated damage, or other random events, when the provisions of TSR 3.7.4.5 and 3.7.4.6 have been met and any other appropriate corrective action implemented, shall not be counted in determining the next visual inspection interval.

TSR 3.7.4.2

Functional Test Schedule, Lot Size, and Composition:

Once per fuel cycle, a representative sample of 10% of the total of each type of code snubbers in use in the plant shall be functionally tested either in place or in a bench test. The sample population is rounded up to the next whole integer. Safety-related snubbers that are not ASME Class 1, 2, or 3 equivalent snubbers shall not be included in the snubber population when selecting the initial or additional samples.

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.4.2 (continued)

As practical, the representative sample selected for functional testing shall include representation from the Defined Test Plan Group (DTPG) based on the significant features (i.e., the various designs, configurations, operating environments, and the range of size and capacity of snubbers within the types) and based on the ratio of the number of snubbers of each significant feature, to the total number of snubbers in the DTPG. The sample shall be generally representative as specified in ISTD-5311(a), but may also be selected from snubbers concurrently scheduled for seal replacement or other similar activity related to service life monitoring. The snubbers shall be tested on a generally rotational basis to coincide with the service life monitoring. The representative sample should be weighed to include more snubbers from severe service areas such as near heavy equipment. After testing, at the time of reinstallation, the snubber shall meet visual examination attributes described in ISTD-4110(a), -4110(c), -4110(d), and -4110(e). The stroke setting shall be verified.

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.4.2 (continued)

Functional Test Acceptance Criteria:

The snubber functional test shall verify that:

a. Activation (restraining action) is achieved in both tension and compression within the specified range of velocity or acceleration.

b. Snubber bleed, or release where required, is present in both compression and tension within the specified range.

c. For mechanical snubbers, the drag force is within the specified limits, in tension and compression.

d. For hydraulic snubbers, if required to verify proper assembly, drag force is within specific limits in tension and compression.

e. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

f. Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

TSR 3.7.4.3

Functional Test Failure Analysis and Additional Test Lots:

A failure analysis shall be performed for each failure to meet the functional test acceptance criteria to determine the cause of the failure. The result of this analysis shall be used, if applicable, in selecting snubbers to be tested in the subsequent lot in an effort to determine the OPERABILITY of other snubbers which may be subject to the same failure mode. Selection of snubbers for future testing may also be based on the failure analysis.

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.4.3 (continued)

For each snubber that does not meet the functional test acceptance criteria, an additional lot equal to 5 percent of the subject snubber DTPG shall be functionally tested. When establishing additional sample testing, failure analysis results of unacceptable snubbers are to be used to determine if establishing an FMG is appropriate. Additional lot (i.e., sample) population is rounded up to next whole integer. Rounding up satisfies ISTD-5312 requirement that additional samples shall be at least one-half the size of the initial sample from the DTPG. Additional samples selected from a DTPG follow the same composition as the original sample. However, additional samples from FMGs are random samples. Testing shall continue until no additional inoperable snubbers are found within subsequent lots or all snubbers of the original functional test type have been tested or all suspect snubbers identified by the failure analysis have been tested, as applicable.

The discovery of loose or missing attachment fasteners will be evaluated to determine whether the cause may be localized or generic. The result of the evaluation will be used to select other suspect snubbers for verifying the attachment fasteners, as applicable.

TSR 3.7.4.4

(Deleted by TRM Revision 125.)

TSR 3.7.4.5

Functional Test Failure - Supported Component or System Analysis:

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the component or system which is restrained by the snubber(s) due to not meeting their functional test acceptance criteria. The purpose of this engineering evaluation shall be to determine if the component or system restrained by the snubber(s) was adversely affected by the inoperability of the snubber(s), and in order to ensure that the restrained component or system remains capable of meeting the designed service.

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)	<u>TSR 3.7.4.6</u> Functional Testing of Repaired and Spare Snubbers: Snubbers which fail the visual inspection or the functional test acceptance criteria shall be adjusted, repaired, modified, scrapped, or replaced. Replacement snubbers and snubbers having repairs, adjustments, or modifications which might affect the functional test results shall meet the functional test acceptance criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the functional test must have been performed within 12 months prior to being installed in the unit.
REFERENCES	1. BFN Technical Specifications (version prior to standardized version)

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ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. As required by Required Action A.1 and referenced in Table 3.3.5-1.	D.1	Monitor torus temperature to observe any unexplained temperature increase which might be indicative of an open relief valve.	Once per 12 hours
	<u>AND</u>		
	D.2	Restore control room indication by either the Tailpipe Thermocouple Temperature or Acoustic Monitor to OPERABLE status for each relief valve.	30 days
	<u>AND</u>		
	D.3	When inoperable for more than 30 days, enter the condition into the Corrective Action Program.	24 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action A.1 and referenced in Table 3.3.5-1.	-------------------NOTE---------------- Required Actions E.1.1 and E.1.2 are not applicable when in MODES 4 and 5. --	
	E.1.1 Restore required control room indication channel to OPERABLE status.	72 hours
	<u>OR</u>	
	E.1.2 Initiate the preplanned alternate method of monitoring the parameter.	72 hours
	<u>AND</u>	
	E.2 When inoperable for more than seven days, enter the condition into the Corrective Action Program.	24 hours

(continued)

TR 3.3 INSTRUMENTATION

TR 3.3.7 Meteorological Monitoring Instrumentation

LCO 3.3.7 The meteorological monitoring instrumentation listed in Table 3.3.7-1 shall be OPERABLE.

APPLICABILITY: At all times

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The number of OPERABLE meteorological monitoring channels less than required by Table 3.3.7-1.	A.1 Restore inoperable channel(s) to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Enter the condition into the Corrective Action Program.	24 hours

TR 3.3 INSTRUMENTATION

TR 3.3.8 Seismic Monitoring Instrumentation

LCO 3.3.8 The seismic monitoring instruments listed in Table 3.3.8-1 shall be OPERABLE.

APPLICABILITY: At all times

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more seismic monitoring instruments in Panel 1-PNLA-9-44 or foundation instrument BFN-0-ACGR-052-0001 on the Unit 1 Reactor Building Base Slab inoperable.	A.1 Restore inoperable instrument(s) to an OPERABLE status.	30 days
B. One or more seismic monitoring instruments BFN-0-ACGR-052-0002 or BFN-0-ACGR-052-0003 inoperable.	B.1 Restore inoperable instrument(s) to an OPERABLE status.	60 days
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Enter the condition into the Corrective Action Program.	24 hours

---NOTES---

1. As used in this Technical Requirement, "code snubber" shall mean snubbers that are identified by BFN ASME Code IST Program as ASME Class 1, 2, and 3 equivalent snubbers. It shall also mean those BFN safety-related snubbers that are not identified as ASME Class 1, 2, and 3 equivalent, but are treated as such.

2. Each code snubber, identified by those snubbers listed in plant procedures, shall be demonstrated OPERABLE by performance of the following inservice examination and test program requirements, which is derived from ASME OM Code Subsection ISTD.

3. As used in this Technical Requirement, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

4. As used in this Technical Requirement, "population or category" shall mean the total number of snubbers being visually inspected as a lot.

TECHNICAL SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
TSR 3.7.4.1	---NOTE--- At BFN, code snubbers are visually examined as one population or category, regardless of accessibility, in accordance with the schedule and interval determined by Table 3.7.4-1, which is Table ISTD-4252-1. The first inspection interval determined using Table 3.7.4-1 criteria shall be based on the previous inspection interval established by the requirements in effect before Revision 005 of these Technical Requirements were issued. --- Perform visual examination of required snubber(s) based on the acceptance criteria of plant visual examination procedures and the frequency based on Table 3.7.4-1, which are both based on ASME OM Code, Subsection ISTD. Visual examination shall confirm: a. No visible indications of damage, leakage, corrosion, degradation or other external characteristics that may indicate impaired OPERABILITY, pin to pin, inclusive.	In accordance with Table 3.7.4.1.

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.7.4.1 (continued)	b. Fasteners for the attachment of the snubber are functional. c. No indication of binding, misalignment, or deformation of the snubber. d. Hydraulic snubber fluid is at the recommended level and that vented reservoir is oriented such that fluid can gravitate to snubber body. e. The absence of weld arc strikes, paint, weld slag, adhesive, or other deposits on piston rod or support cylinder that could result in unacceptable snubber performance. f. Snubber spherical bearing is fully engaged in attachment lug. g. Snubber position setting is adequate Snubbers which appear inoperable as a result of visual inspection shall be classified unacceptable. Snubbers confirmed as unacceptable snubbers are adjusted, repaired, modified, or replaced, and counted in determination of the subsequent examination interval in accordance with Table 3.7.4-1, regardless if the affected snubber is functionally tested in the as-found condition and determined OPERABLE per the criteria of TSR 3.7.4.2. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the system or train shall be declared inoperable.	

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.7.4.1 (continued)	Additionally, snubbers attached to sections of safety related systems that have experienced unexpected potentially damaging transients since the last inspection period shall be evaluated for the possibility of concealed damage and functionally tested, if applicable, to confirm OPERABILITY. Snubbers which have been made inoperable as the result of unexpected transients, isolated damage, or other random events, when the provisions of TSR 3.7.4.5. and TSR 3.7.4.6 have been met and any other appropriate corrective action implemented, shall not be counted in determining the next visual inspection interval.	

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.7.4.2	Perform an in-place or bench functional test of a representative sample of 10% of the total of each type of code snubber(s). Sample population is rounded up to next whole integer. Safety-related snubbers that are not ASME Class 1, 2, or 3 equivalent snubbers shall not be included in the snubber population when selecting the initial or additional samples. a. As practical, the representative sample selected for functional testing shall include representation from the Defined Test Plan Group (DTPG) based on the significant features (i.e., the various designs, configurations, operating environments, and the range of size and capacity of snubbers within the types) and based on the ratio of the number of snubbers of each significant feature, to the total number of snubbers in the DTPG. b. The sample shall be generally representative as specified in ISTD-5311 (a), but may also be selected from snubbers concurrently scheduled for seal replacement or other similar activity related to service life monitoring. The snubbers shall be tested on a generally rotational basis to coincide with the service life monitoring. c. The representative sample should be weighed to include more snubbers from severe service areas such as near heavy equipment. d. After testing, at the time of reinstallation, the snubber shall meet visual examination attributes as described in ISTD-4110(a), -4110(c), -4110(d), and -4110(e). The stroke setting shall be verified.	In accordance with Inservice Testing Program

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.7.4.2 (continued)	Functional Test Acceptance Criteria: The snubber functional test shall verify that: a. Activation (restraining action) is achieved in both tension and compression within the specified range of velocity or acceleration. b. Snubber bleed or release, where required, is present in both compression and tension within the specified range. c. For mechanical snubbers, the drag force is within the specified limits, in tension and compression. d. For hydraulic snubbers, if required to verify proper assembly, drag force is within specific limits in tension and compression. e. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified. f. Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.	

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TSR 3.7.4.3	A failure analysis shall be made of each failure to meet the functional test acceptance criteria of TSR 3.7.4.2 to determine the cause of the failure. The result of this analysis shall be used, if applicable, in selecting snubbers to be tested in the subsequent lot in an effort to determine the OPERABILITY of other snubbers which may be subject to the same failure mode. Selection of snubbers for future testing may also be based on the failure analysis. For each failed snubber, perform in-place or bench functional test on an additional lot equal to 5% of the subject snubber DTPG. When establishing additional sample testing, failure analysis results of unacceptable snubbers are to be used to determine if establishing a FMG is appropriate. Additional lot (i.e., sample) population is rounded up to next whole integer. Rounding up satisfies ISTD-5312 requirements that additional samples shall be at least one-half the size of the initial sample from that DTPG. Additional samples selected from DTPGs follow the composition of the original sample. However, additional samples selected from FMGs are random samples. Testing shall continue until no additional inoperable snubbers are found within subsequent lots or all snubbers of the original test type are tested or all suspect snubbers identified by the failure analysis have been tested, as applicable. The functional test criteria shall be as specified in TSR 3.7.4.2.	Once for each discovery of snubber failure to meet functional test acceptance criteria
	Prior to functional testing the discovery of loose or missing attachment fasteners will be evaluated to determine whether the cause may be localized or generic. The result of the evaluation will be used to select other suspect snubbers for verifying the attachment fasteners, as applicable.	Once for each discovery of loose or missing attachment fasteners

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TSR 3.7.4.4	(Deleted by TRM Revision 125.)	
TSR 3.7.4.5	Perform an engineering evaluation on the component or system which is restrained by the snubber(s) found inoperable due to not meeting their functional test acceptance criteria as specified in TSR 3.7.4.2.	Once for each discovery of an inoperable snubber
TSR 3.7.4.6	Verify replacement snubbers and snubbers having repairs, adjustment, or modifications which might affect the functional test results meet the test criteria of TSR 3.7.4.2. a. These snubbers shall have met the acceptance criteria subsequent to their most recent service; and b. The functional test must have been performed within the 12 months prior to being installed in the unit.	Once prior to installation in the unit for each replaced, repaired, adjusted, or modified snubber where functional test results might be affected

Table 3.7.4-1
Visual Examination Table
(Ref Table ISTD-4252-1)

	NUMBER OF UNACCEPTABLE SNUBBERS		
Population or Category (Note 1)	Column A Extend Interval (Notes 2 and 3)	Column B Repeat Interval (Notes 2, 4 and 5)	Column C Reduce Interval to 2/3 (Notes 2, 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or more	29	56	109

Note 1: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. The next lower integer shall be used when interpolation results in a fraction.

Note 2: The basic interval shall be the normal fuel cycle up to 24 months. The examination interval may be as great as twice, the same, or as small as fractions of the previous interval as required by the following Notes. The examination interval may vary + / - 25% of the current interval.

Table 3.7.4-1
Visual Examination Table
(Ref Table ISTD-4252-1)
(continued)

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months. In that case, the next visual examination according to the previous interval may be skipped.

Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is less than the number in Column C, but greater than the number is Column B, the next interval shall be reduced to two-thirds of the previous examination interval or, in accordance with the interpolation between Columns B and C, in proportion to the exact number of unacceptable snubbers.

Note 6: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval.

The provisions of TSR 3.0.1, 3.0.2, and 3.0.3 are applicable for all inspection intervals up to and including 48 months.

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QA Document
Pages Affected: all
BFE-3843, Revision 3

TVA NPG

Reactor Engineering and Fuels - BWRFE
1101 Market Street, Chattanooga, TN 37402

Browns Ferry Unit 2 Cycle 19

Core Operating Limits Report, (105% OLTP)

TVA-COLR-BF2C19 Revision 3 (Final)
(Revision Log, Page v)

July 2015

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Approved: ______________ Date: 7-22-15
Plant Manager

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Total Number of Pages = 54 (*including review cover sheet*)

List of Tables

List of Figures

Revision Log

Number	Page	Description
0-R3		No Changes to Section 2; Sections 5 through 8.
1-R3	ix	Updated References 1 & 4.
2-R3	xi	Removed References 31 & 32 as they are no longer applicable due to Reference 1 update.
3-R3	1	Revised Section 1.1.1; identify new basis of revised reload safety analysis report.
4-R3	8	Revised Section 3.2; removed cross references coincident with change 2-R3. Also, revised Section 3.2.1; removed last sentence as items no longer applicable. Updated pointers to Figure numbers.
5-R3	9	Revised Section 3.4; updated Figure number cross references coincident with change 8-R3.
6-R3	13-15, 19-24	Revised Section 3, Figures 3.4 -3.6; 3.10 - 3.15 based on new reload safety analysis report.
7-R3	old 25-26	Removed Figures 3.16 - 3.17 as they are no longer applicable.
8-R3	25	Revised Section 4.2.1; removed last sentence as items no longer applicable. Update pointers to Table numbers.
9-R3	27	Section 4.2.5 updated Figure number cross reference.
10-R3	29-38	Updated Tables 4.2 through 4.9 with the results from revised reload safety analysis report.
11-R3	old 40-43	Removed old Tables 4.9 - 4.12 as they are no longer applicable.
1-R2	x	Added Reference 32 supplement to BFN Reload Analysis Report
2-R2	1	Revised Section 1.1.1 to identify the new range of applicability for startup with feedwater heating out of service
3-R2	8	Revised last sentence of Section 3.2.1 to reflect the updated cycle exposure range of applicability (3000 MWd/MTU); also added Reference 32
4-R2	25-26	Revised Section 3, Figure 3.16-17 titles to reflect the updated cycle exposure range of applicability (3000 MWd/MTU). No limits were changed, only titles.
5-R2	27	Revised last sentence of Section 4.2.1 to reflect the updated cycle exposure range of applicability (3000 MWd/MTU); also added Reference 32.
6-R2	40-43	Revised Section 4, Tables 4.9 - 4.12 to reflect the updated cycle exposure range of applicability (3000 MWd/MTU). No limits were changed, only titles
1-R1	x	Added Reference 31 supplement to BFN Unit 2 Cycle 19 Reload Analysis
2-R1	1-49	Corrected page numbering. Revision 0 did not reset the page numbering at Section 1 to start at 1, after the roman numeral sections

		at the front of the document. The new Arabic numbered pages for Sections 1 through 8 begin with page 1 and go through page 49.
3-R1	1	Added Section 1.1.1 to describe Revision 1 purpose
4-R1	8	Added discussion of Reference 31 for $LHGRFAC_p$
5-R1	9	Figure number range increased to Figure 3.17
6-R1	25-26	Added Figure 3.16 & Figure 3.17, $LHGRFAC_p$ for ATRIUM-10 fuel BOC - 250 MWd/MTU
7-R1	27	Added discussion of Reference 31 for $MCPR_p$
8-R1	29	Renumbered Table 4.9 as Table 4.13
9-R1	40-43	Inserted Table 4.9 through Table 4.12, BOC – 250 MWd/MTU Startup Operation $MCPR_p$ Limits
10-R1	44	Renumbered Table 4.9 as Table 4.13
0-R0	All	New document.

Nomenclature

APLHGR	Average Planar LHGR
APRM	Average Power Range Monitor
AREVA NP	Vendor (Framatome, Siemens)
BOC	Beginning of Cycle
BSP	Backup Stability Protection
BWR	Boiling Water Reactor
CAVEX	Core Average Exposure
CD	Coast Down
CMSS	Core Monitoring System Software
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRWE	Control Rod Withdrawal Error
CSDM	Cold SDM
DIVOM	Delta CPR over Initial CPR vs. Oscillation Magnitude
EOC	End of Cycle
EOCLB	End-of-Cycle Licensing Basis
EOOS	Equipment OOS
FFTR	Final Feedwater Temperature Reduction
FFWTR	Final Feedwater Temperature Reduction
FHOOS	Feedwater Heaters OOS
ft	Foot: english unit of measure for length
GNF	Vendor (General Electric, Global Nuclear Fuels)
GWd	Giga Watt Day
HTSP	High TSP
ICA	Interim Corrective Action
ICF	Increased Core Flow (beyond rated)
IS	In-Service
kW	kilo watt: SI unit of measure for power.
LCO	License Condition of Operation
LFWH	Loss of Feedwater Heating
LHGRFAC	LHGR Multiplier (Power or Flow dependent)
LPRM	Low Power Range Monitor
LRNB	Generator Load Reject, No Bypass
MAPFAC	MAPLHGR multiplier (Power or Flow dependent)

MCPR	Minimum CPR
MSRV	Moisture Separator Reheater Valve
MSRVOOS	MSRV OOS
MTU	Metric Ton Uranium
MWd/MTU	Mega Watt Day per Metric Ton Uranium
NEOC	Near EOC
NRC	United States Nuclear Regulatory Commission
NSS	Nominal Scram Speed
NTSP	Nominal TSP
OLMCPR	MCPR Operating Limit
OOS	Out-Of-Service
OPRM	Oscillation Power Range Monitor
OSS	Optimum Scram Speed
PBDA	Period Based Detection Algorithm
Pbypass	Power, below which TSV Position and TCV Fast Closure Scrams are Bypassed
PLU	Power Load Unbalance
PLUOOS	PLU OOS
PRNM	Power Range Neutron Monitor
RBM	Rod Block Monitor
RCPOOS	Recirculation Pump OOS (*SLO*)
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPTOOS	RPT OOS
SDM	Shutdown Margin
SLMCPR	MCPR Safety Limit
SLO	Single Loop Operation
TBV	Turbine Bypass Valve
TBVIS	TBV IS
TBVOOS	Turbine Bypass Valves OOS
TIP	Transversing In-core Probe
TIPOOS	TIP OOS
TLO	Two Loop Operation
TSP	Trip Setpoint
TSSS	Technical Specification Scram Speed
TVA	Tennessee Valley Authority

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TVA NPG

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Date: July 13, 2015

1 Introduction

In anticipation of cycle startup, it is necessary to describe the expected limits of operation.

1.1 Purpose

The primary purpose of this document is to satisfy requirements identified by unit technical specification section 5.6.5. This document may be provided, upon final approval, to the NRC.

1.1.1 Revision 3

The purpose of this revision is a consequence of the Reference 1 reload safety analyses revision. The Reference 1 revision is in response to condition report (CR) 1014412.

1.2 Scope

This document will discuss the following areas:

- Average Planar Linear Heat Generation Rate (APLHGR) Limit
  (Technical Specifications 3.2.1 and 3.7.5)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Linear Heat Generation Rate (LHGR) Limit
  (Technical Specification 3.2.3, 3.3.4.1, and 3.7.5)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Minimum Critical Power Ratio Operating Limit (OLMCPR)
  (Technical Specifications 3.2.2, 3.3.4.1, 3.7.5 and Table 3.3.2.1-1)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Oscillation Power Range Monitor (OPRM) Setpoint
  (Technical Specification Table 3.3.1.1)
  Applicability: Mode 1, ≥ (as specified in Technical Specifications Table 3.3.1.1-1)
- Average Power Range Monitor (APRM) Flow Biased Rod Block Trip Setting
  (Technical Requirements Manual Section 5.3.1 and Table 3.3.4-1)
  Applicability: Mode 1, ≥ (as specified in Technical Requirements Manuals Table 3.3.4-1)
- Rod Block Monitor (RBM) Trip Setpoints and Operability
  (Technical Specification Table 3.3.2.1-1)
  Applicability: Mode 1, ≥ % RTP as specified in Table 3.3.2.1-1 (TS definition of RTP)
- Shutdown Margin (SDM) Limit
  (Technical Specification 3.1.1)
  Applicability: All Modes

1.3 Fuel Loading

The core will contain previously exposed AREVA NP, Inc., ATRIUM-10 fuel, along with fresh ATRIUM-10XM and ATRIUM-11 fuel. Nuclear fuel types used in the core loading are shown in Table 1.1. The core shuffle and final loading were explicitly evaluated for BOC cold shutdown margin performance as documented per Reference 5.

Table 1.1 Nuclear Fuel Types*

Fuel Description	Original Cycle	Number of Assemblies	Nuclear Fuel Type (NFT)	Fuel Names (Range)
ATRIUM-10 A10-3799B-14GV80-FBD	17	73	10	FBD001-FBD136
ATRIUM-10 A10-4004B-15GV80-FBD	17	115	11	FBD137-FBD272
ATRIUM-10 A10-4165B-15GV75-FBE	18	176	12	FBE001-FBE176
ATRIUM-10 A10-4107B-13GV75-FBE	18	68	13	FBE177-FBE244
ATRIUM-10 A10-4176B-10GV75-FBE	18	72	14	FBE245-FBE316
ATRIUM 10XM XMLC-3904B-15GV80-FBF	19	172	15	FBF401-FBF572
ATRIUM 10XM XMLC-4035B-13GV80-FBF	19	80	16	FBF573-FBF652
ATRIUM 11 A11-3693B-13GV80-FBF	19	8	17	FBF653-FBF660

1.4 Acceptability

Limits discussed in this document were generated based on NRC approved methodologies per References 6 through 25.

* The table identifies the expected fuel type breakdown in anticipation of final core loading. The final composition of the core depends upon uncertainties during the outage such as discovering a failed fuel bundle, or other bundle damage. Minor core loading changes, due to unforeseen events, will conform to the safety and monitoring requirements identified in this document.

2 APLHGR Limits

(Technical Specifications 3.2.1 & 3.7.5)

The APLHGR limit is determined by adjusting the rated power APLHGR limit for off-rated power, off-rated flow, and SLO conditions. The most limiting of these is then used as follows:

$$\text{APLHGR limit} = \text{MIN}\ (\ \text{APLHGR}_P\ ,\ \text{APLHGR}_F,\ \text{APLHGR}_{SLO}\ )$$

where:

$APLHGR_P$	off-rated power APLHGR limit	[$APLHGR_{RATED}$ * $MAPFAC_P$]
$APLHGR_F$	off-rated flow APLHGR limit	[$APLHGR_{RATED}$ * $MAPFAC_F$]
$APLHGR_{SLO}$	SLO APLHGR limit	[$APLHGR_{RATED}$ * SLO Multiplier]

2.1 Rated Power and Flow Limit: $APLHGR_{RATED}$

The rated conditions APLHGR for ATRIUM-10 fuel is identified in Reference 1 and shown in Figure 2.1. The rated conditions APLHGR for ATRIUM-10XM are shown in Figure 2.2. The rated conditions APLHGR for ATRIUM-11 are shown in Figure 2.3.

2.2 Off-Rated Power Dependent Limit: $APLHGR_P$

Reference 1 does not specify a power dependent APLHGR. Therefore, $MAPFAC_P$ is set to a value of **1.0**.

2.2.1 Startup without Feedwater Heaters

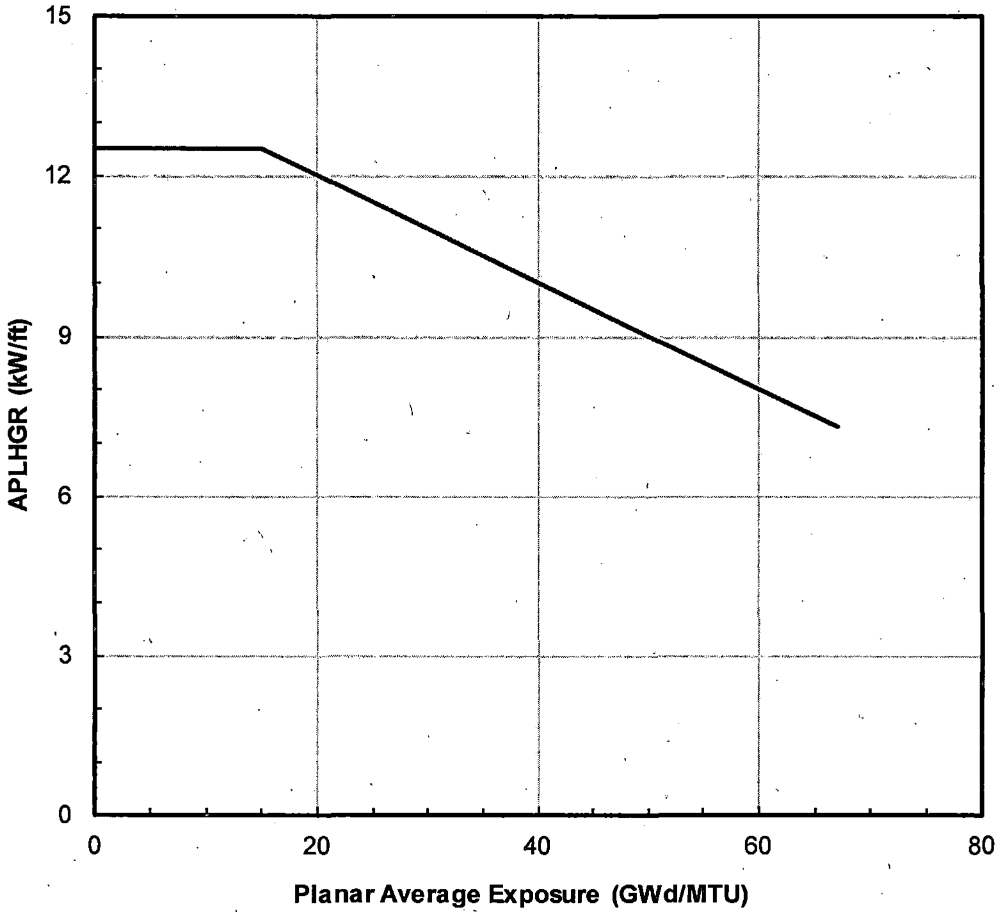
There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. No additional power dependent limitation is required.

2.3 Off-Rated Flow Dependent Limit: $APLHGR_F$

Reference 1 does not specify a flow dependent APLHGR. Therefore, $MAPFAC_F$ is set to a value of **1.0**.

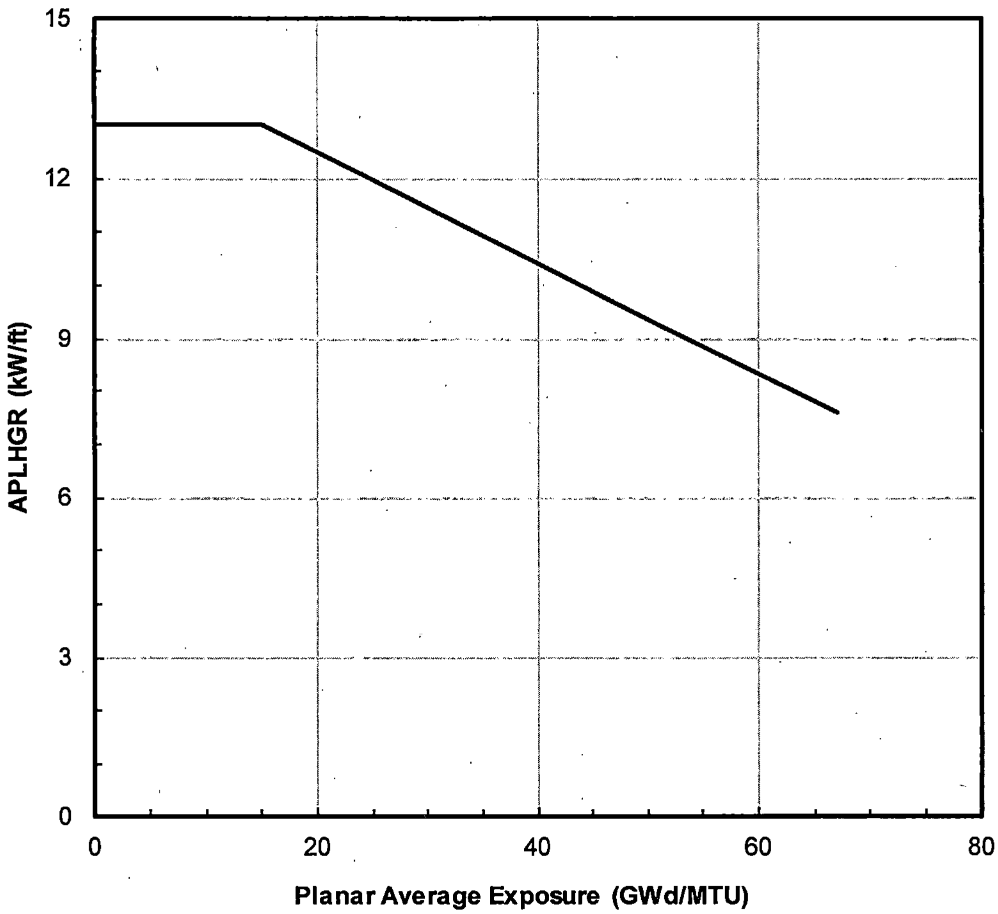
2.4 Single Loop Operation Limit: $APLHGR_{SLO}$

The single loop operation multiplier for ATRIUM-10, ATRIUM-10XM, and ATRIUM-11 fuel is **0.85**, per Reference 1.



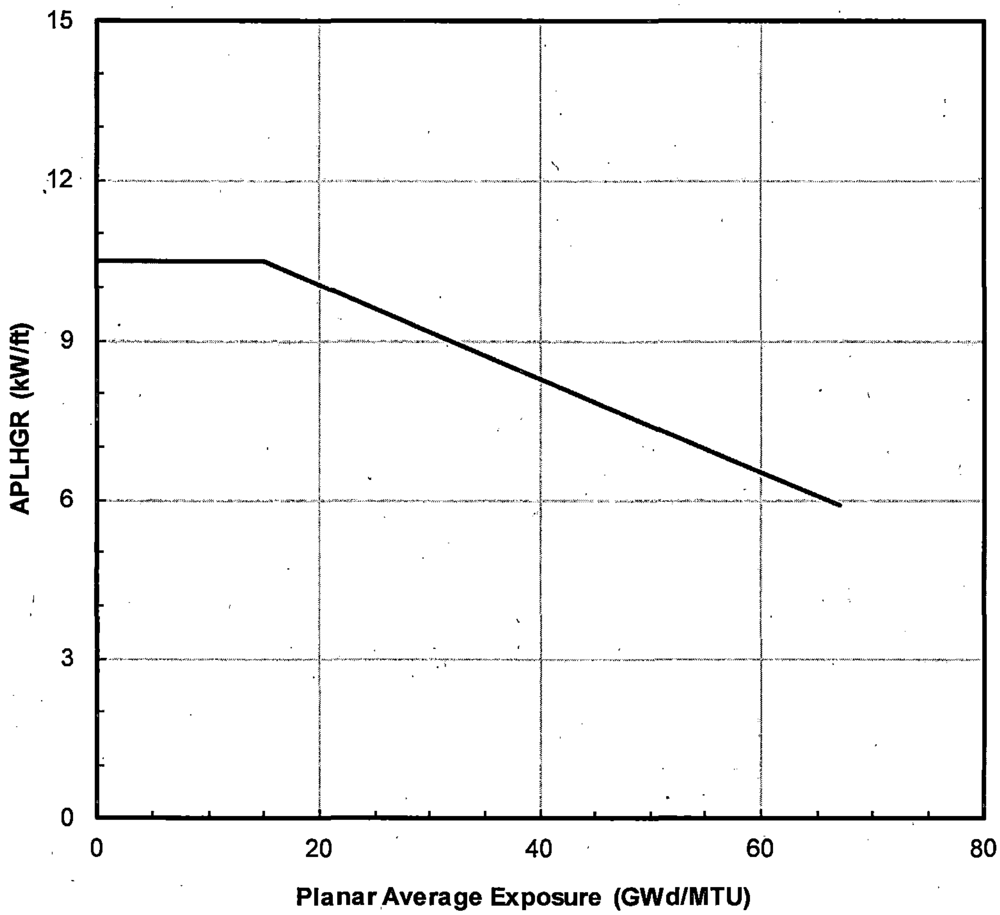
Planar Avg. Exposure (GWd/MTU)	APLHGR Limit (kW/ft)
0.0	12.5
15.0	12.5
67.0	7.3

Figure 2.1 $APLHGR_{RATED}$ for ATRIUM-10 Fuel



Planar Avg. Exposure (GWd/MTU)	APLHGR Limit (kW/ft)
0.0	13.0
15.0	13.0
67.0	7.6

Figure 2.2 APLHGR$_{RATED}$ for ATRIUM-10XM Fuel



Planar Avg. Exposure (GWd/MTU)	APLHGR Limit (kW/ft)
0.0	10.5
15.0	10.5
67.0	5.9

Figure 2.3 APLHGR$_{RATED}$ for ATRIUM-11 Fuel

2.5 Equipment Out-Of-Service Corrections

The limits shown in Figure 2.1, Figure 2.2, and Figure 2.3 are applicable for operation with all equipment In-Service as well as the following Equipment Out-Of-Service (EOOS) options; including combinations of the options.

In-Service	All equipment In-Service*
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
PLUOOS	Power Load Unbalance Out-Of-Service
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service or Final Feedwater Temperature Reduction
RCPOOS	One Recirculation Pump Out-Of-Service

* All equipment service conditions assume 1 SRVOOS.

3 LHGR Limits

(Technical Specification 3.2.3, 3.3.4.1, & 3.7.5)

The LHGR limit is determined by adjusting the rated power LHGR limit for off-rated power and off-rated flow conditions. The most limiting of these is then used as follows:

LHGR limit = MIN ($LHGR_P$, $LHGR_F$)

where:

$LHGR_P$	off-rated power LHGR limit	[$LHGR_{RATED}$ * $LHGRFAC_P$]
$LHGR_F$	off-rated flow LHGR limit	[$LHGR_{RATED}$ * $LHGRFAC_F$]

3.1 Rated Power and Flow Limit: $LHGR_{RATED}$

The rated conditions LHGR for ATRIUM-10 fuel is identified in Reference 1 and shown in Figure 3.1. The rated conditions LHGR for ATRIUM-10XM fuel is shown in Figure 3.2. The rated conditions LHGR for ATRIUM-11 fuel is shown in Figure 3.3. The LHGR limit is consistent with References 2 and 3.

3.2 Off-Rated Power Dependent Limit: $LHGR_P$

LHGR limits are adjusted for off-rated power conditions using the $LHGRFAC_P$ multiplier provided in Reference 1. The multiplier is split into two sub cases: turbine bypass valves in and out-of-service. The base case multipliers are shown in Figure 3.4 through Figure 3.6.

3.2.1 Startup without Feedwater Heaters

There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. Additional limits are shown in Figure 3.10 through Figure 3.15, based on temperature conditions identified in Table 3.1.

Table 3.1 Startup Feedwater Temperature Basis

Power	Temperature Range 1	Temperature Range 2
(% Rated)	(°F)	(°F)
25	160.0	155.0
30	165.0	160.0
40	175.0	170.0
50	185.0	180.0

3.3 Off-Rated Flow Dependent Limit: $LHGR_F$

LHGR limits are adjusted for off-rated flow conditions using the $LHGRFAC_F$ multiplier provided in Reference 1. Multipliers are shown in Figure 3.7 through Figure 3.9.

3.4 Equipment Out-Of-Service Corrections

The limits shown in Figure 3.1 through Figure 3.3 are applicable for operation with all equipment In-Service as well as the following Equipment Out-Of-Service (EOOS) options; including combinations of the options.*

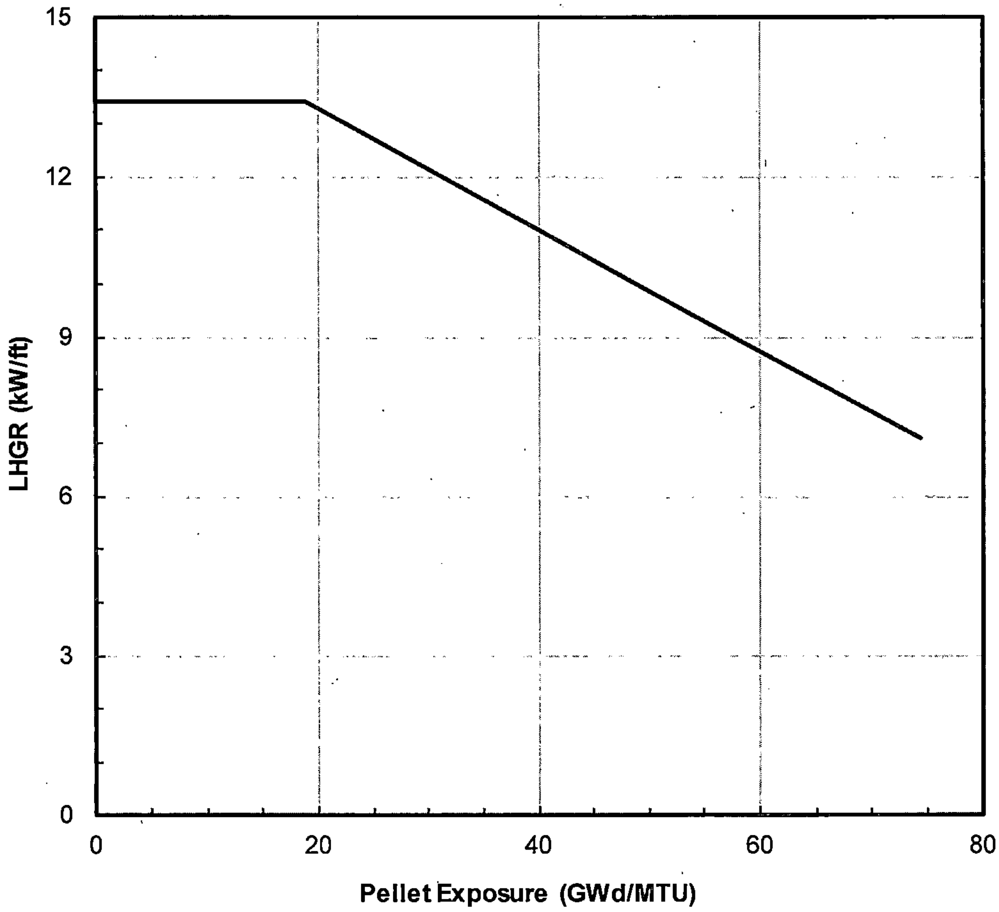
In-Service	All equipment In-Service
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
PLUOOS	Power Load Unbalance Out-Of-Service
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service or Final Feedwater Temperature Reduction
RCPOOS	One Recirculation Pump Out-Of-Service

Off-rated power corrections shown in Figure 3.4 through Figure 3.6 are dependent on operation of the Turbine Bypass Valve system. For this reason, separate limits are to be applied for TBVIS or TBVOOS operation. The limits have no dependency on RPTOOS, PLUOOS, FHOOS/FFWTR, or SLO.

Off-rated flow corrections shown in Figure 3.7 through Figure 3.9 are bounding for all EOOS conditions.

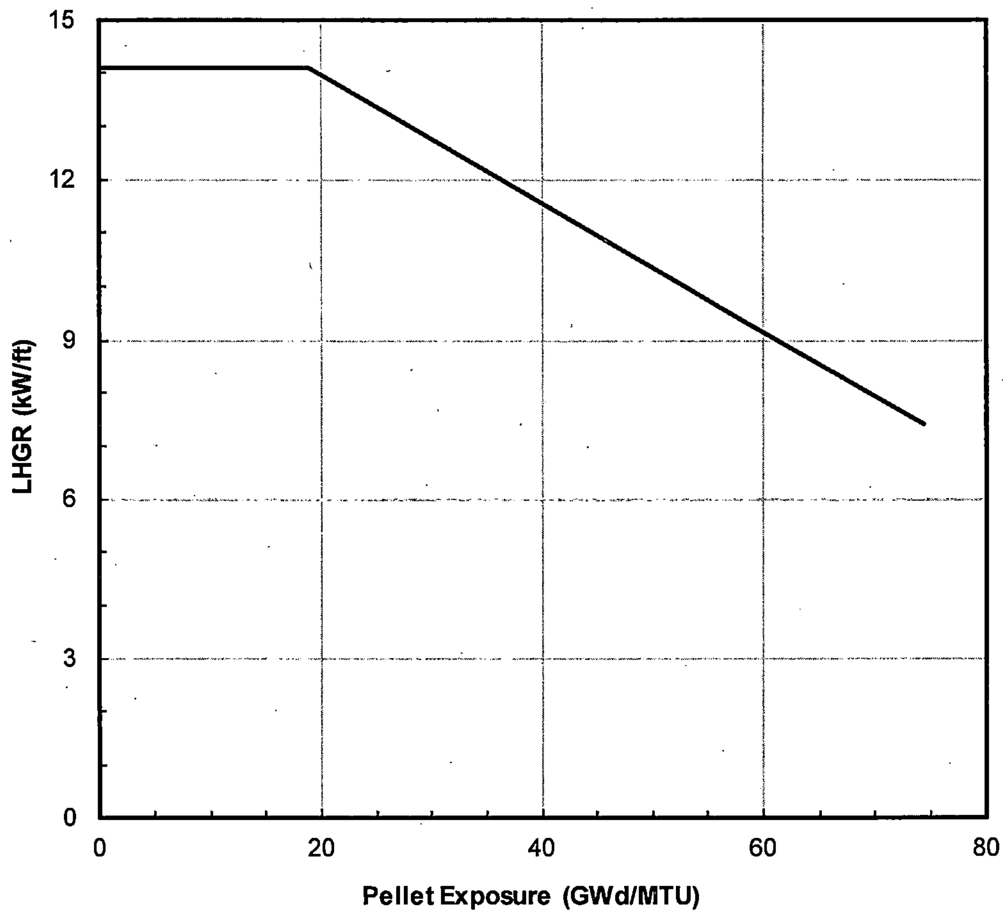
Off-rated power corrections shown in Figure 3.10 through Figure 3.15 are also dependent on operation of the Turbine Bypass Valve system. In this case, limits support FHOOS operation during startup. These limits have no dependency on RPTOOS, PLUOOS, or SLO.

* All equipment service conditions assume 1 SRVOOS.



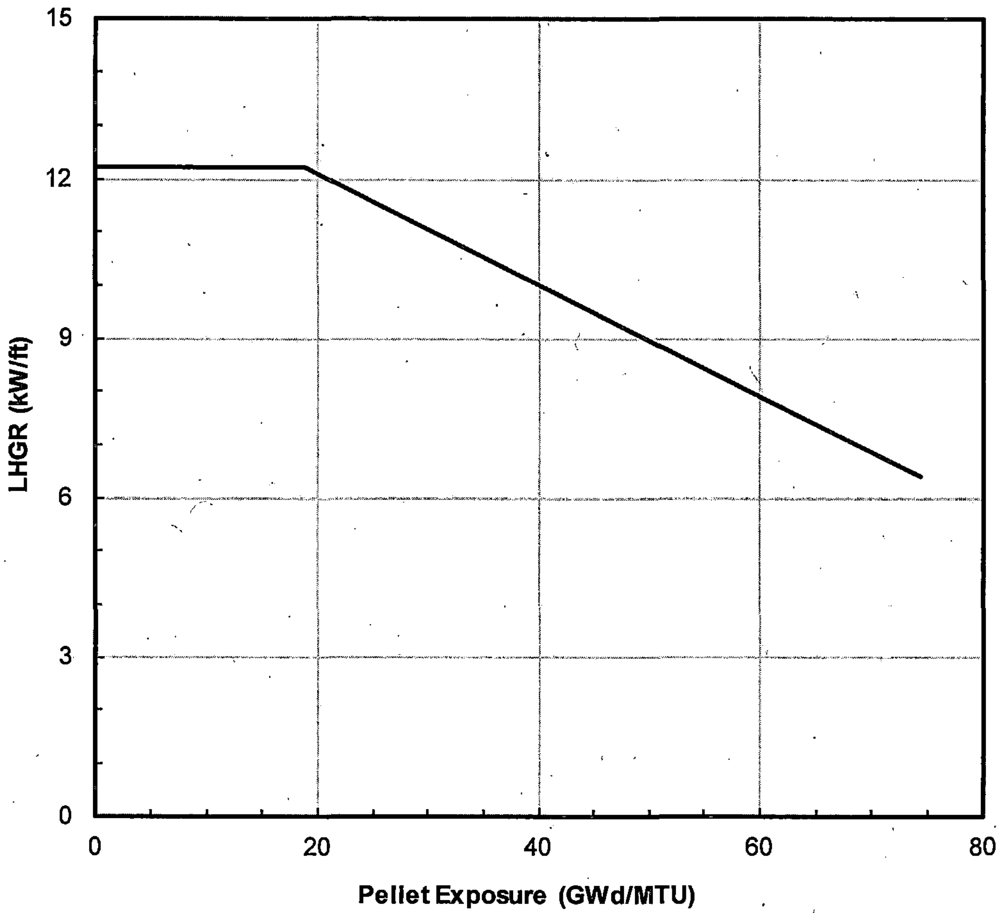
Pellet Exposure (GWd/MTU)	LHGR Limit (kW/ft)
0.0	13.4
18.9	13.4
74.4	7.1

Figure 3.1 $LHGR_{RATED}$ for ATRIUM-10 Fuel



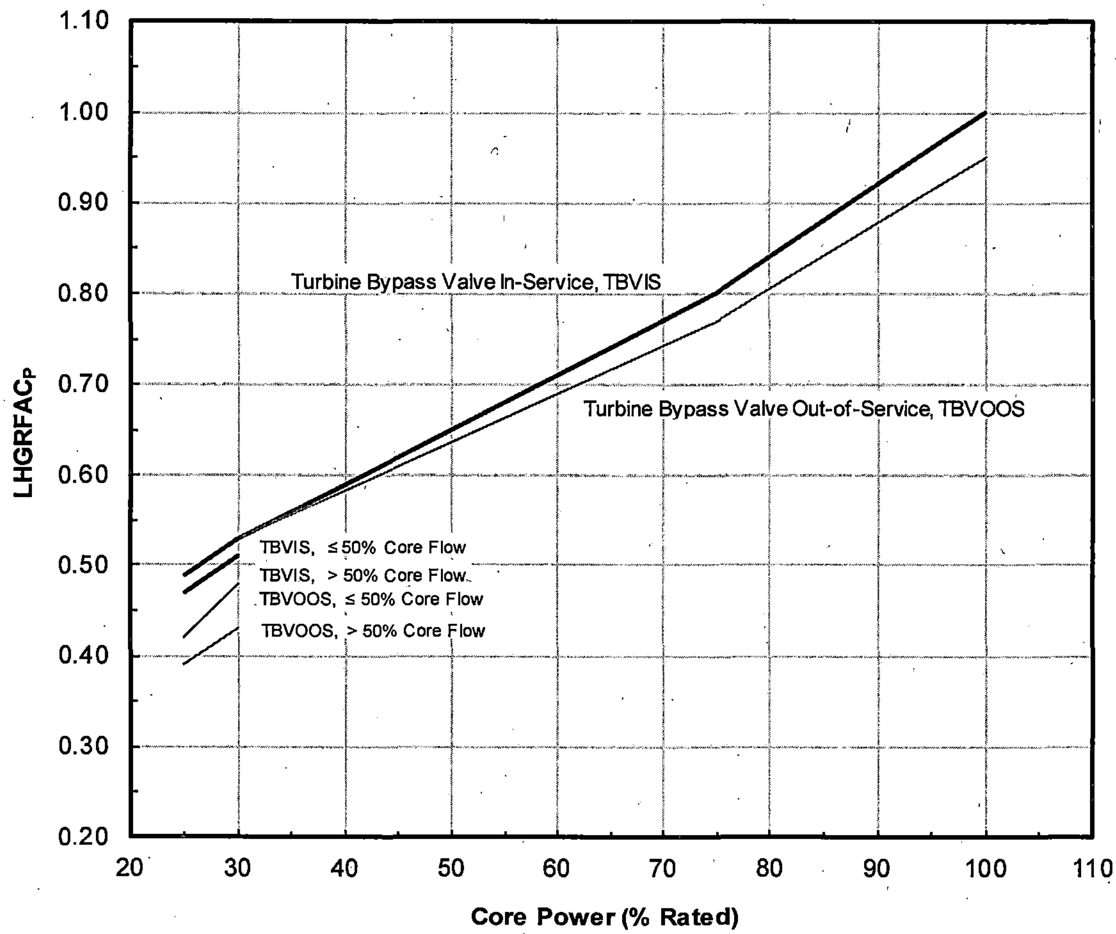
Pellet Exposure (GWd/MTU)	LHGR Limit (kW/ft)
0.0	14.1
18.9	14.1
74.4	7.4

Figure 3.2 $LHGR_{RATED}$ for ATRIUM-10XM Fuel



Pellet Exposure (GWd/MTU)	LHGR Limit (kW/ft)
0.0	12.2
18.9	12.2
74.4	6.4

Figure 3.3 LHGR$_{RATED}$ for ATRIUM-11 Fuel

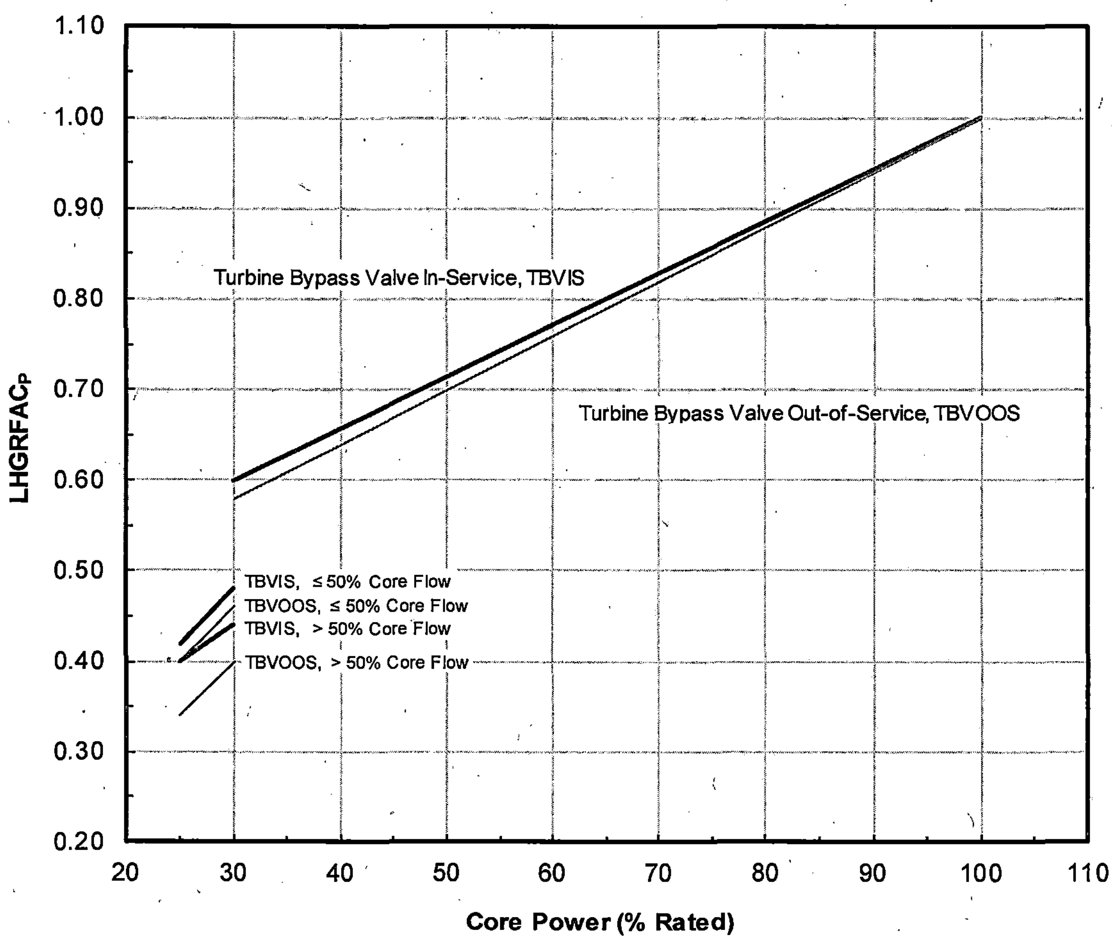


Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00
75.0	0.80
30.0	0.53
Core Flow > 50% Rated	
30.0	0.51
25.0	0.47
Core Flow ≤ 50% Rated	
30.0	0.53
25.0	0.49

Turbine Bypass Out-of-Service Core Power (% Rated)	LHGRFAC$_P$
100.0	0.95
75.0	0.77
30.0	0.53
Core Flow > 50% Rated	
30.0	0.43
25.0	0.39
Core Flow ≤ 50% Rated	
30.0	0.48
25.0	0.42

Figure 3.4 Base Operation LHGRFAC$_P$ for ATRIUM-10 Fuel

(*Independent of other EOOS conditions*)

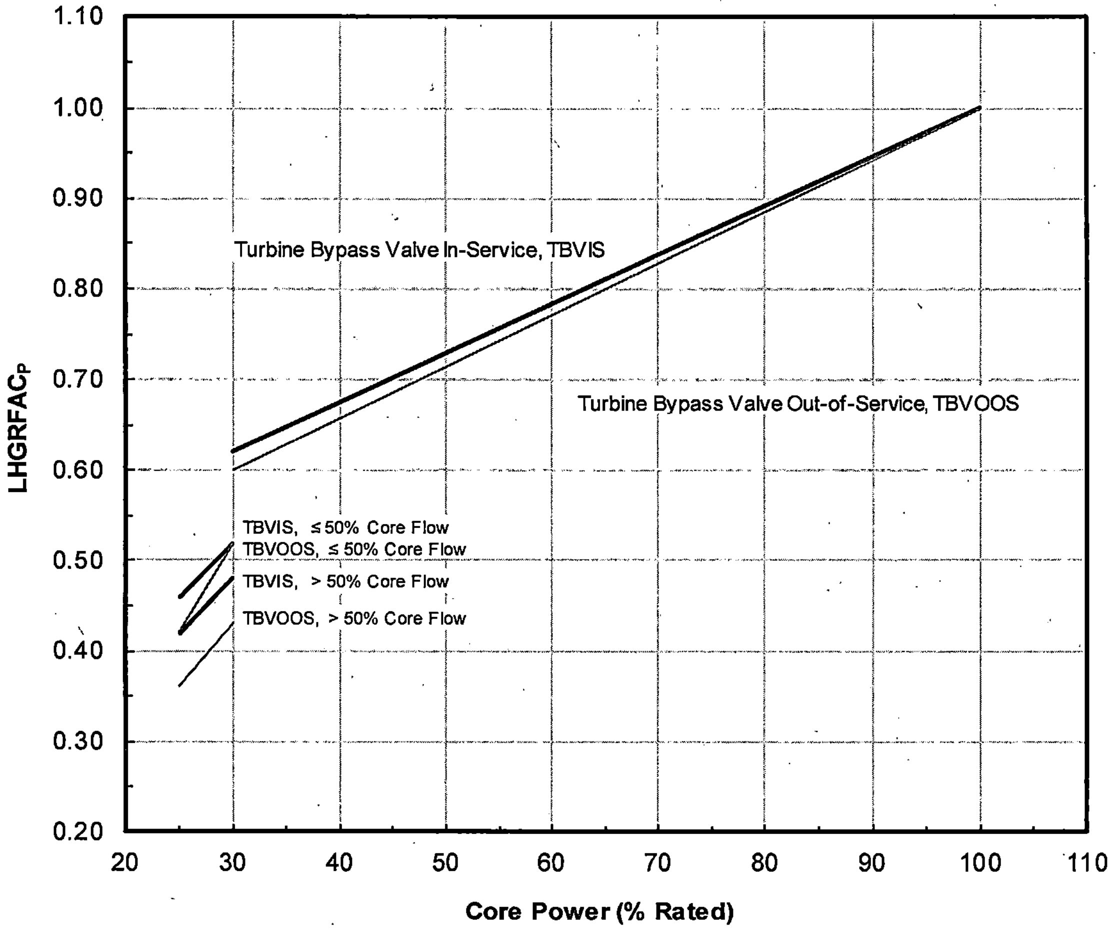


Turbine Bypass In-Service	
Core Power (% Rated)	**LHGRFAC$_P$**
100.0	1.00
30.0	0.60
Core Flow > 50% Rated	
30.0	0.44
25.0	0.40
Core Flow ≤ 50% Rated	
30.0	0.48
25.0	0.42

Turbine Bypass Out-of-Service	
Core Power (% Rated)	**LHGRFAC$_P$**
100.0	1.00
30.0	0.58
Core Flow > 50% Rated	
30.0	0.40
25.0	0.34
Core Flow ≤ 50% Rated	
30.0	0.46
25.0	0.40

Figure 3.5 Base Operation LHGRFAC$_P$ for ATRIUM-10XM Fuel

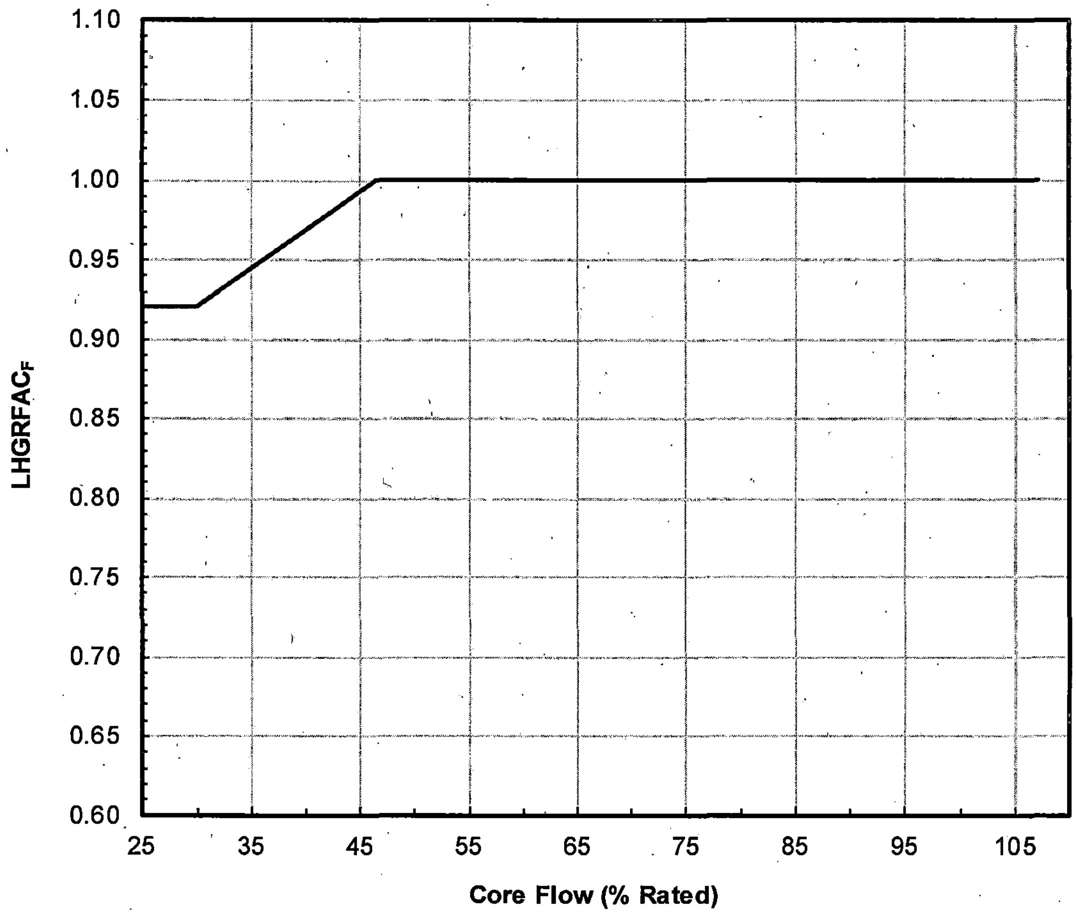
(*Independent of other EOOS conditions*)



Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$	*Turbine Bypass Out-of-Service* Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00	100.0	1.00
30.0	0.62	30.0	0.60
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.48	30.0	0.43
25.0	0.42	25.0	0.36
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.52	30.0	0.52
25.0	0.46	25.0	0.42

Figure 3.6 Base Operation LHGRFAC$_P$ for ATRIUM-11 Fuel

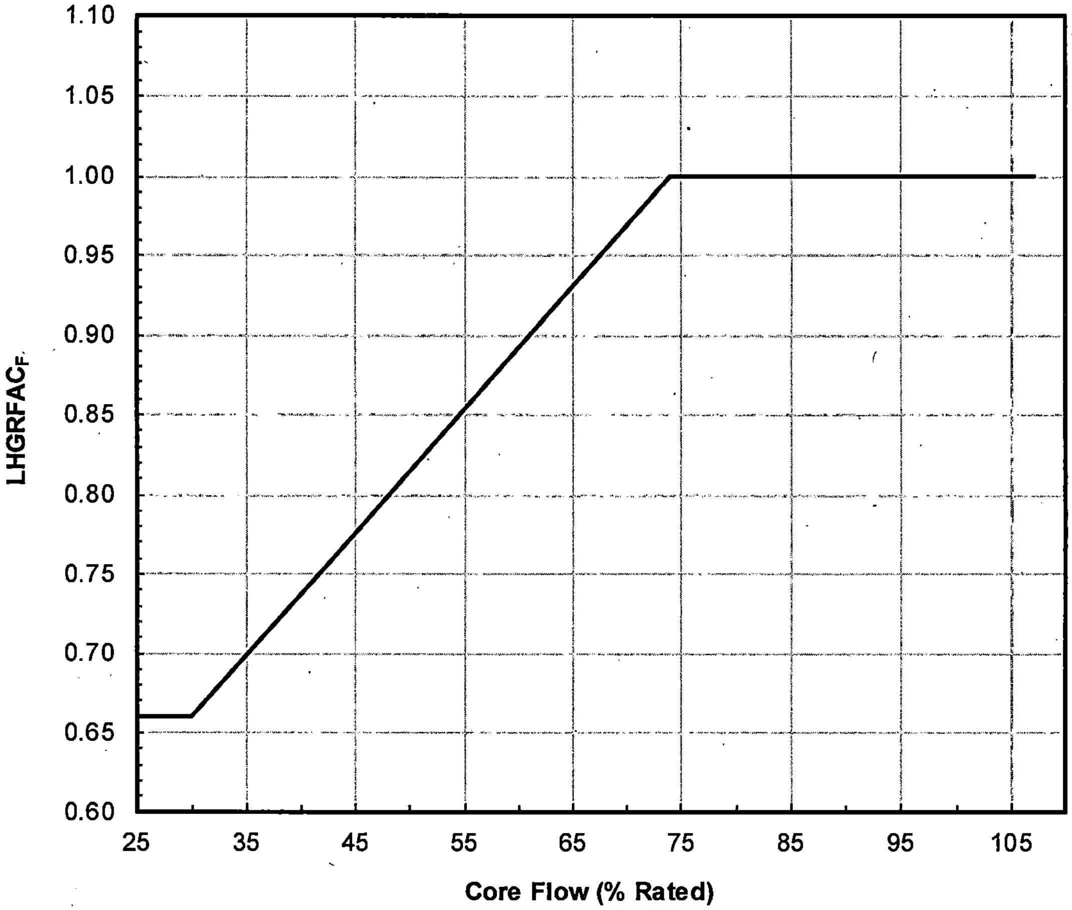
(*Independent of other EOOS conditions*)



Core Flow (% Rated)	LHGRFAC$_F$
0.0	0.92
30.0	0.92
46.4	1.00
107.0	1.00

Figure 3.7 LHGRFAC$_F$ for ATRIUM-10 Fuel
(*Values bound all EOOS conditions*)

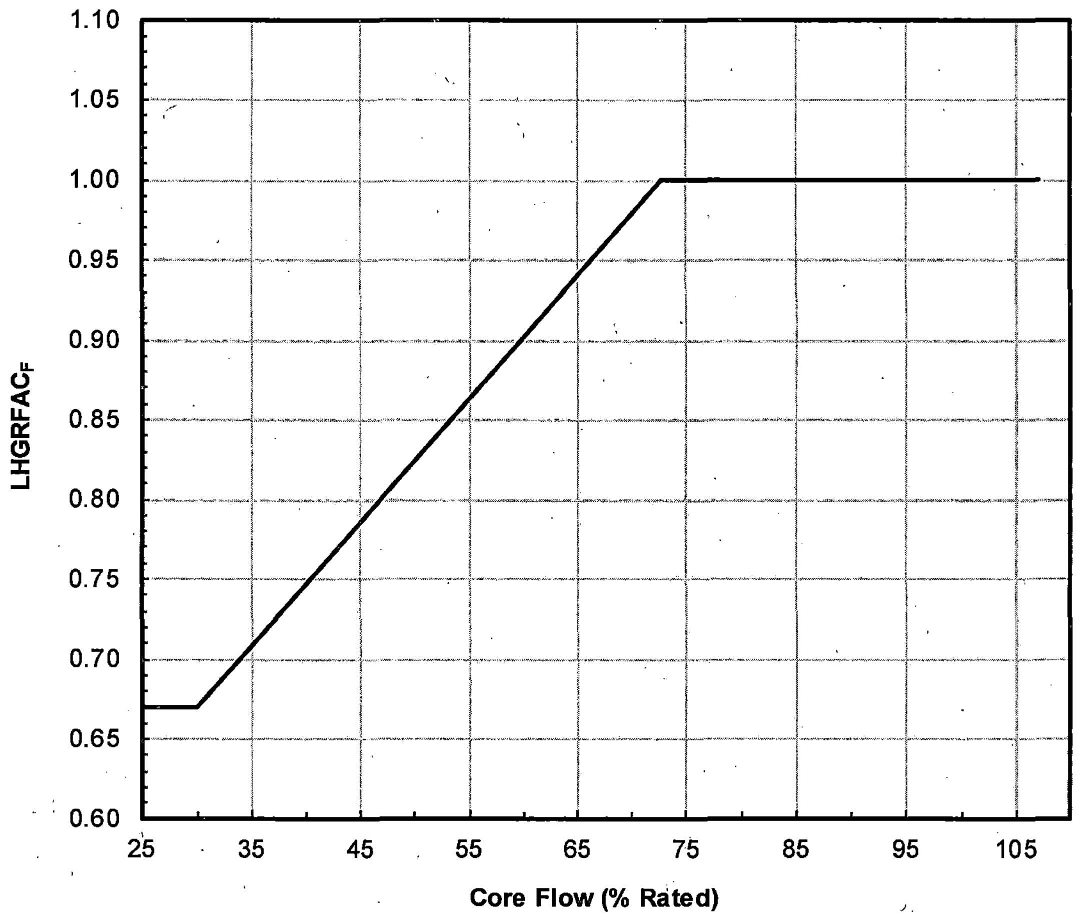
(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)



Core Flow (% Rated)	LHGRFAC$_F$
0.0	0.66
30.0	0.66
73.9	1
107.0	1

Figure 3.8 LHGRFAC$_F$ for ATRIUM-10XM Fuel
(*Values bound all EOOS conditions*)

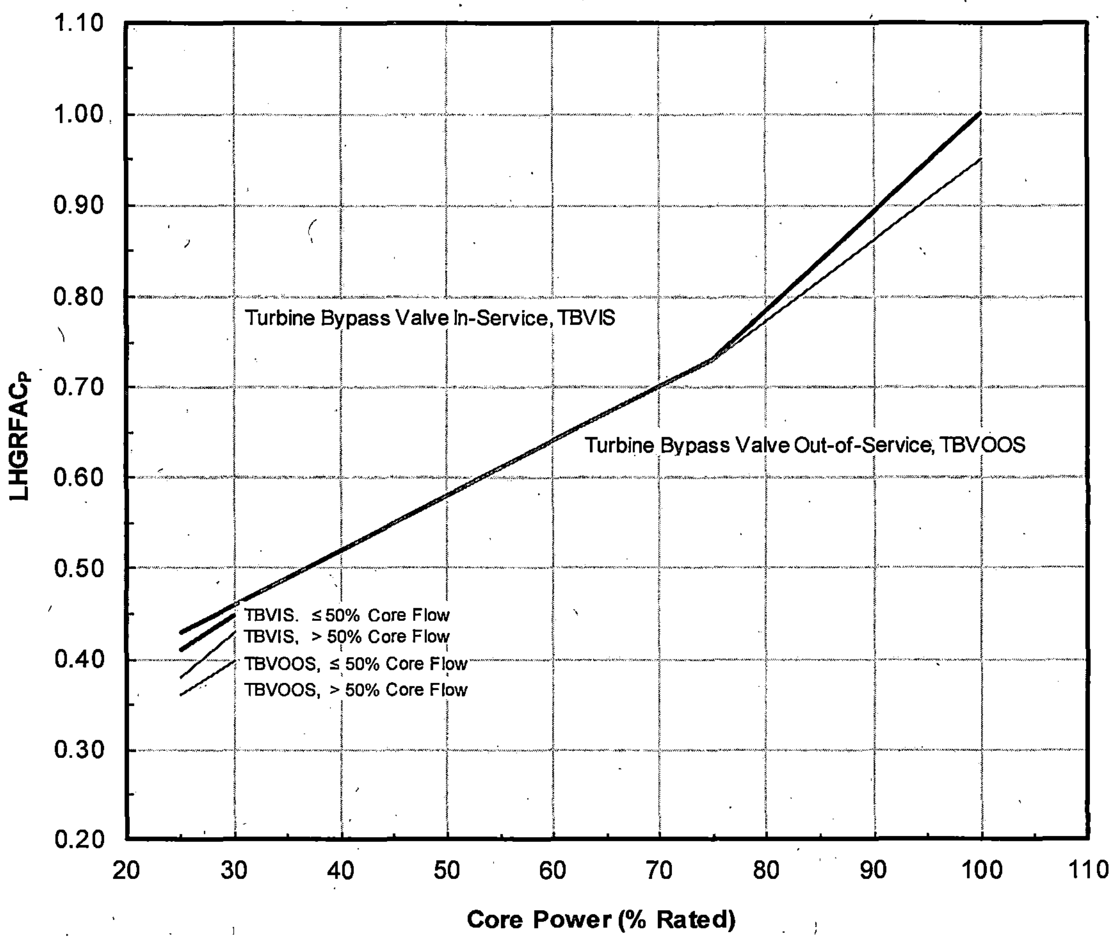
(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)



Core Flow (% Rated)	LHGRFAC$_F$
0.0	0.67
30.0	0.67
72.7	1
107.0	1

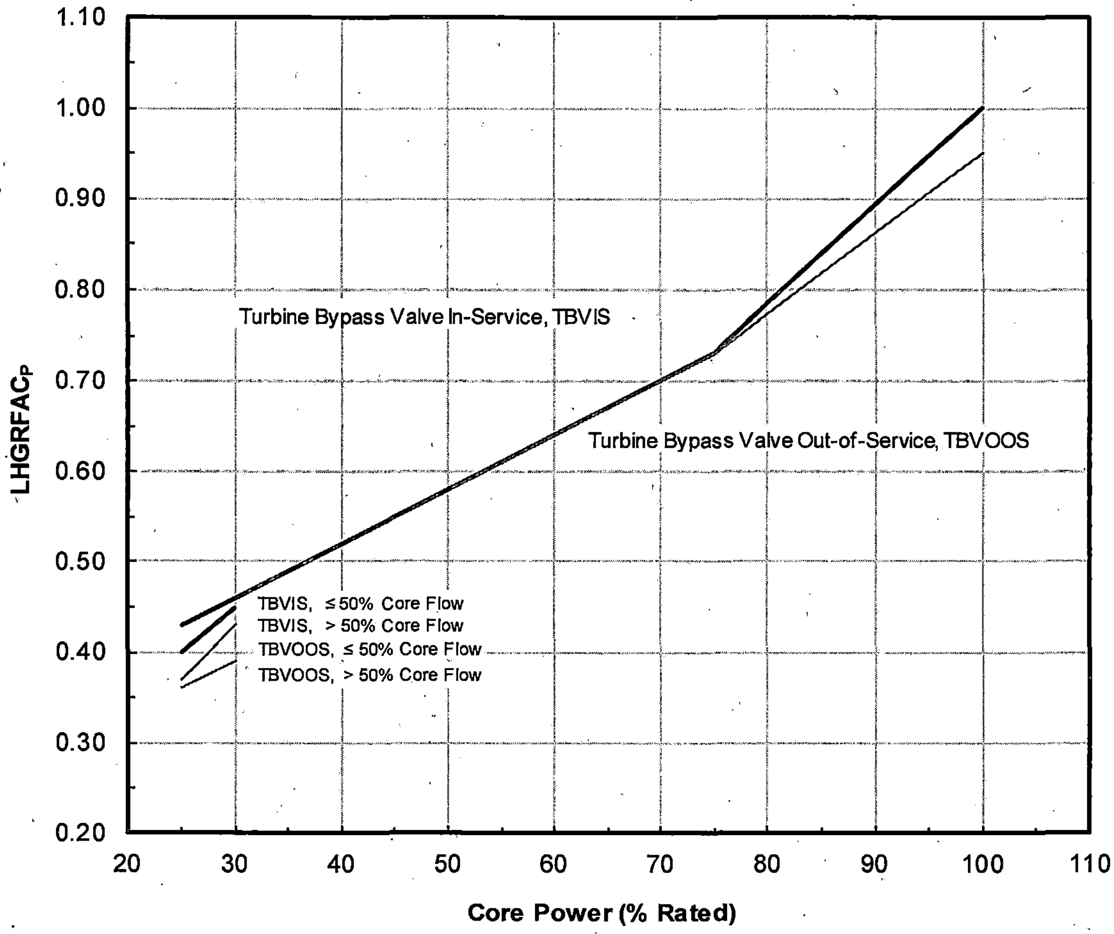
Figure 3.9 LHGRFAC$_F$ for ATRIUM-11 Fuel
(*Values bound all EOOS conditions*)

(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)



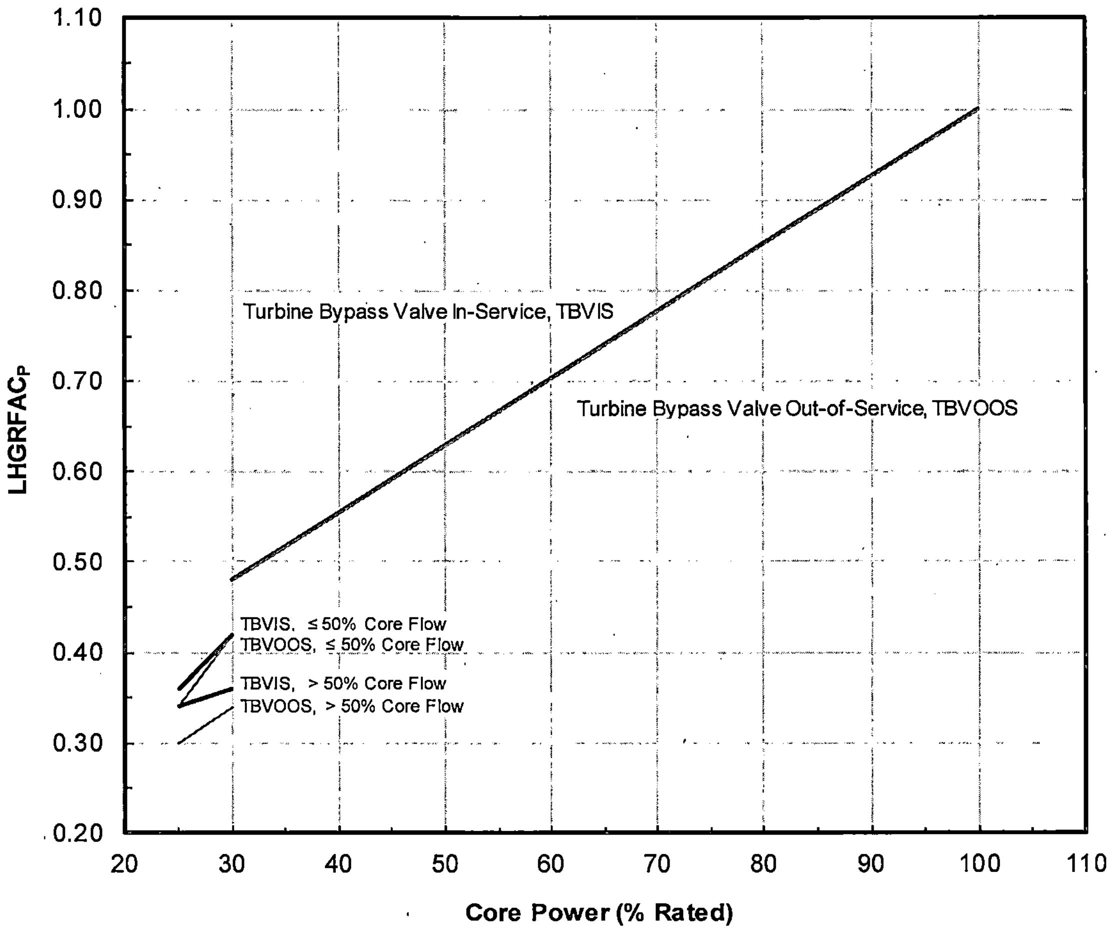
Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$	*Turbine Bypass Out-of-Service* Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00	100.0	0.95
75.0	0.73	75.0	0.73
30.0	0.46	30.0	0.46
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.45	30.0	0.40
25.0	0.41	25.0	0.36
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.46	30.0	0.43
25.0	0.43	25.0	0.38

Figure 3.10 Startup Operation LHGRFAC$_P$ for ATRIUM-10 Fuel:
Table 3.1 Temperature Range 1
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



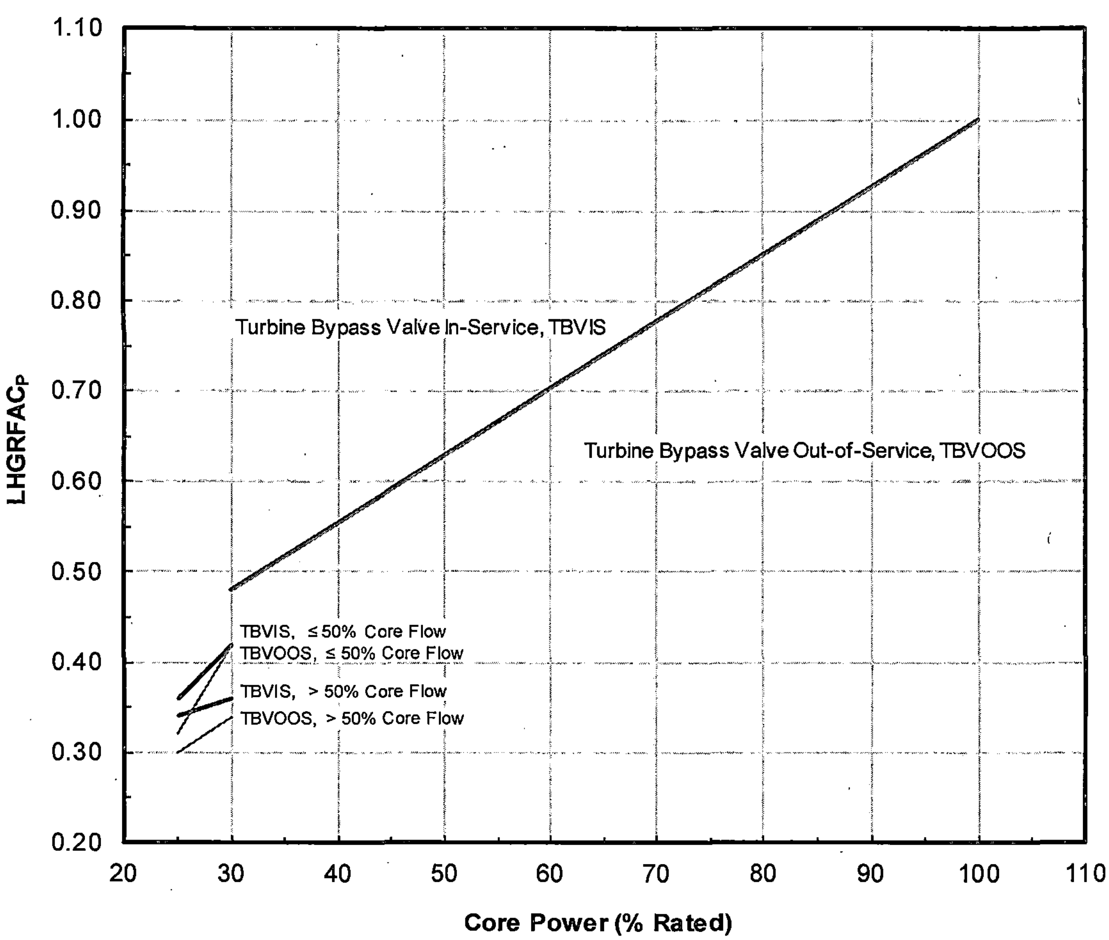
Turbine Bypass In-Service Core Power (% Rated)	$LHGRFAC_P$	*Turbine Bypass Out-of-Service* Core Power (% Rated)	$LHGRFAC_P$
100.0	1.00	100.0	0.95
75.0	0.73	75.0	0.73
30.0	0.46	30.0	0.46
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.45	30.0	0.39
25.0	0.40	25.0	0.36
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.46	30.0	0.43
25.0	0.43	25.0	0.37

Figure 3.11 Startup Operation $LHGRFAC_P$ for ATRIUM-10 Fuel:
Table 3.1 Temperature Range 2
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



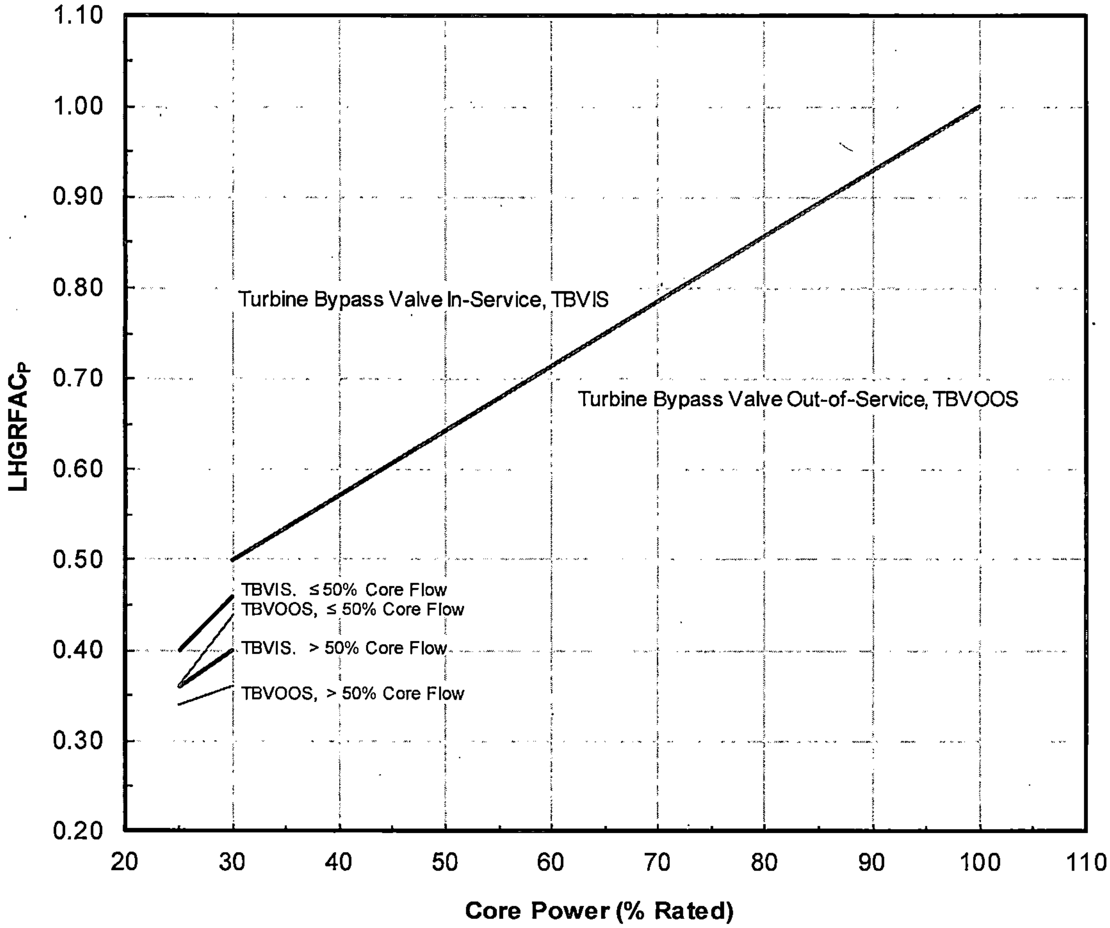
Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$	*Turbine Bypass Out-of-Service* Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00	100.0	1.00
30.0	0.48	30.0	0.48
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.36	30.0	0.34
25.0	0.34	25.0	0.30
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.42	30.0	0.42
25.0	0.36	25.0	0.34

Figure 3.12 Startup Operation LHGRFAC$_P$ for ATRIUM-10XM
Fuel: Table 3.1 Temperature Range 1
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



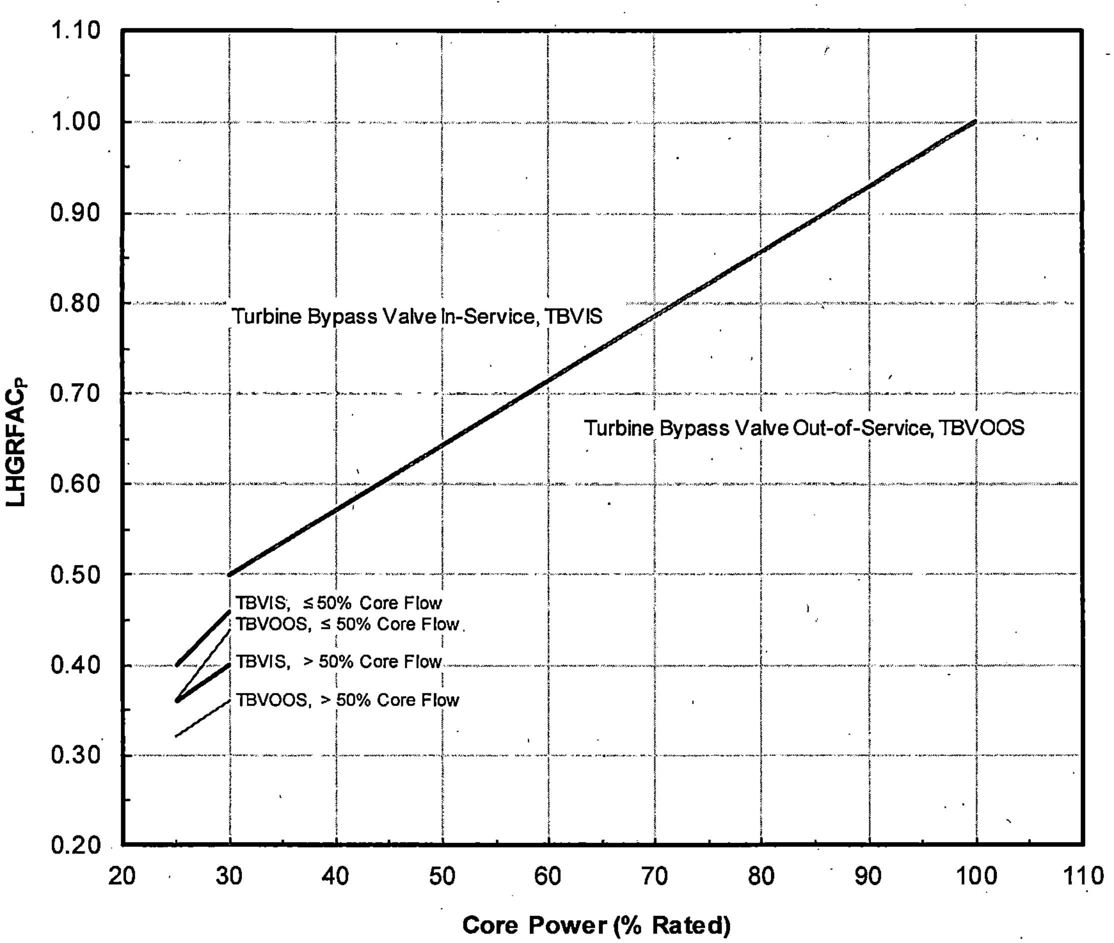
Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$	*Turbine Bypass Out-of-Service* Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00	100.0	1.00
30.0	0.48	30.0	0.48
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.36	30.0	0.34
25.0	0.34	25.0	0.30
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.42	30.0	0.42
25.0	0.36	25.0	0.32

Figure 3.13 Startup Operation LHGRFAC$_P$ for ATRIUM-10XM
Fuel: Table 3.1 Temperature Range 2
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



Turbine Bypass In-Service Core Power (%Rated)	LHGRFAC$_P$	*Turbine Bypass Out-of-Service* Core Power (%Rated)	LHGRFAC$_P$
100.0	1.00	100.0	1.00
30.0	0.50	30.0	0.50
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.40	30.0	0.36
25.0	0.36	25.0	0.34
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.46	30.0	0.44
25.0	0.40	25.0	0.36

Figure 3.14 Startup Operation LHGRFAC$_P$ for ATRIUM-11 Fuel:
Table 3.1 Temperature Range 1
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00
30.0	0.50
Core Flow > 50% Rated	
30.0	0.40
25.0	0.36
Core Flow ≤ 50% Rated	
30.0	0.46
25.0	0.40

Turbine Bypass Out-of-Service Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00
30.0	0.50
Core Flow > 50% Rated	
30.0	0.36
25.0	0.32
Core Flow ≤ 50% Rated	
30.0	0.44
25.0	0.36

Figure 3.15 Startup Operation LHGRFAC$_P$ for ATRIUM-11 Fuel:
Table 3.1 Temperature Range 2
(no Feedwater heating during startup)
(Limits valid at and below 50% power)

4 OLMCPR Limits

(Technical Specification 3.2.2, 3.3.4.1, & 3.7.5)

OLMCPR is calculated to be the most limiting of the flow or power dependent values

OLMCPR limit = MAX ($MCPR_F$, $MCPR_P$)

where:

$MCPR_F$	core flow-dependent MCPR limit
$MCPR_P$	power-dependent MCPR limit

4.1 Flow Dependent MCPR Limit: $MCPR_F$

$MCPR_F$ limits are dependent upon core flow (% of Rated), and the max core flow limit, (Rated or Increased Core Flow, ICF). $MCPR_F$ limits are shown in Figure 4.1, per Reference 1. Limits are valid for all EOOS combinations. No adjustment is required for SLO conditions.

4.2 Power Dependent MCPR Limit: $MCPR_P$

$MCPR_P$ limits are dependent upon:

- Core Power Level (% of Rated)
- Technical Specification Scram Speed (TSSS), Nominal Scram Speed (NSS), or Optimum Scram Speed (OSS)
- Cycle Operating Exposure (NEOC, EOC, and CD - as defined in this section)
- Equipment Out-Of-Service Options
- Two or Single recirculation Loop Operation (TLO vs. SLO)

The $MCPR_P$ limits are provided in Table 4.2 through Table 4.4, where each table contains the limits for all fuel types and EOOS options (for a specified scram speed and exposure range). The CMSS determines $MCPR_P$ limits, from these tables, based on linear interpolation between the specified powers.

4.2.1 <u>Startup without Feedwater Heaters</u>

There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. Additional power dependent limits are shown in Table 4.5 through Table 4.8 based on temperature conditions identified in Table 3.1.

4.2.2 Scram Speed Dependent Limits (TSSS vs. NSS vs. OSS)

$MCPR_P$ limits are provided for three different sets of assumed scram speeds. The Technical Specification Scram Speed (TSSS) $MCPR_P$ limits are applicable at all times, as long as the scram time surveillance demonstrates the times in Technical Specification Table 3.1.4-1 are met. Both Nominal Scram Speeds (NSS) and/or Optimum Scram Speeds (OSS) may be used, as long as the scram time surveillance demonstrates Table 4.1 times are applicable.*†

Table 4.1 Nominal Scram Time Basis

Notch Position (index)	Nominal Scram Timing (seconds)	Optimum Scram Timing (seconds)
46	0.420	0.380
36	0.980	0.875
26	1.600	1.465
6	2.900	2.900

In demonstrating compliance with the NSS and/or OSS scram time basis, surveillance requirements from Technical Specification 3.1.4 apply; accepting the definition of SLOW rods should conform to scram speeds shown in Table 4.1. If conformance is not demonstrated, TSSS based $MCPR_P$ limits are applied.

On initial cycle startup, TSSS limits are used until the successful completion of scram timing confirms NSS and/or OSS based limits are applicable.

4.2.3 Exposure Dependent Limits

Exposures are tracked on a Core Average Exposure basis (CAVEX, not Cycle Exposure). Higher exposure $MCPR_P$ limits are always more limiting and may be used for any Core Average Exposure up to the ending exposure. Per Reference 1, $MCPR_P$ limits are provided for the following exposure ranges:

BOC to NEOC	NEOC corresponds to	**30,910.3 MWd / MTU**
BOC to EOCLB	EOCLB corresponds to	**33,210.3 MWd / MTU**
BOC to End of Coast	End of Coast	**34,624.4 MWd / MTU**

NEOC refers to a Near EOC exposure point.

* Reference 1 analysis results are based on information identified in Reference 4.

† Drop out times consistent with method used to perform actual timing measurements (i.e., including pickup/dropout effects).

The EOCLB exposure point is not the true End-Of-Cycle exposure. Instead it corresponds to a licensing exposure window exceeding expected end-of-full-power-life.

The End of Coast exposure point represents a licensing exposure point exceeding the expected end-of-cycle exposure including cycle extension options.

4.2.4 Equipment Out-Of-Service (EOOS) Options

EOOS options* covered by $MCPR_P$ limits are given by the following:

In-Service	All equipment In-Service
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
RPTOOS+TBVOOS	Combined RPTOOS and TBVOOS
PLUOOS	Power Load Unbalance Out-Of-Service
PLUOOS+RPTOOS	Combined PLUOOS and RPTOOS
PLUOOS+TBVOOS	Combined PLUOOS and TBVOOS
PLUOOS+TBVOOS+RPTOOS	Combined PLUOOS, RPTOOS, and TBVOOS
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service (or Final Feedwater Temperature Reduction)
RCPOOS	One Recirculation Pump Out-Of-Service

For exposure ranges up to NEOC and EOCLB, additional combinations of $MCPR_P$ limits are also provided including FHOOS. The coast down exposure range assumes application of FFWTR. FHOOS based $MCPR_P$ limits for the coast down exposure are redundant because the temperature setdown assumption is identical with FFWTR.

4.2.5 Single-Loop-Operation (SLO) Limits

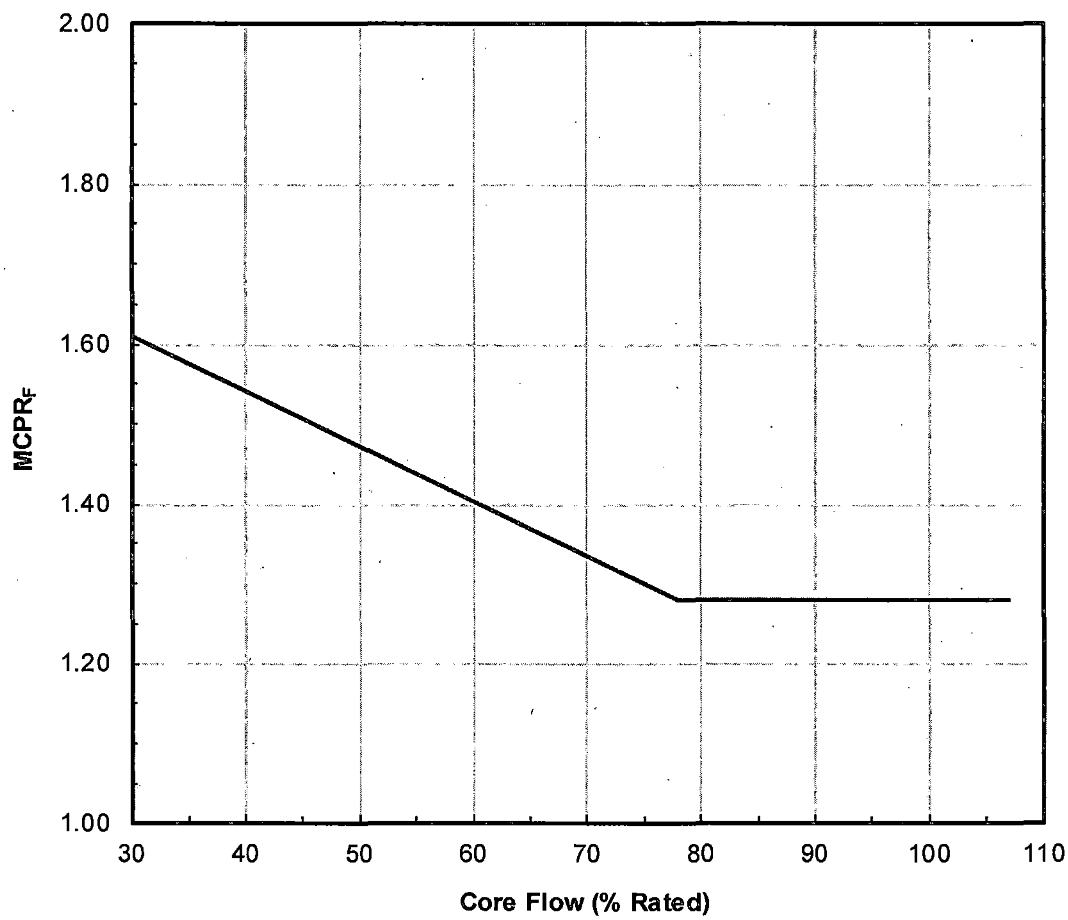
When operating in RCPOOS conditions, MCPRp limits are constructed differently from the normal operating RCP conditions. The limiting event for RCPOOS is a pump seizure scenario, which sets the upper bound for allowed core power and flow†. This event is not impacted by scram time assumptions. Specific $MCPR_P$ limits are shown in Table 4.9.

4.2.6 Below Pbypass Limits

Below Pbypass (30% rated power), $MCPR_P$ limits depend upon core flow. One set of $MCPR_P$ limits applies for core flow above 50% of rated; a second set applies if the core flow is less than or equal to 50% rated.

* All equipment service conditions assume 1 SRVOOS.

† RCPOOS limits are only valid up to 50% rated core power, 50% rated core flow, and an active recirculation drive flow of 17.73 Mlb_m/hr.



Core Flow (% Rated)	$MCPR_F$
30.0	1.61
78.0	1.28
107.0	1.28

Figure 4.1 $MCPR_F$ for All Fuel Types
(*Values bound all EOOS conditions*)

(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)

Table 4.2 MCPR$_P$ Limits for All Fuel Types: Optimum Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
Base Case	100	1.43	1.46	1.48	1.40	1.41	1.43	1.38	1.40	1.42
	75	1.59	1.60	1.64	1.53	1.53	1.56	1.51	1.52	1.55
	65	1.68	1.69	1.75	1.61	1.61	1.66	1.60	1.61	1.69
	50	1.85	1.86	---	1.76	1.76	---	1.83	1.84	---
	50	1.93	1.94	1.96	1.81	1.81	1.84	1.85	1.86	1.95
	40	2.14	2.15	2.26	1.99	1.99	2.09	2.10	2.11	2.24
	30	2.51	2.52	2.67	2.28	2.28	2.42	2.44	2.45	2.62
	30 at > 50%F	2.65	2.66	2.79	2.59	2.59	2.71	2.62	2.63	2.74
	25 at > 50%F	2.93	2.94	3.07	2.87	2.87	3.01	2.88	2.89	3.04
	30 at ≤ 50%F	2.58	2.59	2.69	2.50	2.50	2.60	2.57	2.58	2.69
	25 at ≤ 50%F	2.77	2.78	2.92	2.73	2.73	2.87	2.81	2.82	2.98
FHOOS	100	1.47	1.48	---	1.43	1.43	---	1.41	1.42	---
	75	1.62	1.64	---	1.56	1.56	---	1.54	1.55	---
	65	1.74	1.75	---	1.66	1.66	---	1.68	1.69	---
	50	---	---	---	---	---	---	---	---	---
	50	1.95	1.96	---	1.84	1.84	---	1.94	1.95	---
	40	2.25	2.26	---	2.09	2.09	---	2.23	2.24	---
	30	2.66	2.67	---	2.42	2.42	---	2.61	2.62	---
	30 at > 50%F	2.78	2.79	---	2.71	2.71	---	2.73	2.74	---
	25 at > 50%F	3.06	3.07	---	3.01	3.01	---	3.03	3.04	---
	30 at ≤ 50%F	2.68	2.69	---	2.60	2.60	---	2.68	2.69	---
	25 at ≤ 50%F	2.91	2.92	---	2.87	2.87	---	2.97	2.98	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR/FHOOS is supported for the BOC to End of Coast limits.

Table 4.3 MCPR$_P$ Limits for All Fuel Types: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10			ATRIUM-10XM			ATRIUM-11		
		BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
Base Case	100	1.44	1.47	1.50	1.41	1.42	1.45	1.40	1.42	1.44
	75	1.60	1.61	1.64	1.54	1.55	1.57	1.53	1.54	1.59
	65	1.69	1.70	1.76	1.62	1.62	1.67	1.64	1.65	1.73
	50	1.90	1.91	---	1.79	1.79	---	---	---	---
	50	1.94	1.95	2.01	1.81	1.82	1.87	1.87	1.88	2.00
	40	2.19	2.20	2.32	2.03	2.03	2.12	2.15	2.16	2.29
	30	2.57	2.58	2.74	2.33	2.33	2.47	2.50	2.51	2.68
	30 at > 50%F	2.65	2.66	2.79	2.59	2.59	2.71	2.62	2.63	2.74
	25 at > 50%F	2.93	2.94	3.07	2.87	2.87	3.01	2.88	2.89	3.04
	30 at ≤ 50%F	2.58	2.59	2.74	2.50	2.50	2.60	2.57	2.58	2.69
	25 at ≤ 50%F	2.77	2.78	2.92	2.73	2.73	2.87	2.81	2.82	2.98
TBVOOS	100	1.50	1.52	1.54	1.45	1.46	1.48	1.44	1.46	1.47
	75	1.65	1.66	1.69	1.58	1.58	1.60	1.57	1.58	1.60
	65	1.75	1.76	1.81	1.66	1.66	1.70	1.65	1.66	1.74
	50	1.91	1.92	---	1.80	1.80	---	---	---	---
	50	1.94	1.95	2.01	1.81	1.82	1.88	1.88	1.89	2.00
	40	2.19	2.20	2.32	2.03	2.03	2.12	2.15	2.16	2.29
	30	2.57	2.58	2.74	2.33	2.33	2.47	2.50	2.51	2.68
	30 at > 50%F	3.26	3.27	3.41	3.09	3.09	3.22	3.23	3.24	3.38
	25 at > 50%F	3.67	3.68	3.80	3.52	3.52	3.63	3.61	3.62	3.76
	30 at ≤ 50%F	3.02	3.03	3.18	2.85	2.85	2.99	3.04	3.05	3.20
	25 at ≤ 50%F	3.45	3.46	3.67	3.32	3.32	3.50	3.45	3.46	3.65
FHOOS	100	1.48	1.50	---	1.44	1.45	---	1.43	1.44	---
	75	1.64	1.64	---	1.57	1.57	---	1.58	1.59	---
	65	1.75	1.76	---	1.67	1.67	---	1.72	1.73	---
	50	---	---	---	---	---	---	---	---	---
	50	2.00	2.01	---	1.87	1.87	---	1.99	2.00	---
	40	2.31	2.32	---	2.12	2.12	---	2.28	2.29	---
	30	2.73	2.74	---	2.47	2.47	---	2.67	2.68	---
	30 at > 50%F	2.78	2.79	---	2.71	2.71	---	2.73	2.74	---
	25 at > 50%F	3.06	3.07	---	3.01	3.01	---	3.03	3.04	---
	30 at ≤ 50%F	2.73	2.74	---	2.60	2.60	---	2.68	2.69	---
	25 at ≤ 50%F	2.91	2.92	---	2.87	2.87	---	2.97	2.98	---
PLUOOS	100	1.44	1.47	1.50	1.41	1.42	1.45	1.40	1.42	1.44
	75	1.60	1.61	1.64	1.54	1.55	1.57	1.53	1.54	1.59
	65	1.85	1.87	1.87	1.73	1.74	1.75	1.76	1.78	1.79
	50	---	---	---	---	---	---	---	---	---
	50	1.94	1.95	2.01	1.81	1.82	1.87	1.87	1.88	2.00
	40	2.19	2.20	2.32	2.03	2.03	2.12	2.15	2.16	2.29
	30	2.57	2.58	2.74	2.33	2.33	2.47	2.50	2.51	2.68
	30 at > 50%F	2.65	2.66	2.79	2.59	2.59	2.71	2.62	2.63	2.74
	25 at > 50%F	2.93	2.94	3.07	2.87	2.87	3.01	2.88	2.89	3.04
	30 at ≤ 50%F	2.58	2.59	2.74	2.50	2.50	2.60	2.57	2.58	2.69
	25 at ≤ 50%F	2.77	2.78	2.92	2.73	2.73	2.87	2.81	2.82	2.98

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.3 MCPR$_P$ Limits for All Fuel Types: Nominal Scram Time Basis (*continued*)*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
TBVOOS FHOOS	100	1.53	1.54	---	1.47	1.48	---	1.46	1.47	---
	75	1.68	1.69	---	1.60	1.60	---	1.59	1.60	---
	65	1.80	1.81	---	1.70	1.70	---	1.73	1.74	---
	50	---	---	---	---	---	---	---	---	---
	50	2.00	2.01	---	1.88	1.88	---	1.99	2.00	---
	40	2.31	2.32	---	2.12	2.12	---	2.28	2.29	---
	30	2.73	2.74	---	2.47	2.47	---	2.67	2.68	---
	30 at > 50%F	3.40	3.41	---	3.22	3.22	---	3.37	3.38	---
	25 at > 50%F	3.79	3.80	---	3.63	3.63	---	3.75	3.76	---
	30 at ≤ 50%F	3.17	3.18	---	2.99	2.99	---	3.19	3.20	---
	25 at ≤ 50%F	3.66	3.67	---	3.50	3.50	---	3.64	3.65	---
TBVOOS PLUOOS	100	1.50	1.52	1.54	1.45	1.46	1.48	1.44	1.46	1.47
	75	1.65	1.66	1.69	1.58	1.58	1.60	1.57	1.58	1.60
	65	1.85	1.87	1.87	1.73	1.74	1.75	1.76	1.78	1.79
	50	---	---	---	---	---	---	---	---	---
	50	1.94	1.95	2.01	1.81	1.82	1.88	1.88	1.89	2.00
	40	2.19	2.20	2.32	2.03	2.03	2.12	2.15	2.16	2.29
	30	2.57	2.58	2.74	2.33	2.33	2.47	2.50	2.51	2.68
	30 at > 50%F	3.26	3.27	3.41	3.09	3.09	3.22	3.23	3.24	3.38
	25 at > 50%F	3.67	3.68	3.80	3.52	3.52	3.63	3.61	3.62	3.76
	30 at ≤ 50%F	3.02	3.03	3.18	2.85	2.85	2.99	3.04	3.05	3.20
	25 at ≤ 50%F	3.45	3.46	3.67	3.32	3.32	3.50	3.45	3.46	3.65
FHOOS PLUOOS	100	1.48	1.50	---	1.44	1.45	---	1.43	1.44	---
	75	1.64	1.64	---	1.57	1.57	---	1.58	1.59	---
	65	1.85	1.87	---	1.73	1.74	---	1.76	1.78	---
	50	---	---	---	---	---	---	---	---	---
	50	2.00	2.01	---	1.87	1.87	---	1.99	2.00	---
	40	2.31	2.32	---	2.12	2.12	---	2.28	2.29	---
	30	2.73	2.74	---	2.47	2.47	---	2.67	2.68	---
	30 at > 50%F	2.78	2.79	---	2.71	2.71	---	2.73	2.74	---
	25 at > 50%F	3.06	3.07	---	3.01	3.01	---	3.03	3.04	---
	30 at ≤ 50%F	2.73	2.74	---	2.60	2.60	---	2.68	2.69	---
	25 at ≤ 50%F	2.91	2.92	---	2.87	2.87	---	2.97	2.98	---
TBVOOS FHOOS PLUOOS	100	1.53	1.54	---	1.47	1.48	---	1.46	1.47	---
	75	1.68	1.69	---	1.60	1.60	---	1.59	1.60	---
	65	1.85	1.87	---	1.73	1.74	---	1.76	1.78	---
	50	---	---	---	---	---	---	---	---	---
	50	2.00	2.01	---	1.88	1.88	---	1.99	2.00	---
	40	2.31	2.32	---	2.12	2.12	---	2.28	2.29	---
	30	2.73	2.74	---	2.47	2.47	---	2.67	2.68	---
	30 at > 50%F	3.40	3.41	---	3.22	3.22	---	3.37	3.38	---
	25 at > 50%F	3.79	3.80	---	3.63	3.63	---	3.75	3.76	---
	30 at ≤ 50%F	3.17	3.18	---	2.99	2.99	---	3.19	3.20	---
	25 at ≤ 50%F	3.66	3.67	---	3.50	3.50	---	3.64	3.65	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.4 MCPR$_P$ Limits for All Fuel Types: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
Base Case	100	1.47	1.48	1.51	1.43	1.43	1.46	1.42	1.43	1.45
	75	1.61	1.62	1.65	1.55	1.55	1.58	1.55	1.56	1.63
	65	1.70	1.71	1.79	1.63	1.63	1.68	1.69	1.70	1.77
	50	---	---	---	---	1.82	---	---	---	---
	50	1.95	1.96	2.07	1.82	1.83	1.90	1.92	1.93	2.05
	40	2.25	2.26	2.37	2.06	2.06	2.16	2.20	2.21	2.34
	30	2.64	2.65	2.81	2.38	2.38	2.52	2.55	2.56	2.74
	30 at > 50%F	2.65	2.66	2.81	2.59	2.59	2.71	2.62	2.63	2.74
	25 at > 50%F	2.93	2.94	3.07	2.87	2.87	3.01	2.88	2.89	3.04
	30 at ≤ 50%F	2.64	2.65	2.81	2.50	2.50	2.60	2.57	2.58	2.74
	25 at ≤ 50%F	2.77	2.78	2.92	2.73	2.73	2.87	2.81	2.82	2.98
TBVOOS	100	1.52	1.54	1.56	1.47	1.47	1.48	1.46	1.47	1.49
	75	1.66	1.67	1.70	1.59	1.59	1.61	1.58	1.59	1.64
	65	1.76	1.77	1.82	1.67	1.67	1.71	1.70	1.71	1.78
	50	---	---	---	---	---	---	---	---	---
	50	1.96	1.97	2.07	1.83	1.83	1.91	1.93	1.94	2.06
	40	2.26	2.27	2.38	2.07	2.07	2.16	2.20	2.21	2.34
	30	2.64	2.65	2.81	2.38	2.38	2.52	2.55	2.56	2.74
	30 at > 50%F	3.26	3.27	3.41	3.09	3.09	3.22	3.23	3.24	3.38
	25 at > 50%F	3.67	3.68	3.80	3.52	3.52	3.63	3.61	3.62	3.76
	30 at ≤ 50%F	3.02	3.03	3.18	2.85	2.85	2.99	3.04	3.05	3.20
	25 at ≤ 50%F	3.45	3.46	3.67	3.32	3.32	3.50	3.45	3.46	3.65
FHOOS	100	1.50	1.51	---	1.46	1.46	---	1.44	1.45	---
	75	1.64	1.65	---	1.58	1.58	---	1.62	1.63	---
	65	1.78	1.79	---	1.68	1.68	---	1.76	1.77	---
	50	---	---	---	---	---	---	---	---	---
	50	2.06	2.07	---	1.90	1.90	---	2.04	2.05	---
	40	2.36	2.37	---	2.16	2.16	---	2.33	2.34	---
	30	2.80	2.81	---	2.52	2.52	---	2.73	2.74	---
	30 at > 50%F	2.80	2.81	---	2.71	2.71	---	2.73	2.74	---
	25 at > 50%F	3.06	3.07	---	3.01	3.01	---	3.03	3.04	---
	30 at ≤ 50%F	2.80	2.81	---	2.60	2.60	---	2.73	2.74	---
	25 at ≤ 50%F	2.91	2.92	---	2.87	2.87	---	2.97	2.98	---
PLUOOS	100	1.47	1.48	1.51	1.43	1.43	1.46	1.42	1.43	1.45
	75	1.61	1.62	1.65	1.55	1.55	1.58	1.55	1.56	1.63
	65	1.87	1.89	1.89	1.75	1.76	1.78	1.78	1.81	1.81
	50	---	---	---	---	---	---	---	---	---
	50	1.95	1.96	2.07	1.82	1.83	1.90	1.92	1.93	2.05
	40	2.25	2.26	2.37	2.06	2.06	2.16	2.20	2.21	2.34
	30	2.64	2.65	2.81	2.38	2.38	2.52	2.55	2.56	2.74
	30 at > 50%F	2.65	2.66	2.81	2.59	2.59	2.71	2.62	2.63	2.74
	25 at > 50%F	2.93	2.94	3.07	2.87	2.87	3.01	2.88	2.89	3.04
	30 at ≤ 50%F	2.64	2.65	2.81	2.50	2.50	2.60	2.57	2.58	2.74
	25 at ≤ 50%F	2.77	2.78	2.92	2.73	2.73	2.87	2.81	2.82	2.98

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.4 MCPRP Limits for All Fuel Types: Technical Specification Scram Time Basis (*continued*)*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
TBVOOS FHOOS	100	1.54	1.56	---	1.48	1.48	---	1.47	1.49	---
	75	1.69	1.70	---	1.61	1.61	---	1.63	1.64	---
	65	1.81	1.82	---	1.71	1.71	---	1.77	1.78	---
	50	---	---	---	---	---	---	---	---	---
	50	2.06	2.07	---	1.91	1.91	---	2.05	2.06	---
	40	2.37	2.38	---	2.16	2.16	---	2.33	2.34	---
	30	2.80	2.81	---	2.52	2.52	---	2.73	2.74	---
	30 at > 50%F	3.40	3.41	---	3.22	3.22	---	3.37	3.38	---
	25 at > 50%F	3.79	3.80	---	3.63	3.63	---	3.75	3.76	---
	30 at ≤ 50%F	3.17	3.18	---	2.99	2.99	---	3.19	3.20	---
	25 at ≤ 50%F	3.66	3.67	---	3.50	3.50	---	3.64	3.65	---
TBVOOS PLUOOS	100	1.52	1.54	1.56	1.47	1.47	1.48	1.46	1.47	1.49
	75	1.66	1.67	1.70	1.59	1.59	1.61	1.58	1.59	1.64
	65	1.87	1.89	1.89	1.75	1.76	1.78	1.78	1.81	1.81
	50	---	---	---	---	---	---	---	---	---
	50	1.96	1.97	2.07	1.83	1.83	1.91	1.93	1.94	2.06
	40	2.26	2.27	2.38	2.07	2.07	2.16	2.20	2.21	2.34
	30	2.64	2.65	2.81	2.38	2.38	2.52	2.55	2.56	2.74
	30 at > 50%F	3.26	3.27	3.41	3.09	3.09	3.22	3.23	3.24	3.38
	25 at > 50%F	3.67	3.68	3.80	3.52	3.52	3.63	3.61	3.62	3.76
	30 at ≤ 50%F	3.02	3.03	3.18	2.85	2.85	2.99	3.04	3.05	3.20
	25 at ≤ 50%F	3.45	3.46	3.67	3.32	3.32	3.50	3.45	3.46	3.65
FHOOS PLUOOS	100	1.50	1.51	---	1.46	1.46	---	1.44	1.45	---
	75	1.64	1.65	---	1.58	1.58	---	1.62	1.63	---
	65	1.87	1.89	---	1.75	1.76	---	1.78	1.81	---
	50	---	---	---	---	---	---	---	---	---
	50	2.06	2.07	---	1.90	1.90	---	2.04	2.05	---
	40	2.36	2.37	---	2.16	2.16	---	2.33	2.34	---
	30	2.80	2.81	---	2.52	2.52	---	2.73	2.74	---
	30 at > 50%F	2.80	2.81	---	2.71	2.71	---	2.73	2.74	---
	25 at > 50%F	3.06	3.07	---	3.01	3.01	---	3.03	3.04	---
	30 at ≤ 50%F	2.80	2.81	---	2.60	2.60	---	2.73	2.74	---
	25 at ≤ 50%F	2.91	2.92	---	2.87	2.87	---	2.97	2.98	---
TBVOOS FHOOS PLUOOS	100	1.54	1.56	---	1.48	1.48	---	1.47	1.49	---
	75	1.69	1.70	---	1.61	1.61	---	1.63	1.64	---
	65	1.87	1.89	---	1.75	1.76	---	1.78	1.81	---
	50	---	---	---	---	---	---	---	---	---
	50	2.06	2.07	---	1.91	1.91	---	2.05	2.06	---
	40	2.37	2.38	---	2.16	2.16	---	2.33	2.34	---
	30	2.80	2.81	---	2.52	2.52	---	2.73	2.74	---
	30 at > 50%F	3.40	3.41	---	3.22	3.22	---	3.37	3.38	---
	25 at > 50%F	3.79	3.80	---	3.63	3.63	---	3.75	3.76	---
	30 at ≤ 50%F	3.17	3.18	---	2.99	2.99	---	3.19	3.20	---
	25 at ≤ 50%F	3.66	3.67	---	3.50	3.50	---	3.64	3.65	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.5 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 1 for All Fuel Types: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10			ATRIUM-10XM			ATRIUM-11		
		BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVIS	100	1.48	1.50	1.50	1.44	1.45	1.45	1.43	1.44	1.44
	75	1.64	1.64	1.64	1.57	1.57	1.57	1.58	1.59	1.59
	65	1.85	1.87	1.87	1.73	1.74	1.75	1.76	1.78	1.79
	50	---	---	---	---	---	---	---	---	---
	50	2.23	2.24	2.24	2.07	2.07	2.07	2.26	2.27	2.27
	40	2.60	2.61	2.61	2.38	2.38	2.38	2.62	2.63	2.63
	30	3.10	3.11	3.11	2.79	2.79	2.79	3.05	3.06	3.06
	30 at > 50%F	3.10	3.11	3.11	2.98	2.98	2.98	3.05	3.06	3.06
	25 at > 50%F	3.42	3.43	3.43	3.30	3.30	3.30	3.39	3.40	3.40
	30 at ≤ 50%F	3.10	3.11	3.11	2.86	2.86	2.86	3.05	3.06	3.06
	25 at ≤ 50%F	3.38	3.39	3.39	3.18	3.18	3.18	3.30	3.31	3.31
TBVOOS	100	1.53	1.54	1.54	1.47	1.48	1.48	1.46	1.47	1.47
	75	1.68	1.69	1.69	1.60	1.60	1.60	1.59	1.60	1.60
	65	1.85	1.87	1.87	1.73	1.74	1.75	1.76	1.78	1.79
	50	---	---	---	---	---	---	---	---	---
	50	2.23	2.24	2.24	2.07	2.07	2.07	2.26	2.27	2.27
	40	2.60	2.61	2.61	2.38	2.38	2.38	2.62	2.63	2.63
	30	3.10	3.11	3.11	2.79	2.79	2.79	3.05	3.06	3.06
	30 at > 50%F	3.60	3.61	3.61	3.42	3.42	3.42	3.62	3.63	3.63
	25 at > 50%F	4.03	4.04	4.04	3.86	3.86	3.86	4.03	4.04	4.04
	30 at ≤ 50%F	3.43	3.44	3.44	3.24	3.24	3.24	3.43	3.44	3.44
	25 at ≤ 50%F	3.93	3.94	3.94	3.78	3.78	3.78	3.94	3.95	3.95

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.6 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 2 for All Fuel Types: Nominal Scram Time Basis*

		ATRIUM-10			ATRIUM-10XM			ATRIUM-11		
Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVIS	100	1.48	1.50	1.50	1.44	1.45	1.45	1.43	1.44	1.44
	75	1.64	1.64	1.64	1.57	1.57	1.57	1.58	1.59	1.59
	65	1.85	1.87	1.87	1.73	1.74	1.75	1.76	1.78	1.79
	50	---	---	---	---	---	---	---	---	---
	50	2.25	2.26	2.26	2.08	2.08	2.08	2.28	2.29	2.29
	40	2.62	2.63	2.63	2.39	2.39	2.39	2.64	2.65	2.65
	30	3.12	3.13	3.13	2.82	2.82	2.82	3.08	3.09	3.09
	30 at > 50%F	3.12	3.13	3.13	2.99	2.99	2.99	3.08	3.09	3.09
	25 at > 50%F	3.44	3.45	3.45	3.32	3.32	3.32	3.41	3.42	3.42
	30 at ≤ 50%F	3.12	3.13	3.13	2.88	2.88	2.88	3.08	3.09	3.09
	25 at ≤ 50%F	3.40	3.41	3.41	3.20	3.20	3.20	3.32	3.33	3.33
TBVOOS	100	1.53	1.54	1.54	1.47	1.48	1.48	1.46	1.47	1.47
	75	1.68	1.69	1.69	1.60	1.60	1.60	1.59	1.60	1.60
	65	1.85	1.87	1.87	1.73	1.74	1.75	1.76	1.78	1.79
	50	---	---	---	---	---	---	---	---	---
	50	2.25	2.26	2.26	2.08	2.08	2.08	2.28	2.29	2.29
	40	2.62	2.63	2.63	2.39	2.39	2.39	2.64	2.65	2.65
	30	3.12	3.13	3.13	2.82	2.82	2.82	3.08	3.09	3.09
	30 at > 50%F	3.62	3.63	3.63	3.43	3.43	3.43	3.64	3.65	3.65
	25 at > 50%F	4.04	4.05	4.05	3.88	3.88	3.88	4.05	4.06	4.06
	30 at ≤ 50%F	3.45	3.46	3.46	3.26	3.26	3.26	3.45	3.46	3.46
	25 at ≤ 50%F	3.95	3.96	3.96	3.80	3.80	3.80	3.96	3.97	3.97

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.7 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 1 for All Fuel Types: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
TBVIS	100	1.50	1.51	1.51	1.46	1.46	1.46	1.44	1.45	1.45
	75	1.64	1.65	1.65	1.58	1.58	1.58	1.62	1.63	1.63
	65	1.87	1.89	1.89	1.75	1.76	1.78	1.78	1.81	1.81
	50	---	---	---	---	---	---	---	---	---
	50	2.29	2.30	2.30	2.10	2.10	2.10	2.32	2.33	2.33
	40	2.66	2.67	2.67	2.42	2.42	2.42	2.68	2.69	2.69
	30	3.17	3.18	3.18	2.85	2.85	2.85	3.12	3.13	3.13
	30 at > 50%F	3.17	3.18	3.18	2.98	2.98	2.98	3.12	3.13	3.13
	25 at > 50%F	3.42	3.43	3.43	3.30	3.30	3.30	3.39	3.40	3.40
	30 at ≤ 50%F	3.17	3.18	3.18	2.86	2.86	2.86	3.12	3.13	3.13
	25 at ≤ 50%F	3.38	3.39	3.39	3.18	3.18	3.18	3.30	3.31	3.31
TBVOOS	100	1.54	1.56	1.56	1.48	1.48	1.48	1.47	1.49	1.49
	75	1.69	1.70	1.70	1.61	1.61	1.61	1.63	1.64	1.64
	65	1.87	1.89	1.89	1.75	1.76	1.78	1.78	1.81	1.81
	50	---	---	---	---	---	---	---	---	---
	50	2.29	2.30	2.30	2.11	2.11	2.11	2.32	2.33	2.33
	40	2.66	2.67	2.67	2.42	2.42	2.42	2.68	2.69	2.69
	30	3.17	3.18	3.18	2.85	2.85	2.85	3.12	3.13	3.13
	30 at > 50%F	3.60	3.61	3.61	3.42	3.42	3.42	3.62	3.63	3.63
	25 at > 50%F	4.03	4.04	4.04	3.86	3.86	3.86	4.03	4.04	4.04
	30 at ≤ 50%F	3.43	3.44	3.44	3.24	3.24	3.24	3.43	3.44	3.44
	25 at ≤ 50%F	3.93	3.94	3.94	3.78	3.78	3.78	3.94	3.95	3.95

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.8 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 2 for All Fuel Types: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
TBVIS	100	1.50	1.51	1.51	1.46	1.46	1.46	1.44	1.45	1.45
	75	1.64	1.65	1.65	1.58	1.58	1.58	1.62	1.63	1.63
	65	1.87	1.89	1.89	1.75	1.76	1.78	1.78	1.81	1.81
	50	---	---	---	---	---	---	---	---	---
	50	2.31	2.32	2.32	2.11	2.11	2.11	2.33	2.34	2.34
	40	2.68	2.69	2.69	2.43	2.43	2.43	2.70	2.71	2.71
	30	3.20	3.21	3.21	2.88	2.88	2.88	3.15	3.16	3.16
	30 at > 50%F	3.20	3.21	3.21	2.99	2.99	2.99	3.15	3.16	3.16
	25 at > 50%F	3.44	3.45	3.45	3.32	3.32	3.32	3.41	3.42	3.42
	30 at ≤ 50%F	3.20	3.21	3.21	2.88	2.88	2.88	3.15	3.16	3.16
	25 at ≤ 50%F	3.40	3.41	3.41	3.20	3.20	3.20	3.32	3.33	3.33
TBVOOS	100	1.54	1.56	1.56	1.48	1.48	1.48	1.47	1.49	1.49
	75	1.69	1.70	1.70	1.61	1.61	1.61	1.63	1.64	1.64
	65	1.87	1.89	1.89	1.75	1.76	1.78	1.78	1.81	1.81
	50	---	---	---	---	---	---	---	---	---
	50	2.31	2.32	2.32	2.12	2.12	2.12	2.33	2.34	2.34
	40	2.68	2.69	2.69	2.43	2.43	2.43	2.70	2.71	2.71
	30	3.20	3.21	3.21	2.88	2.88	2.88	3.15	3.16	3.16
	30 at > 50%F	3.62	3.63	3.63	3.43	3.43	3.43	3.64	3.65	3.65
	25 at > 50%F	4.04	4.05	4.05	3.88	3.88	3.88	4.05	4.06	4.06
	30 at ≤ 50%F	3.45	3.46	3.46	3.26	3.26	3.26	3.45	3.46	3.46
	25 at ≤ 50%F	3.95	3.96	3.96	3.80	3.80	3.80	3.96	3.97	3.97

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.9 MCPR$_P$ Limits for All Fuel Types: Single Loop Operation for All Scram Times*

Operating Condition	Power (% of rated)	BOC to End of COAST		
		ATRIUM-10	ATRIUM-10XM	ATRIUM-11
RCPOOS FHOOS	100	2.09	2.01	2.07
	50	2.09	2.01	2.07
	40	2.39	2.18	2.36
	30	2.83	2.54	2.76
	30 at > 50%F	2.83	2.73	2.76
	25 at > 50%F	3.09	3.03	3.06
	30 at ≤ 50%F	2.83	2.62	2.76
	25 at ≤ 50%F	2.94	2.89	3.00
RCPOOS TBVOOS PLUOOS FHOOS	100	2.09	2.01	2.08
	50	2.09	2.01	2.08
	40	2.40	2.18	2.36
	30	2.83	2.54	2.76
	30 at > 50%F	3.43	3.24	3.40
	25 at > 50%F	3.82	3.65	3.78
	30 at ≤ 50%F	3.20	3.01	3.22
	25 at ≤ 50%F	3.69	3.52	3.67
RCPOOS TBVOOS FHOOS1	100	2.32	2.13	2.35
	50	2.32	2.13	2.35
	40	2.69	2.44	2.71
	30	3.20	2.87	3.15
	30 at > 50%F	3.63	3.44	3.65
	25 at > 50%F	4.06	3.88	4.06
	30 at ≤ 50%F	3.46	3.26	3.46
	25 at ≤ 50%F	3.96	3.80	3.97
RCPOOS TBVOOS FHOOS2	100	2.34	2.14	2.36
	50	2.34	2.14	2.36
	40	2.71	2.45	2.73
	30	3.23	2.90	3.18
	30 at > 50%F	3.65	3.45	3.67
	25 at > 50%F	4.07	3.90	4.08
	30 at ≤ 50%F	3.48	3.28	3.48
	25 at ≤ 50%F	3.98	3.82	3.99

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop.

RCPOOS limits are only valid up to 50% rated core power, 50% rated core flow, and an active recirculation drive flow of 17.73 Mlbm/hr.

5 Oscillation Power Range Monitor (OPRM) Setpoint

(Technical Specification 3.3.1.1)

Technical Specification Table 3.3.1.1-1, Function 2f, identifies the OPRM upscale function.

Instrument setpoints are established, such that the reactor will be tripped before an oscillation can grow to the point where the SLMCPR is exceeded. An Option III stability analysis is performed for each reload core to determine allowable OLMCPR's as a function of OPRM setpoint. Analyses consider both steady state startup operation, and the case of a two recirculation pump trip from rated power.

The resulting stability based OLMCPR's are reported in Reference 1. The OPRM setpoint (*sometimes referred to as the Amplitude Trip, S_p*) is selected, such that required margin to the SLMCPR is provided without stability being a limiting event. Analyses are based on cycle specific DIVOM analyses performed per Reference 22. The calculated OLMCPR's are shown in Table 5.1. Review of results shown in COLR Table 4.2 indicates an OPRM setpoint of **1.14** may be used. The successive confirmation count (*sometimes referred to as N_p*) is provided in Table 5.2, per Reference 30.

Table 5.1 OPRM Setpoint Range*

OPRM Setpoint	OLMCPR (SS)	OLMCPR (2PT)
1.05	1.15	1.16
1.06	1.17	1.18
1.07	1.19	1.19
1.08	1.20	1.21
1.09	1.22	1.23
1.10	1.24	1.25
1.11	1.26	1.27
1.12	1.28	1.29
1.13	1.30	1.31
1.14	1.33	1.34
1.15	1.35	1.36

Table 5.2 OPRM Successive Confirmation Count Setpoint

Count	OPRM Setpoint
6	≥ 1.04
8	≥ 1.05
10	≥ 1.07
12	≥ 1.09
14	≥ 1.11
16	≥ 1.14
18	≥ 1.18
20	≥ 1.24

* Extrapolation beyond a setpoint of 1.15 is not allowed

6 APRM Flow Biased Rod Block Trip Settings

(Technical Requirements Manual Section 5.3.1 and Table 3.3.4-1)

The APRM rod block trip setting is based upon References 26 & 27, and is defined by the following:

$SRB \leq (0.66(W-\Delta W) + 61\%)$	Allowable Value
$SRB \leq (0.66(W-\Delta W) + 59\%)$	Nominal Trip Setpoint (NTSP)

where:

SRB	=	Rod Block setting in percent of rated thermal power (3458 MW_t)
W	=	Loop recirculation flow rate in percent of rated
ΔW	=	Difference between two-loop and single-loop effective recirculation flow at the same core flow (ΔW=0.0 for two-loop operation)

The APRM rod block trip setting is clamped at a maximum allowable value of 115% (corresponding to a NTSP of 113%).

7 Rod Block Monitor (RBM) Trip Setpoints and Operability

(Technical Specification Table 3.3.2.1-1)

The RBM trip setpoints and applicable power ranges, based on References 26 & 27, are shown in Table 7.1. Setpoints are based on an HTSP, unfiltered analytical limit of 114%. Unfiltered setpoints are consistent with a nominal RBM filter setting of 0.0 seconds; filtered setpoints are consistent with a nominal RBM filter setting less than 0.5 seconds. Cycle specific CRWE analyses of OLMCPR are documented in Reference 1, superseding values reported in References 26, 27, and 29.

Table 7.1 Analytical RBM Trip Setpoints*

RBM Trip Setpoint		Allowable Value (AV)	Nominal Trip Setpoint (NTSP)
LPSP		27%	25%
IPSP		62%	60%
HPSP		82%	80%
LTSP	- unfiltered	121.7%	120.0%
	- filtered	120.7%	119.0%
ITSP	- unfiltered	116.7%	115.0%
	- filtered	115.7%	114.0%
HTSP	- unfiltered	111.7%	110.0%
	- filtered	110.9%	109.2%
DTSP		90%	92%

As a result of cycle specific CRWE analyses, RBM setpoints in Technical Specification Table 3.3.2.1-1 are applicable as shown in Table 7.2. Cycle specific setpoint analysis results are shown in Table 7.3, per Reference 1.

Table 7.2 RBM Setpoint Applicability

Thermal Power (% Rated)	Applicable MCPR†	Notes from Table 3.3.2.1-1	Comment
> 27% and < 90%	< 1.70	(a), (b), (f), (h)	two loop operation
	< 1.74	(a), (b), (f), (h)	single loop operation
≥ 90%	< 1.39	(g)	two loop operation‡

* Values are considered maximums. Using lower values, due to RBM system hardware/software limitations, is conservative, and acceptable.

† MCPR values shown correspond with, (support), SLMPCR values identified in Reference 1.

‡ Greater than 90% rated power is not attainable in single loop operation.

Table 7.3 Control Rod Withdrawal Error Results

RBM HTSP Analytical Limit	CRWE OLMCPR
Unfiltered	
107	1.20
111	1.26
114	1.37
117	1.37

Results, compared against the base case OLMCPR results of Table 4.2, indicate SLMCPR remains protected for RBM inoperable conditions (i.e., 114% unblocked).

8 Shutdown Margin Limit

(Technical Specification 3.1.1)

Assuming the strongest OPERABLE control blade is fully withdrawn, and all other OPERABLE control blades are fully inserted, the core shall be sub-critical and meet the following minimum shutdown margin:

SDM > 0.38% dk/k

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1101 Market Street, Chattanooga, TN 37402

Browns Ferry Unit 2 Cycle 19
Core Operating Limits Report, (105% OLTP)

TVA-COLR-BF2C19 Revision 4 (Final)
(Revision Log, Page v)

January 2016

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Revision Log

Number	Page	Description
0-R4		Revision driven by CRs 1090560 and 1097540: code errors
1-R4	ix	Updated Reference 1.
2-R4	1	Revised Section 1.1.1; identify new basis of revised reload safety analysis report.
3-R4	39	Updated Table 5.1 2 pump trip column of results due to change in Reference 1
0-R3		No Changes to Section 2; Sections 5 through 8.
1-R3	ix	Updated References 1 & 4.
2-R3	xi	Removed References 31 & 32 as they are no longer applicable due to Reference 1 update.
3-R3	1	Revised Section 1.1.1; identify new basis of revised reload safety analysis report.
4-R3	8	Revised Section 3.2; removed cross references coincident with change 2-R3. Also, revised Section 3.2.1; removed last sentence as items no longer applicable. Updated pointers to Figure numbers.
5-R3	9	Revised Section 3.4; updated Figure number cross references coincident with change 8-R3.
6-R3	13-15, 19-24	Revised Section 3, Figures 3.4 -3.6; 3.10 - 3.15 based on new reload safety analysis report.
7-R3	old 25-26	Removed Figures 3.16 - 3.17 as they are no longer applicable.
8-R3	25	Revised Section 4.2.1; removed last sentence as items no longer applicable. Update pointers to Table numbers.
9-R3	27	Section 4.2.5 updated Figure number cross reference.
10-R3	29-38	Updated Tables 4.2 through 4.9 with the results from revised reload safety analysis report.
11-R3	old 40-43	Removed old Tables 4.9 - 4.12 as they are no longer applicable.
1-R2	x	Added Reference 32 supplement to BFN Reload Analysis Report
2-R2	1	Revised Section 1.1.1 to identify the new range of applicability for startup with feedwater heating out of service
3-R2	8	Revised last sentence of Section 3.2.1 to reflect the updated cycle exposure range of applicability (3000 MWd/MTU); also added Reference 32
4-R2	25-26	Revised Section 3, Figure 3.16-17 titles to reflect the updated cycle exposure range of applicability (3000 MWd/MTU). No limits were changed, only titles.
5-R2	27	Revised last sentence of Section 4.2.1 to reflect the updated cycle exposure range of applicability (3000 MWd/MTU); also added

		Reference 32.
6-R2	40-43	Revised Section 4, Tables 4.9 - 4.12 to reflect the updated cycle exposure range of applicability (3000 MWd/MTU). No limits were changed, only titles
1-R1	x	Added Reference 31 supplement to BFN Unit 2 Cycle 19 Reload Analysis
2-R1	1-49	Corrected page numbering. Revision 0 did not reset the page numbering at Section 1 to start at 1, after the roman numeral sections at the front of the document . The new Arabic numbered pages for Sections 1 through 8 begin with page 1 and go through page 49.
3-R1	1	Added Section 1.1.1 to describe Revision 1 purpose
4-R1	8	Added discussion of Reference 31 for $LHGRFAC_p$
5-R1	9	Figure number range increased to Figure 3.17
6-R1	25-26	Added Figure 3.16 & Figure 3.17, $LHGRFAC_p$ for ATRIUM-10 fuel BOC - 250 MWd/MTU
7-R1	27	Added discussion of Reference 31 for $MCPR_p$
8-R1	29	Renumbered Table 4.9 as Table 4.13
9-R1	40-43	Inserted Table 4.9 through Table 4.12, BOC – 250 MWd/MTU Startup Operation $MCPR_p$ Limits
10-R1	44	Renumbered Table 4.9 as Table 4.13
0-R0	All	New document.

Nomenclature

APLHGR	Average Planar LHGR
APRM	Average Power Range Monitor
AREVA NP	Vendor (Framatome, Siemens)
BOC	Beginning of Cycle
BSP	Backup Stability Protection
BWR	Boiling Water Reactor
CAVEX	Core Average Exposure
CD	Coast Down
CMSS	Core Monitoring System Software
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRWE	Control Rod Withdrawal Error
CSDM	Cold SDM
DIVOM	Delta CPR over Initial CPR vs. Oscillation Magnitude
EOC	End of Cycle
EOCLB	End-of-Cycle Licensing Basis
EOOS	Equipment OOS
FFTR	Final Feedwater Temperature Reduction
FFWTR	Final Feedwater Temperature Reduction
FHOOS	Feedwater Heaters OOS
ft	Foot: english unit of measure for length
GNF	Vendor (General Electric, Global Nuclear Fuels)
GWd	Giga Watt Day
HTSP	High TSP
ICA	Interim Corrective Action
ICF	Increased Core Flow (beyond rated)
IS	In-Service
kW	kilo watt: SI unit of measure for power.
LCO	License Condition of Operation
LFWH	Loss of Feedwater Heating
LHGRFAC	LHGR Multiplier (Power or Flow dependent)
LPRM	Low Power Range Monitor
LRNB	Generator Load Reject, No Bypass
MAPFAC	MAPLHGR multiplier (Power or Flow dependent)

MCPR	Minimum CPR
MSRV	Moisture Separator Reheater Valve
MSRVOOS	MSRV OOS
MTU	Metric Ton Uranium
MWd/MTU	Mega Watt Day per Metric Ton Uranium
NEOC	Near EOC
NRC	United States Nuclear Regulatory Commission
NSS	Nominal Scram Speed
NTSP	Nominal TSP
OLMCPR	MCPR Operating Limit
OOS	Out-Of-Service
OPRM	Oscillation Power Range Monitor
OSS	Optimum Scram Speed
PBDA	Period Based Detection Algorithm
Pbypass	Power, below which TSV Position and TCV Fast Closure Scrams are Bypassed
PLU	Power Load Unbalance
PLUOOS	PLU OOS
PRNM	Power Range Neutron Monitor
RBM	Rod Block Monitor
RCPOOS	Recirculation Pump OOS (*SLO*)
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPTOOS	RPT OOS
SDM	Shutdown Margin
SLMCPR	MCPR Safety Limit
SLO	Single Loop Operation
TBV	Turbine Bypass Valve
TBVIS	TBV IS
TBVOOS	Turbine Bypass Valves OOS
TIP	Transversing In-core Probe
TIPOOS	TIP OOS
TLO	Two Loop Operation
TSP	Trip Setpoint
TSSS	Technical Specification Scram Speed
TVA	Tennessee Valley Authority

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1 Introduction

In anticipation of cycle startup, it is necessary to describe the expected limits of operation.

1.1 Purpose

The primary purpose of this document is to satisfy requirements identified by unit technical specification section 5.6.5. This document may be provided, upon final approval, to the NRC.

1.1.1 Revision 4

The purpose of this revision is a consequence of the Reference 1 reload safety analyses revision. The Reference 1 revision is in response to condition report (CR) 1090560 and 1097540 code errors.

1.2 Scope

This document will discuss the following areas:

- Average Planar Linear Heat Generation Rate (APLHGR) Limit
  (Technical Specifications 3.2.1 and 3.7.5)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Linear Heat Generation Rate (LHGR) Limit
  (Technical Specification 3.2.3, 3.3.4.1, and 3.7.5)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Minimum Critical Power Ratio Operating Limit (OLMCPR)
  (Technical Specifications 3.2.2, 3.3.4.1, 3.7.5 and Table 3.3.2.1-1)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Oscillation Power Range Monitor (OPRM) Setpoint
  (Technical Specification Table 3.3.1.1)
  Applicability: Mode 1, ≥ (as specified in Technical Specifications Table 3.3.1.1-1)
- Average Power Range Monitor (APRM) Flow Biased Rod Block Trip Setting
  (Technical Requirements Manual Section 5.3.1 and Table 3.3.4-1)
  Applicability: Mode 1, ≥ (as specified in Technical Requirements Manuals Table 3.3.4-1)
- Rod Block Monitor (RBM) Trip Setpoints and Operability
  (Technical Specification Table 3.3.2.1-1)
  Applicability: Mode 1, ≥ % RTP as specified in Table 3.3.2.1-1 (TS definition of RTP)
- Shutdown Margin (SDM) Limit
  (Technical Specification 3.1.1)
  Applicability: All Modes

1.3 Fuel Loading

The core will contain previously exposed AREVA NP, Inc., ATRIUM-10 fuel, along with fresh ATRIUM-10XM and ATRIUM-11 fuel. Nuclear fuel types used in the core loading are shown in Table 1.1. The core shuffle and final loading were explicitly evaluated for BOC cold shutdown margin performance as documented per Reference 5.

Table 1.1 Nuclear Fuel Types*

Fuel Description	Original Cycle	Number of Assemblies	Nuclear Fuel Type (NFT)	Fuel Names (Range)
ATRIUM-10 A10-3799B-14GV80-FBD	17	73	10	FBD001-FBD136
ATRIUM-10 A10-4004B-15GV80-FBD	17	115	11	FBD137-FBD272
ATRIUM-10 A10-4165B-15GV75-FBE	18	176	12	FBE001-FBE176
ATRIUM-10 A10-4107B-13GV75-FBE	18	68	13	FBE177-FBE244
ATRIUM-10 A10-4176B-10GV75-FBE	18	72	14	FBE245-FBE316
ATRIUM 10XM XMLC-3904B-15GV80-FBF	19	172	15	FBF401-FBF572
ATRIUM 10XM XMLC-4035B-13GV80-FBF	19	80	16	FBF573-FBF652
ATRIUM 11 A11-3693B-13GV80-FBF	19	8	17	FBF653-FBF660

1.4 Acceptability

Limits discussed in this document were generated based on NRC approved methodologies per References 6 through 25.

* The table identifies the expected fuel type breakdown in anticipation of final core loading. The final composition of the core depends upon uncertainties during the outage such as discovering a failed fuel bundle, or other bundle damage. Minor core loading changes, due to unforeseen events, will conform to the safety and monitoring requirements identified in this document.

2 APLHGR Limits

(Technical Specifications 3.2.1 & 3.7.5)

The APLHGR limit is determined by adjusting the rated power APLHGR limit for off-rated power, off-rated flow, and SLO conditions. The most limiting of these is then used as follows:

$$\text{APLHGR limit} = \text{MIN}\ (\text{APLHGR}_P\ ,\ \text{APLHGR}_F,\ \text{APLHGR}_{SLO}\ )$$

where:

$APLHGR_P$	off-rated power APLHGR limit	[$APLHGR_{RATED}$ * $MAPFAC_P$]
$APLHGR_F$	off-rated flow APLHGR limit	[$APLHGR_{RATED}$ * $MAPFAC_F$]
$APLHGR_{SLO}$	SLO APLHGR limit	[$APLHGR_{RATED}$ * SLO Multiplier]

2.1 Rated Power and Flow Limit: $APLHGR_{RATED}$

The rated conditions APLHGR for ATRIUM-10 fuel is identified in Reference 1 and shown in Figure 2.1. The rated conditions APLHGR for ATRIUM-10XM are shown in Figure 2.2. The rated conditions APLHGR for ATRIUM-11 are shown in Figure 2.3.

2.2 Off-Rated Power Dependent Limit: $APLHGR_P$

Reference 1 does not specify a power dependent APLHGR. Therefore, $MAPFAC_P$ is set to a value of **1.0**.

2.2.1 *Startup without Feedwater Heaters*

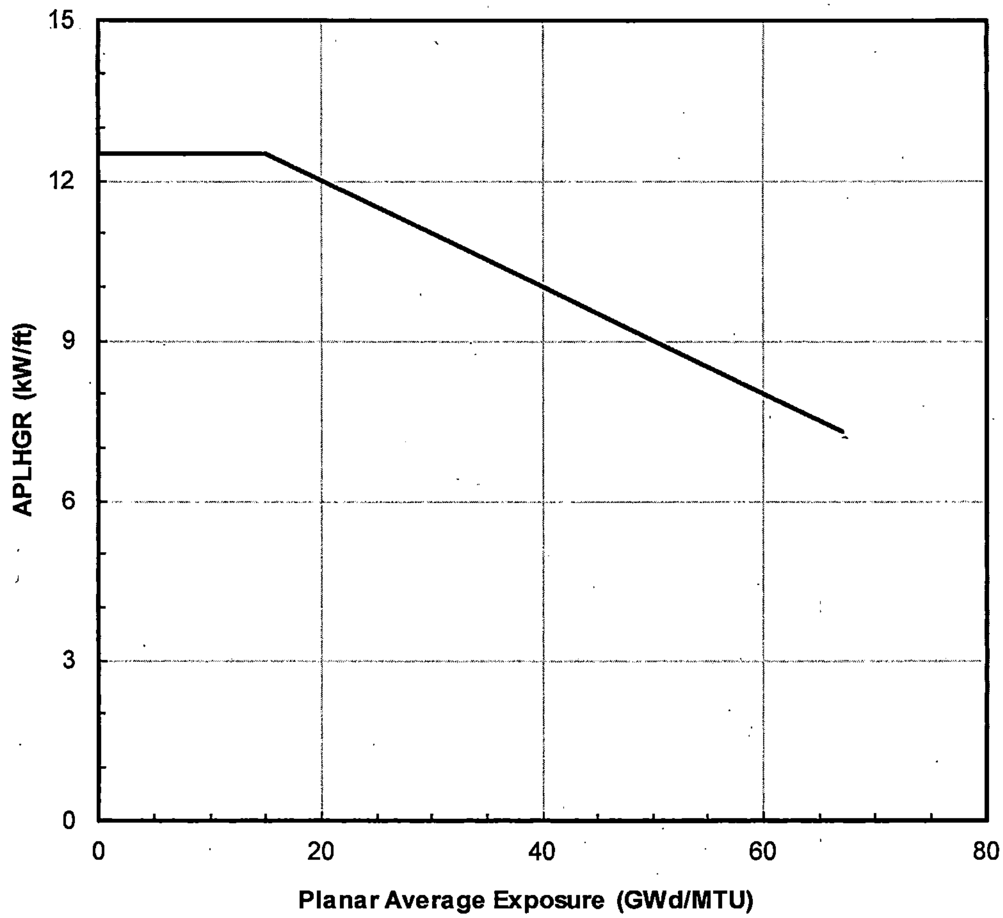
There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. No additional power dependent limitation is required.

2.3 Off-Rated Flow Dependent Limit: $APLHGR_F$

Reference 1 does not specify a flow dependent APLHGR. Therefore, $MAPFAC_F$ is set to a value of **1.0**.

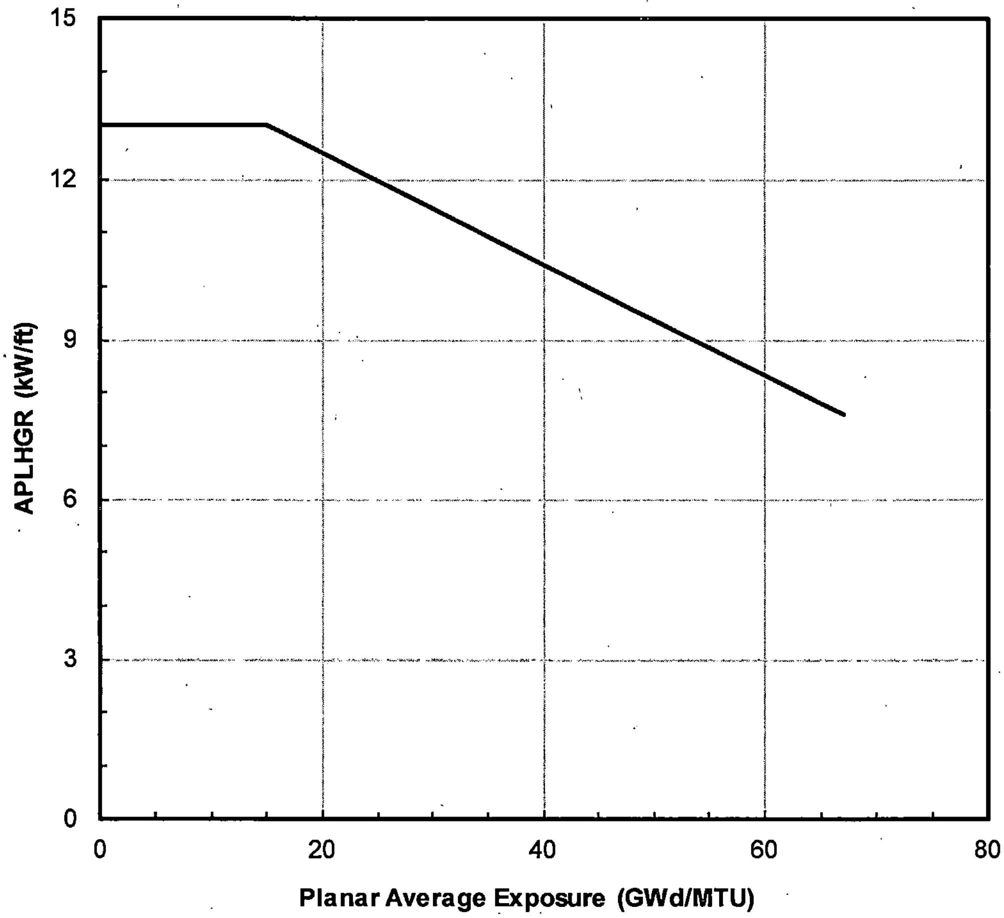
2.4 Single Loop Operation Limit: $APLHGR_{SLO}$

The single loop operation multiplier for ATRIUM-10, ATRIUM-10XM, and ATRIUM-11 fuel is **0.85**, per Reference 1.



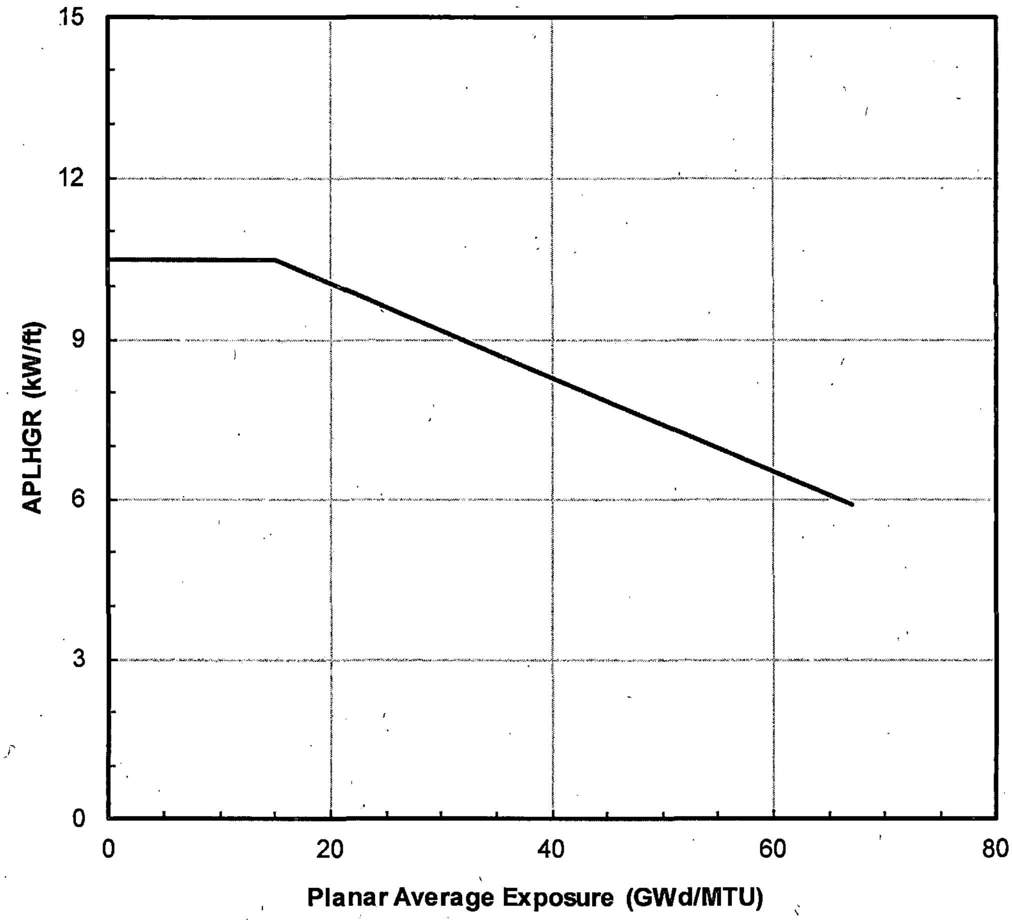
Planar Avg. Exposure (GWd/MTU)	APLHGR Limit (kW/ft)
0.0	12.5
15.0	12.5
67.0	7.3

Figure 2.1 APLHGR$_{RATED}$ for ATRIUM-10 Fuel



Planar Avg. Exposure (GWd/MTU)	APLHGR Limit (kW/ft)
0.0	13.0
15.0	13.0
67.0	7.6

Figure 2.2 APLHGR$_{RATED}$ for ATRIUM-10XM Fuel



Planar Avg. Exposure (GWd/MTU)	APLHGR Limit (kW/ft)
0.0	10.5
15.0	10.5
67.0	5.9

Figure 2.3 $APLHGR_{RATED}$ for ATRIUM-11 Fuel

2.5 Equipment Out-Of-Service Corrections

The limits shown in Figure 2.1, Figure 2.2, and Figure 2.3 are applicable for operation with all equipment In-Service as well as the following Equipment Out-Of-Service (EOOS) options; including combinations of the options.

In-Service	All equipment In-Service*
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
PLUOOS	Power Load Unbalance Out-Of-Service
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service or Final Feedwater Temperature Reduction
RCPOOS	One Recirculation Pump Out-Of-Service

* All equipment service conditions assume 1 SRVOOS.

3 LHGR Limits

(Technical Specification 3.2.3, 3.3.4.1, & 3.7.5)

The LHGR limit is determined by adjusting the rated power LHGR limit for off-rated power and off-rated flow conditions. The most limiting of these is then used as follows:

$$\text{LHGR limit} = \text{MIN} \left(\text{LHGR}_P, \text{LHGR}_F \right)$$

where:

$LHGR_P$	off-rated power LHGR limit	$[LHGR_{RATED} * LHGRFAC_P]$
$LHGR_F$	off-rated flow LHGR limit	$[LHGR_{RATED} * LHGRFAC_F]$

3.1 Rated Power and Flow Limit: $LHGR_{RATED}$

The rated conditions LHGR for ATRIUM-10 fuel is identified in Reference 1 and shown in Figure 3.1. The rated conditions LHGR for ATRIUM-10XM fuel is shown in Figure 3.2. The rated conditions LHGR for ATRIUM-11 fuel is shown in Figure 3.3. The LHGR limit is consistent with References 2 and 3.

3.2 Off-Rated Power Dependent Limit: $LHGR_P$

LHGR limits are adjusted for off-rated power conditions using the $LHGRFAC_P$ multiplier provided in Reference 1. The multiplier is split into two sub cases: turbine bypass valves in and out-of-service. The base case multipliers are shown in Figure 3.4 through Figure 3.6.

3.2.1 Startup without Feedwater Heaters

There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. Additional limits are shown in Figure 3.10 through Figure 3.15, based on temperature conditions identified in Table 3.1.

Table 3.1 Startup Feedwater Temperature Basis

Power	Temperature Range 1	Temperature Range 2
(% Rated)	(°F)	(°F)
25	160.0	155.0
30	165.0	160.0
40	175.0	170.0
50	185.0	180.0

3.3 Off-Rated Flow Dependent Limit: $LHGR_F$

LHGR limits are adjusted for off-rated flow conditions using the $LHGRFAC_F$ multiplier provided in Reference 1. Multipliers are shown in Figure 3.7 through Figure 3.9.

3.4 Equipment Out-Of-Service Corrections

The limits shown in Figure 3.1 through Figure 3.3 are applicable for operation with all equipment In-Service as well as the following Equipment Out-Of-Service (EOOS) options; including combinations of the options.*

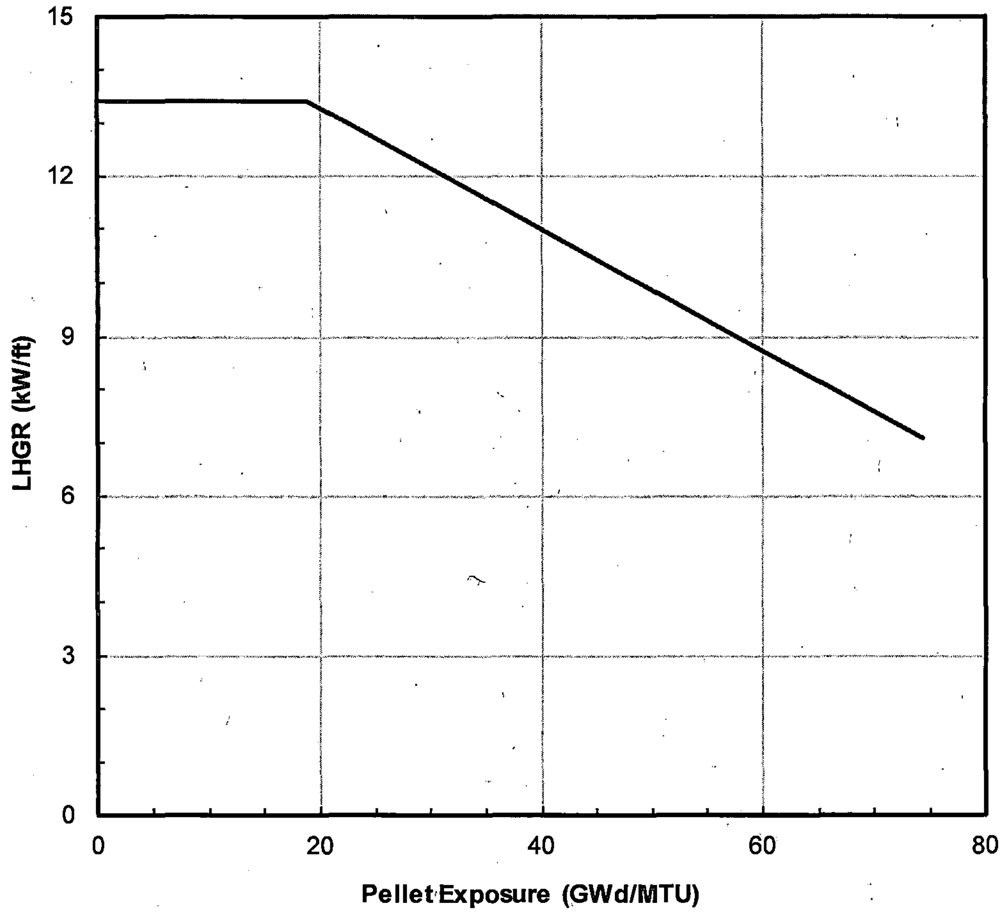
In-Service	All equipment In-Service
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
PLUOOS	Power Load Unbalance Out-Of-Service
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service or Final Feedwater Temperature Reduction
RCPOOS	One Recirculation Pump Out-Of-Service

Off-rated power corrections shown in Figure 3.4 through Figure 3.6 are dependent on operation of the Turbine Bypass Valve system. For this reason, separate limits are to be applied for TBVIS or TBVOOS operation. The limits have no dependency on RPTOOS, PLUOOS, FHOOS/FFWTR, or SLO.

Off-rated flow corrections shown in Figure 3.7 through Figure 3.9 are bounding for all EOOS conditions.

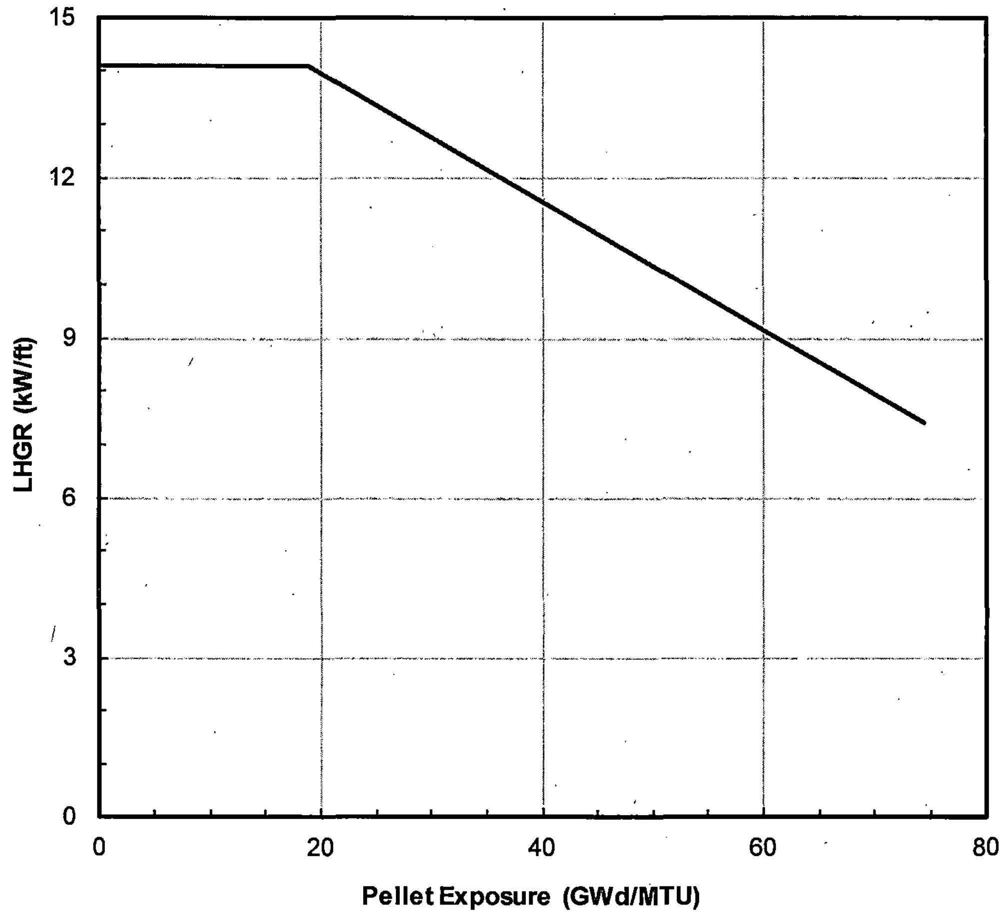
Off-rated power corrections shown in Figure 3.10 through Figure 3.15 are also dependent on operation of the Turbine Bypass Valve system. In this case, limits support FHOOS operation during startup. These limits have no dependency on RPTOOS, PLUOOS, or SLO.

* All equipment service conditions assume 1 SRVOOS.



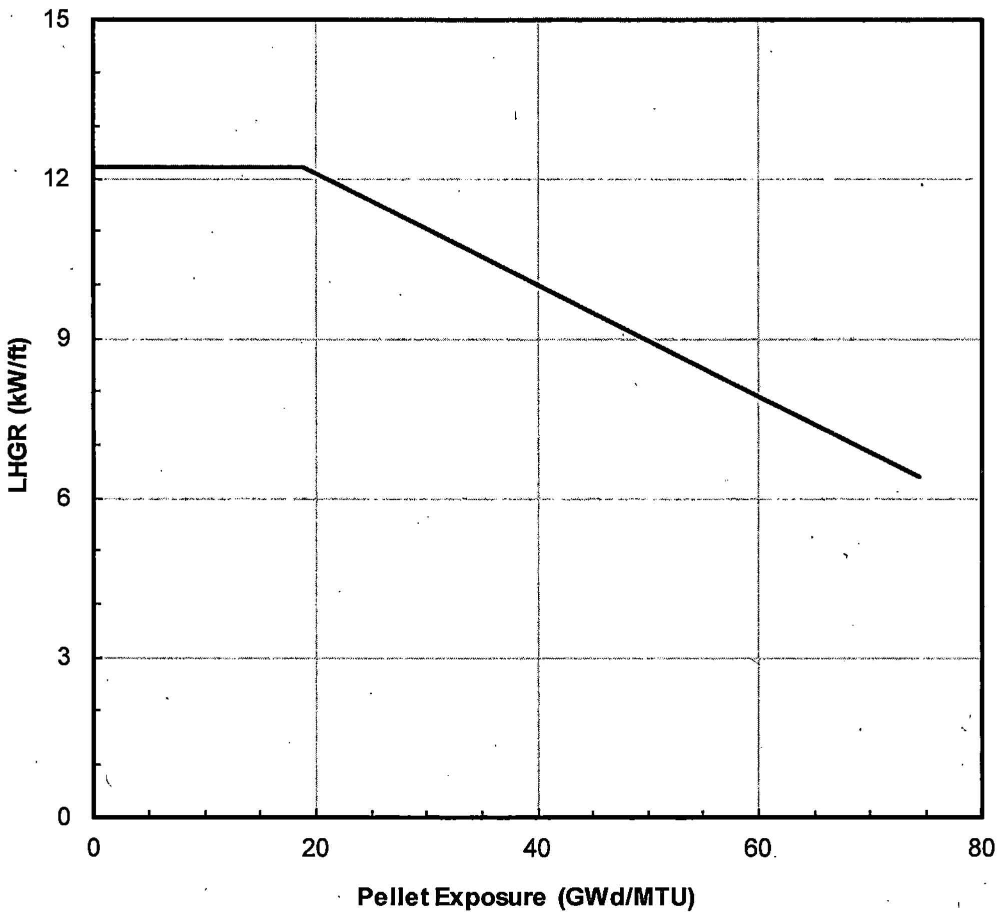
Pellet Exposure (GWd/MTU)	LHGR Limit (kW/ft)
0.0	13.4
18.9	13.4
74.4	7.1

Figure 3.1 $LHGR_{RATED}$ for ATRIUM-10 Fuel



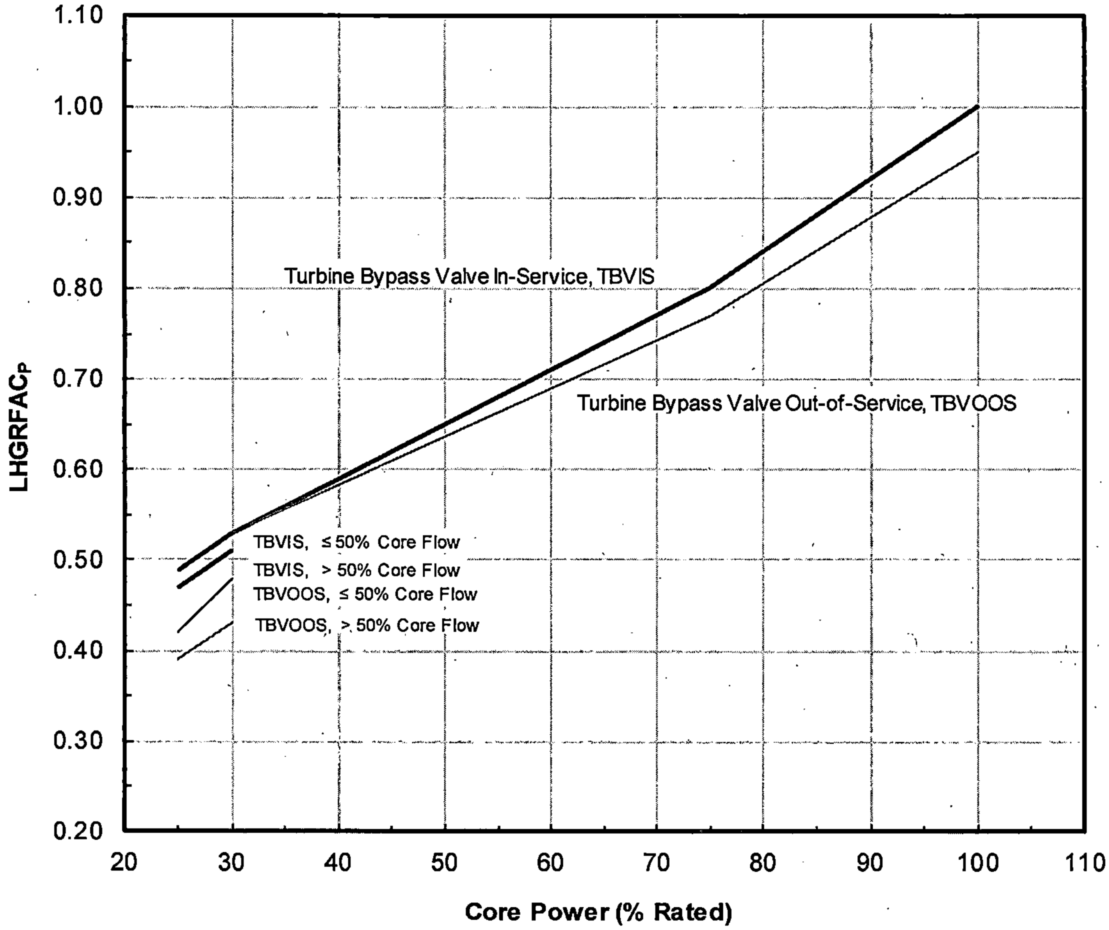
Pellet Exposure (GWd/MTU)	LHGR Limit (kW/ft)
0.0	14.1
18.9	14.1
74.4	7.4

Figure 3.2 $LHGR_{RATED}$ for ATRIUM-10XM Fuel



Pellet Exposure (GWd/MTU)	LHGR Limit (kW/ft)
0.0	12.2
18.9	12.2
74.4	6.4

Figure 3.3 LHGR$_{RATED}$ for ATRIUM-11 Fuel

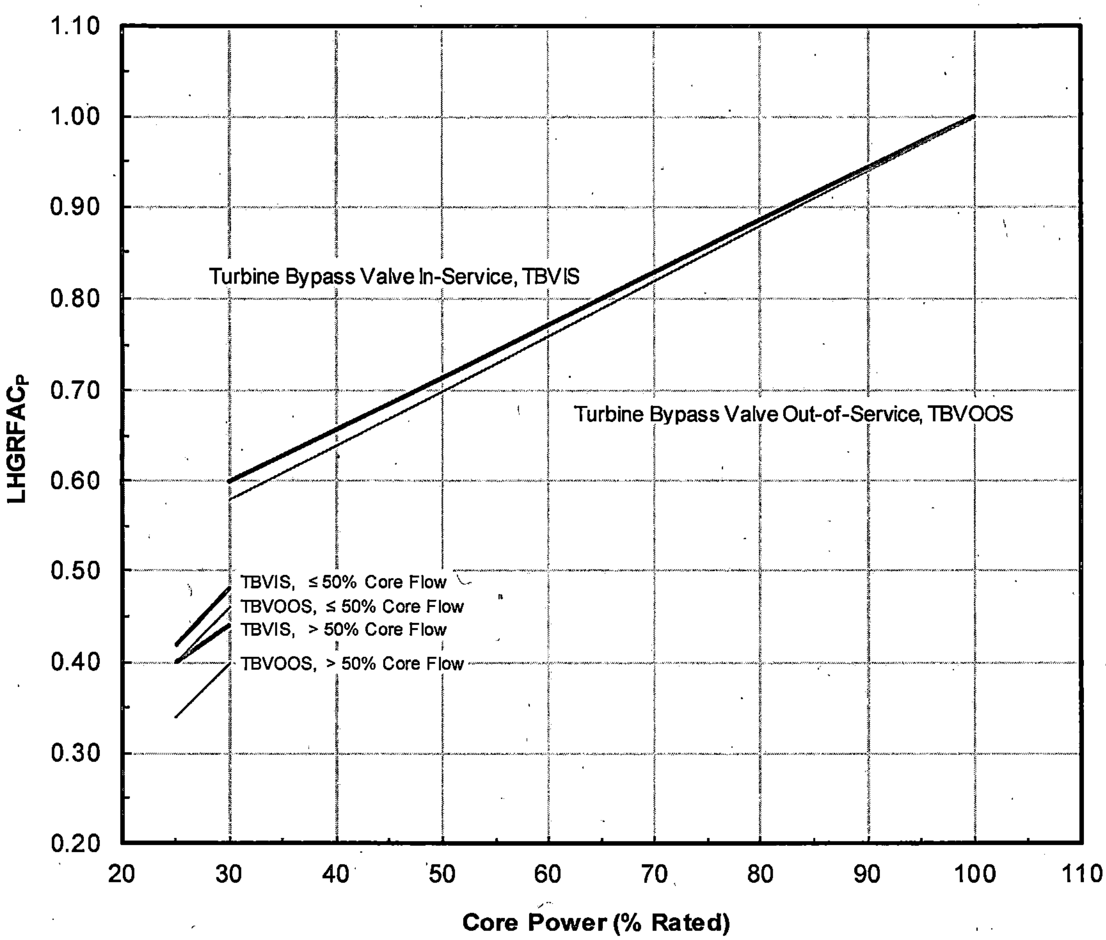


Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00
75.0	0.80
30.0	0.53
Core Flow > 50% Rated	
30.0	0.51
25.0	0.47
Core Flow ≤ 50% Rated	
30.0	0.53
25.0	0.49

Turbine Bypass Out-of-Service Core Power (% Rated)	LHGRFAC$_P$
100.0	0.95
75.0	0.77
30.0	0.53
Core Flow > 50% Rated	
30.0	0.43
25.0	0.39
Core Flow ≤ 50% Rated	
30.0	0.48
25.0	0.42

Figure 3.4 Base Operation LHGRFAC$_P$ for ATRIUM-10 Fuel

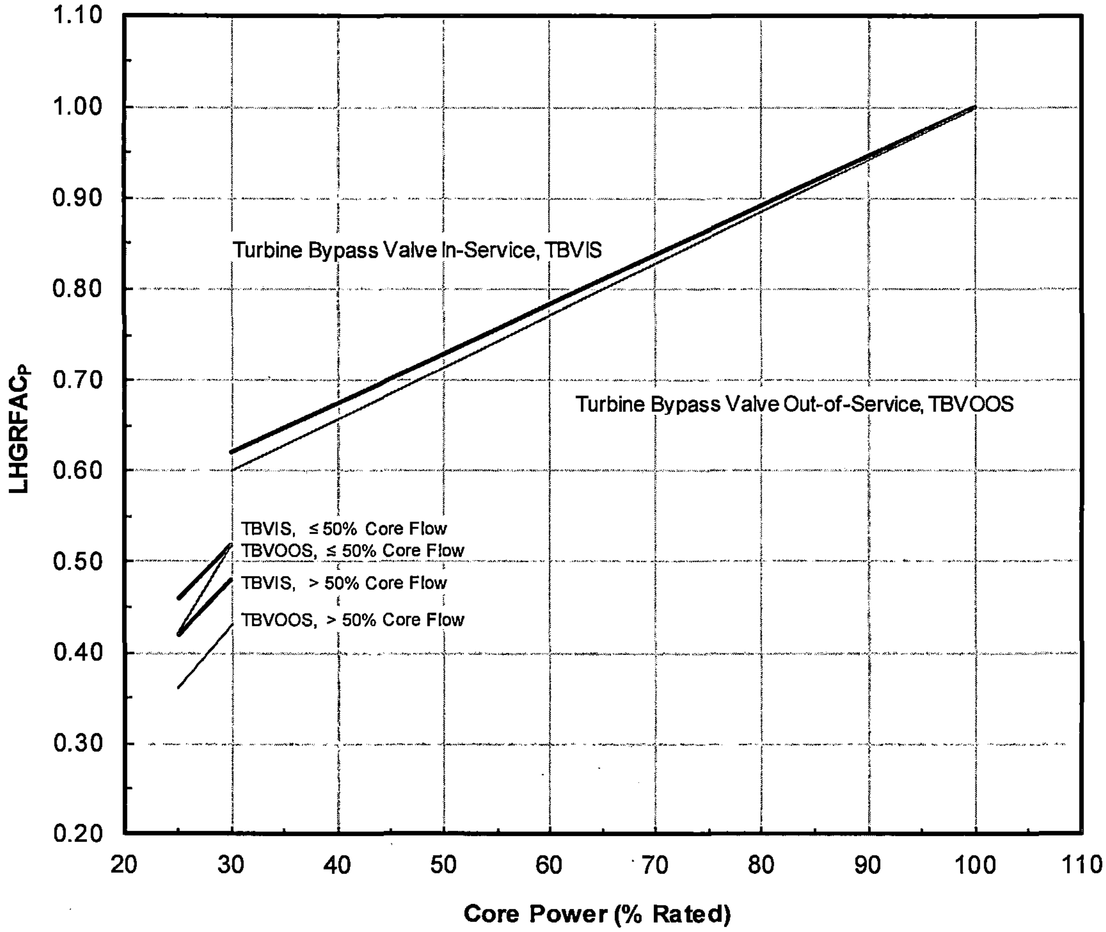
(*Independent of other EOOS conditions*)



Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$	*Turbine Bypass Out-of-Service* Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00	100.0	1.00
30.0	0.60	30.0	0.58
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.44	30.0	0.40
25.0	0.40	25.0	0.34
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.48	30.0	0.46
25.0	0.42	25.0	0.40

Figure 3.5 Base Operation LHGRFAC$_P$ for ATRIUM-10XM Fuel

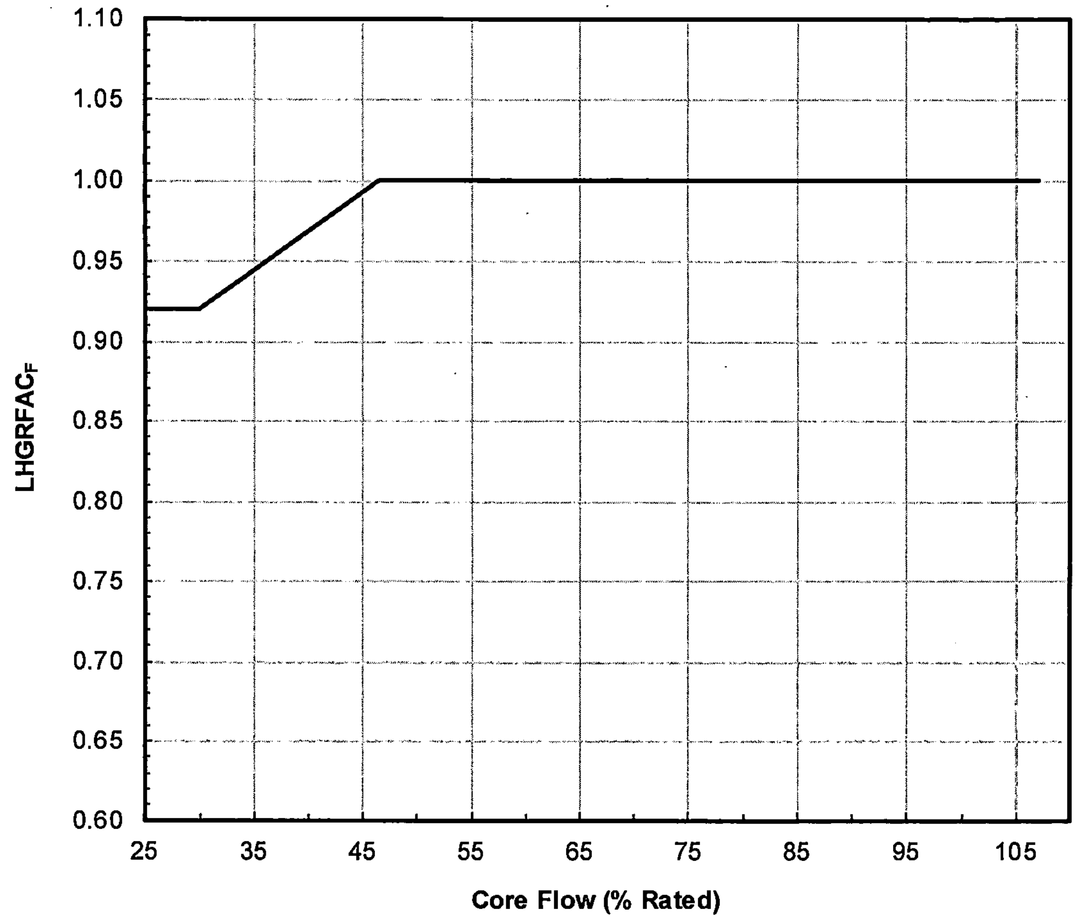
(*Independent of other EOOS conditions*)



Turbine Bypass In-Service Core Power	LHGRFAC$_P$	*Turbine Bypass Out-of-Service* Core Power	LHGRFAC$_P$
(% Rated)		(% Rated)	
100.0	1.00	100.0	1.00
30.0	0.62	30.0	0.60
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.48	30.0	0.43
25.0	0.42	25.0	0.36
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.52	30.0	0.52
25.0	0.46	25.0	0.42

Figure 3.6 Base Operation LHGRFAC$_P$ for ATRIUM-11 Fuel

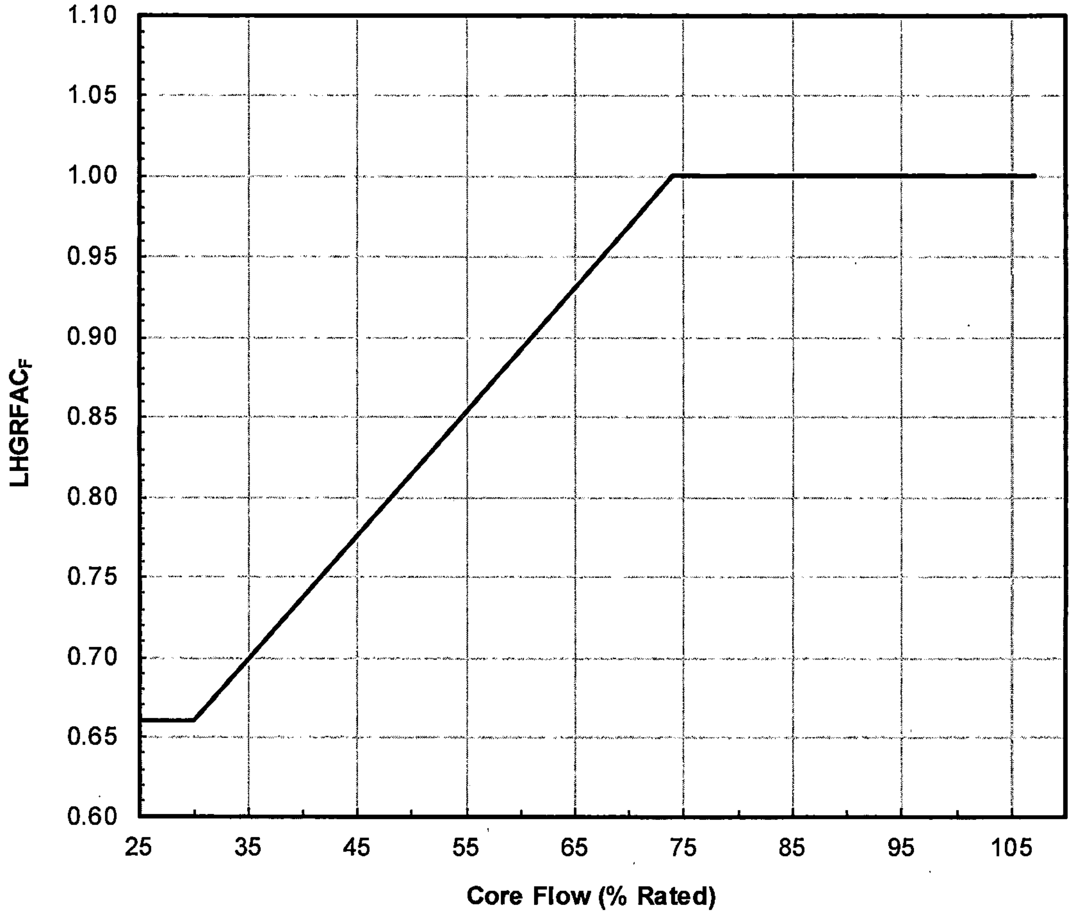
(Independent of other EOOS conditions)



Core Flow (% Rated)	LHGRFAC$_F$
0.0	0.92
30.0	0.92
46.4	1.00
107.0	1.00

Figure 3.7 LHGRFAC$_F$ for ATRIUM-10 Fuel
(*Values bound all EOOS conditions*)

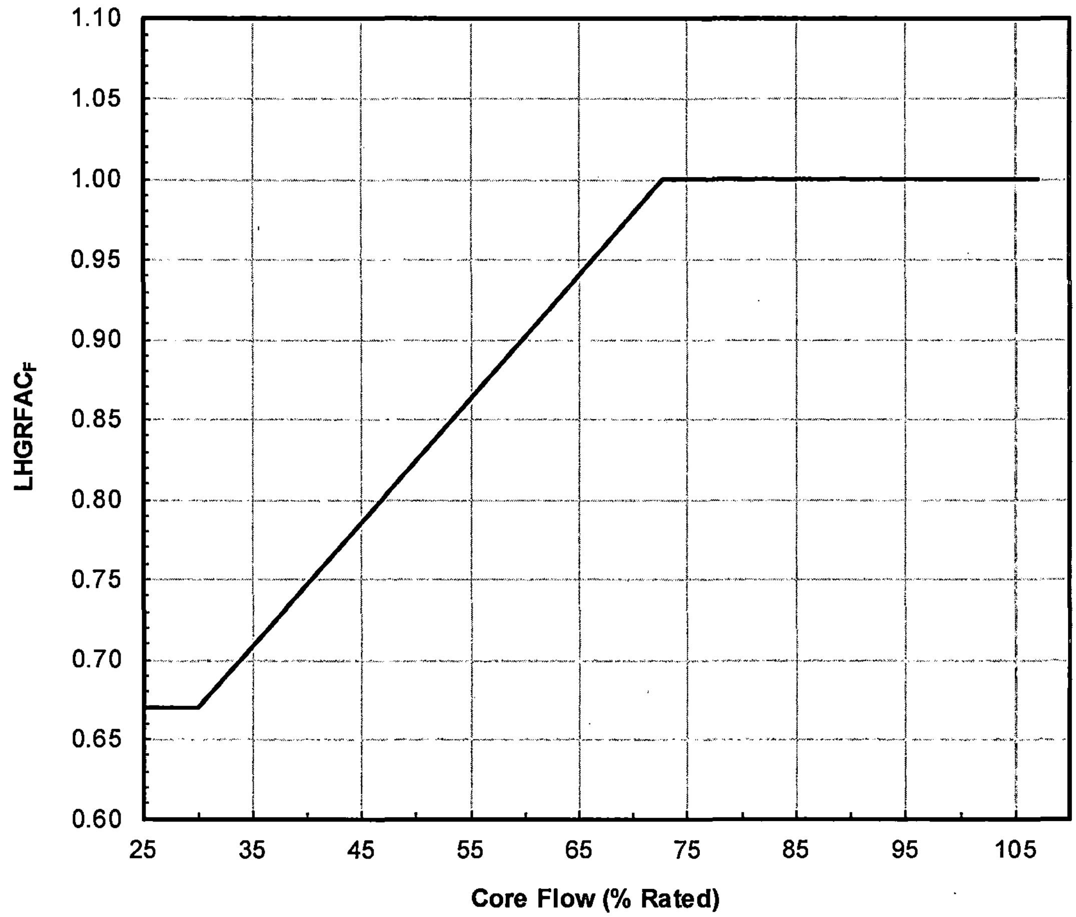
(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)



Core Flow (% Rated)	LHGRFAC$_F$
0.0	0.66
30.0	0.66
73.9	1
107.0	1

Figure 3.8 LHGRFAC$_F$ for ATRIUM-10XM Fuel
(*Values bound all EOOS conditions*)

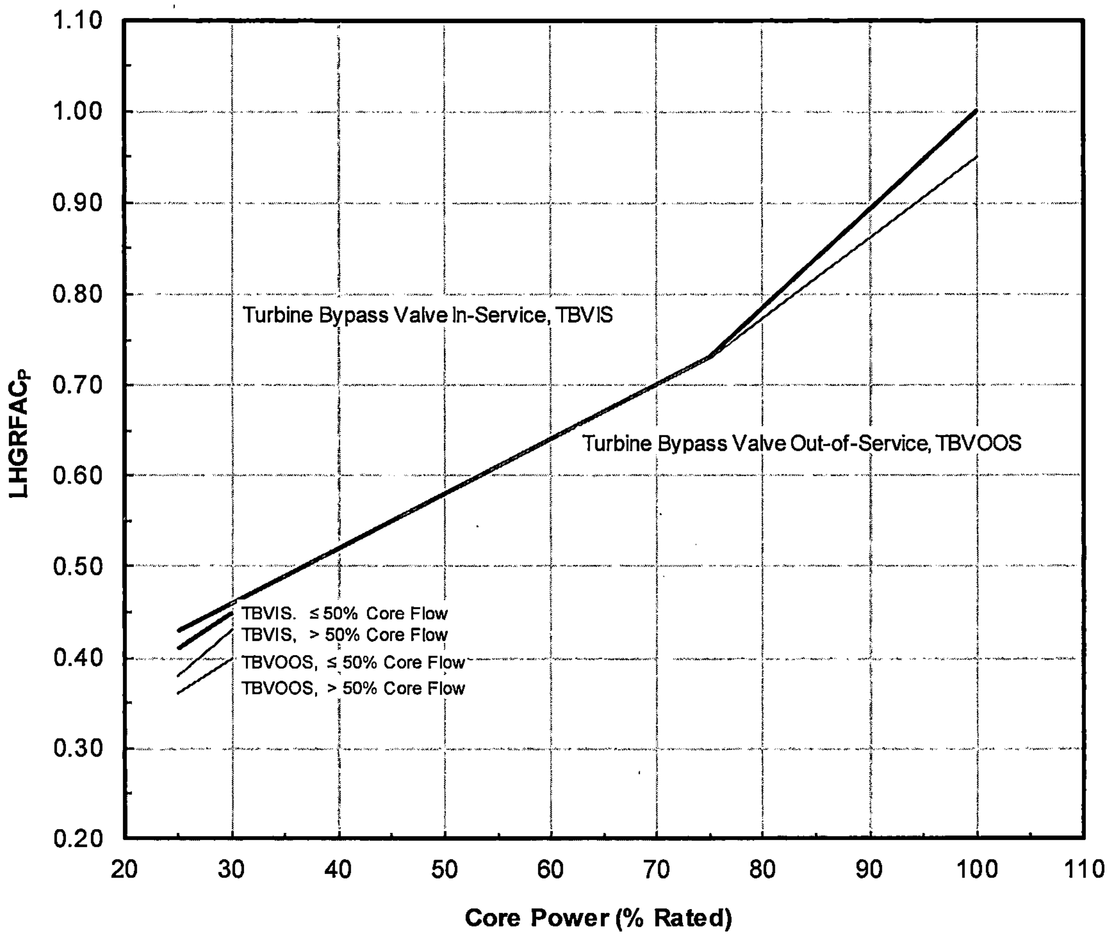
(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)



Core Flow (% Rated)	LHGRFAC$_F$
0.0	0.67
30.0	0.67
72.7	1
107.0	1

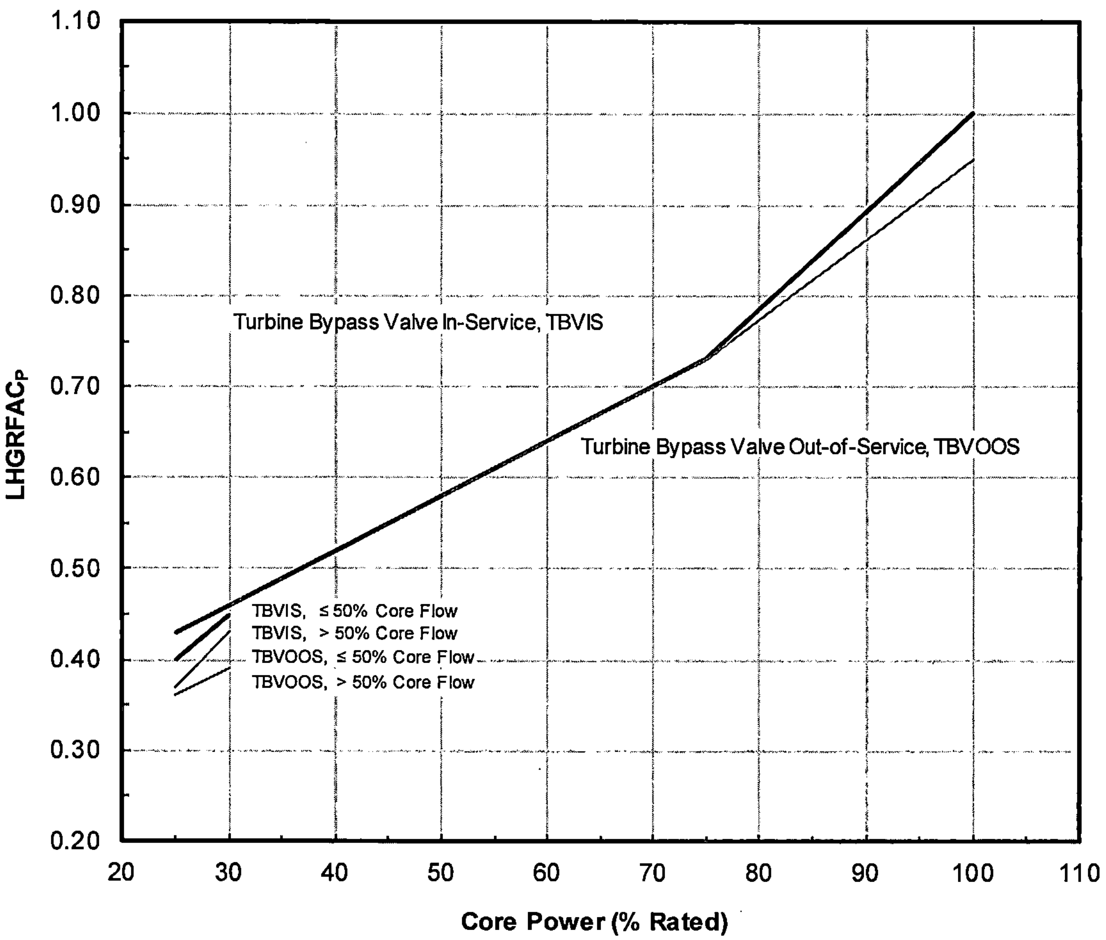
Figure 3.9 LHGRFAC$_F$ for ATRIUM-11 Fuel
(*Values bound all EOOS conditions*)

(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)



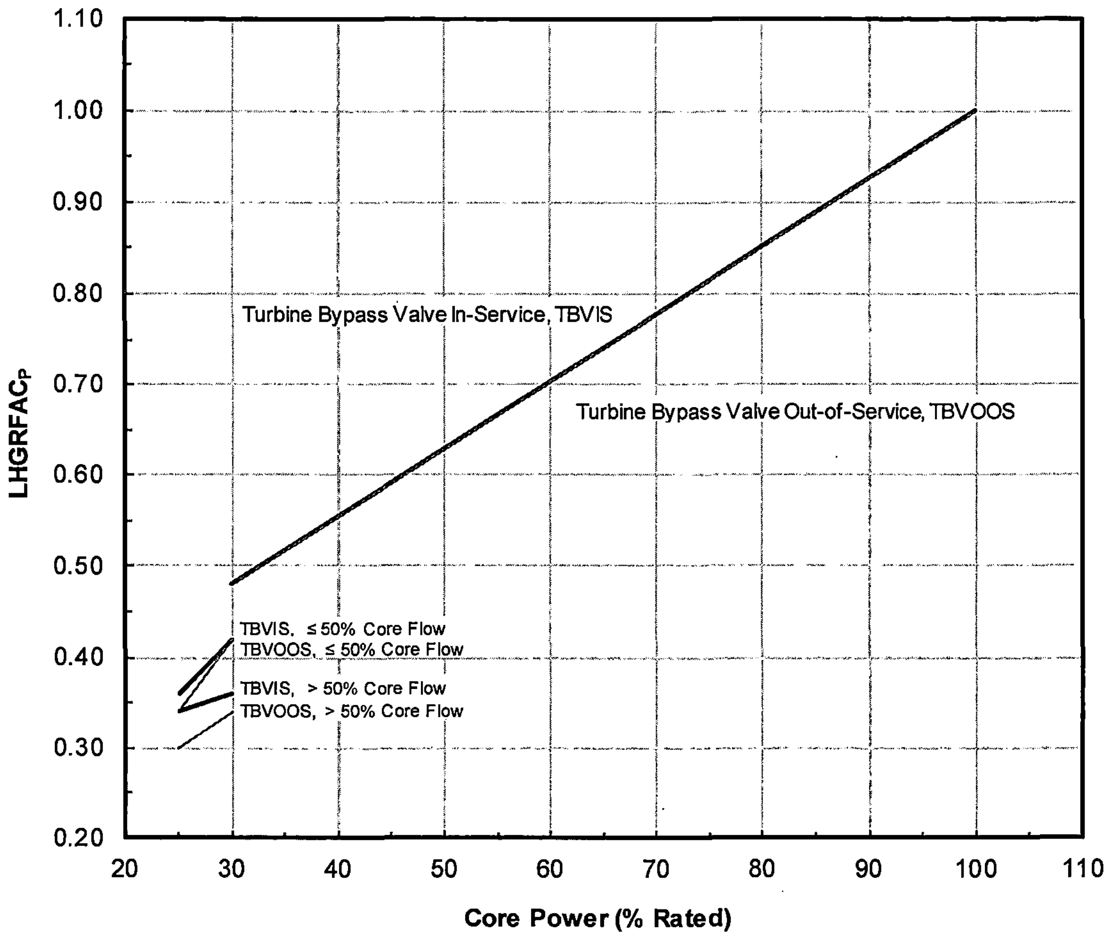
Turbine Bypass In-Service		*Turbine Bypass Out-of-Service*	
Core Power (% Rated)	**LHGRFAC$_P$**	**Core Power (% Rated)**	**LHGRFAC$_P$**
100.0	1.00	100.0	0.95
75.0	0.73	75.0	0.73
30.0	0.46	30.0	0.46
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.45	30.0	0.40
25.0	0.41	25.0	0.36
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.46	30.0	0.43
25.0	0.43	25.0	0.38

Figure 3.10 Startup Operation LHGRFAC$_P$ for ATRIUM-10 Fuel:
Table 3.1 Temperature Range 1
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



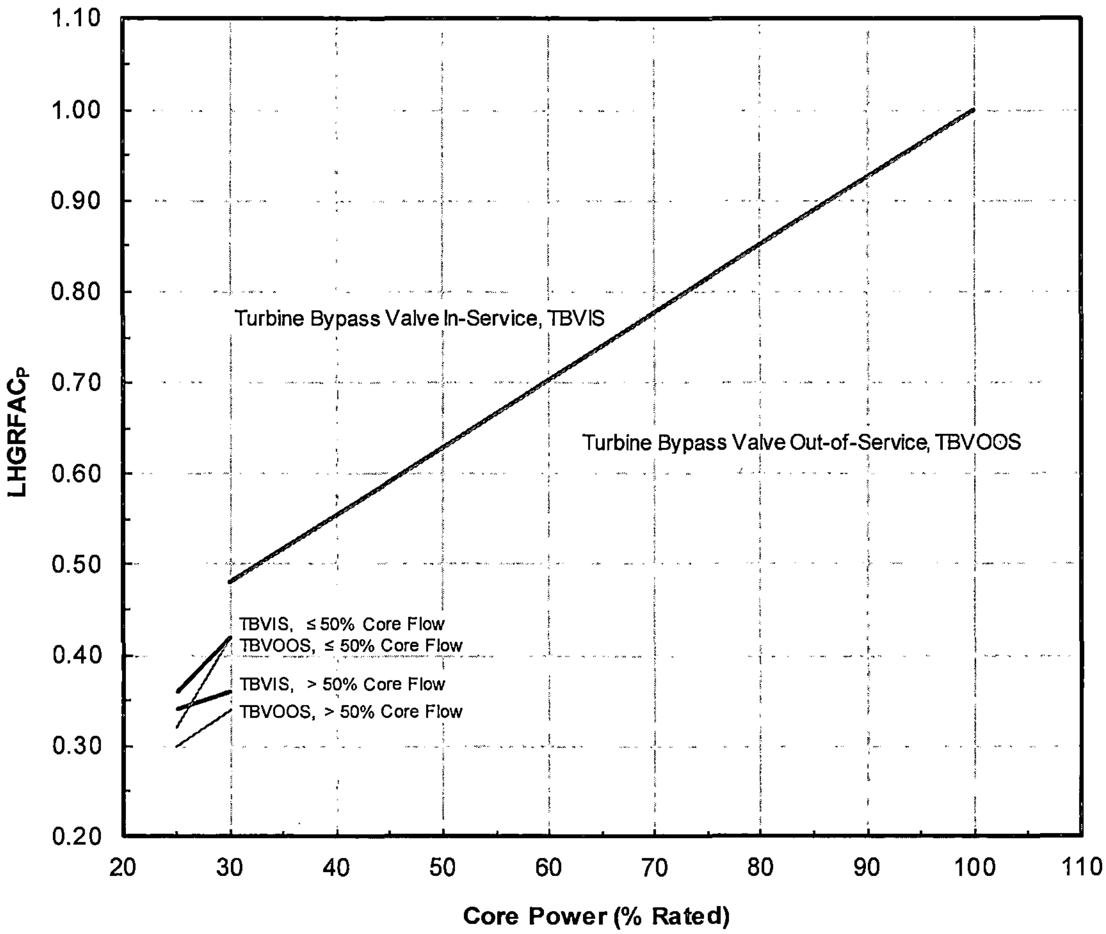
Turbine Bypass In-Service		*Turbine Bypass Out-of-Service*	
Core Power (% Rated)	**$LHGRFAC_P$**	**Core Power (% Rated)**	**$LHGRFAC_P$**
100.0	1.00	100.0	0.95
75.0	0.73	75.0	0.73
30.0	0.46	30.0	0.46
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.45	30.0	0.39
25.0	0.40	25.0	0.36
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.46	30.0	0.43
25.0	0.43	25.0	0.37

Figure 3.11 Startup Operation $LHGRFAC_P$ for ATRIUM-10 Fuel: Table 3.1 Temperature Range 2
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



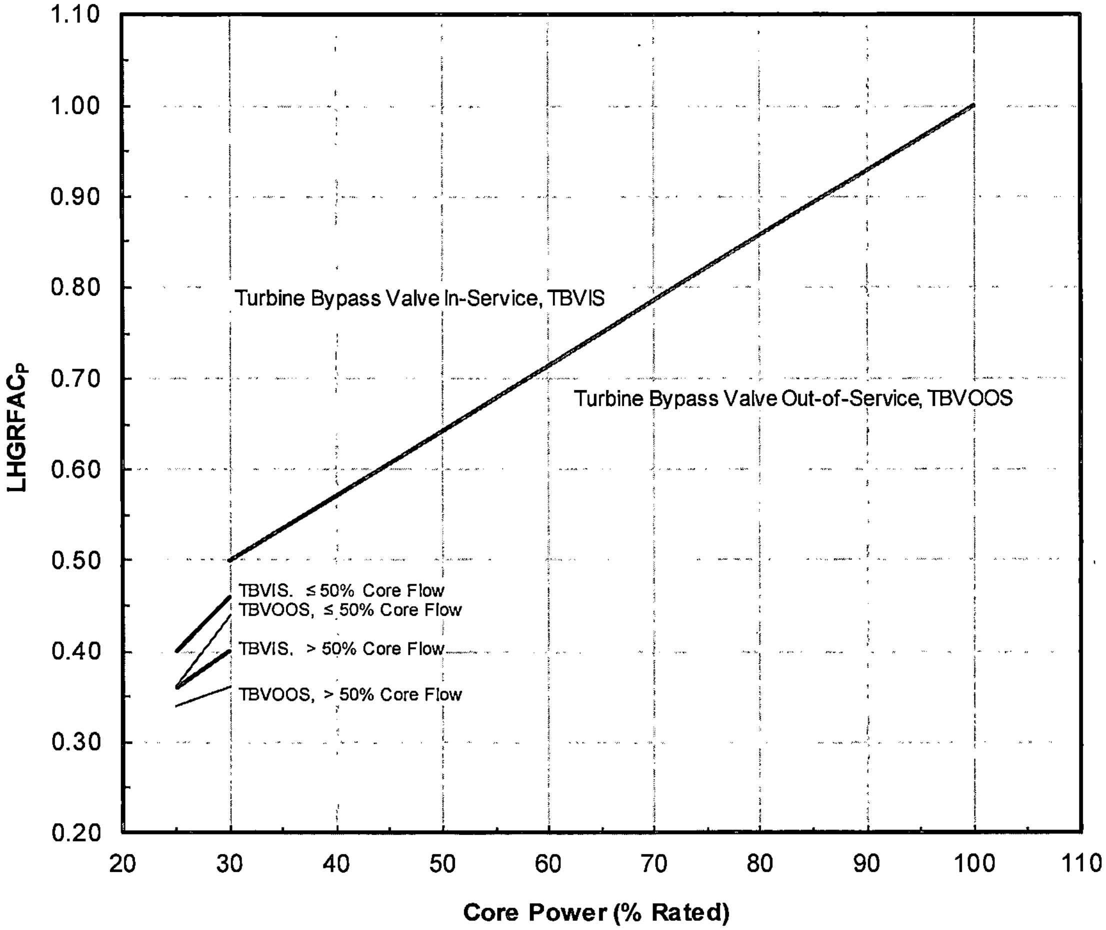
Turbine Bypass In-Service Core Power	$LHGRFAC_P$	*Turbine Bypass Out-of-Service* Core Power	$LHGRFAC_P$
(% Rated)		**(% Rated)**	
100.0	1.00	100.0	1.00
30.0	0.48	30.0	0.48
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.36	30.0	0.34
25.0	0.34	25.0	0.30
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.42	30.0	0.42
25.0	0.36	25.0	0.34

Figure 3.12 Startup Operation $LHGRFAC_P$ for ATRIUM-10XM
Fuel: Table 3.1 Temperature Range 1
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



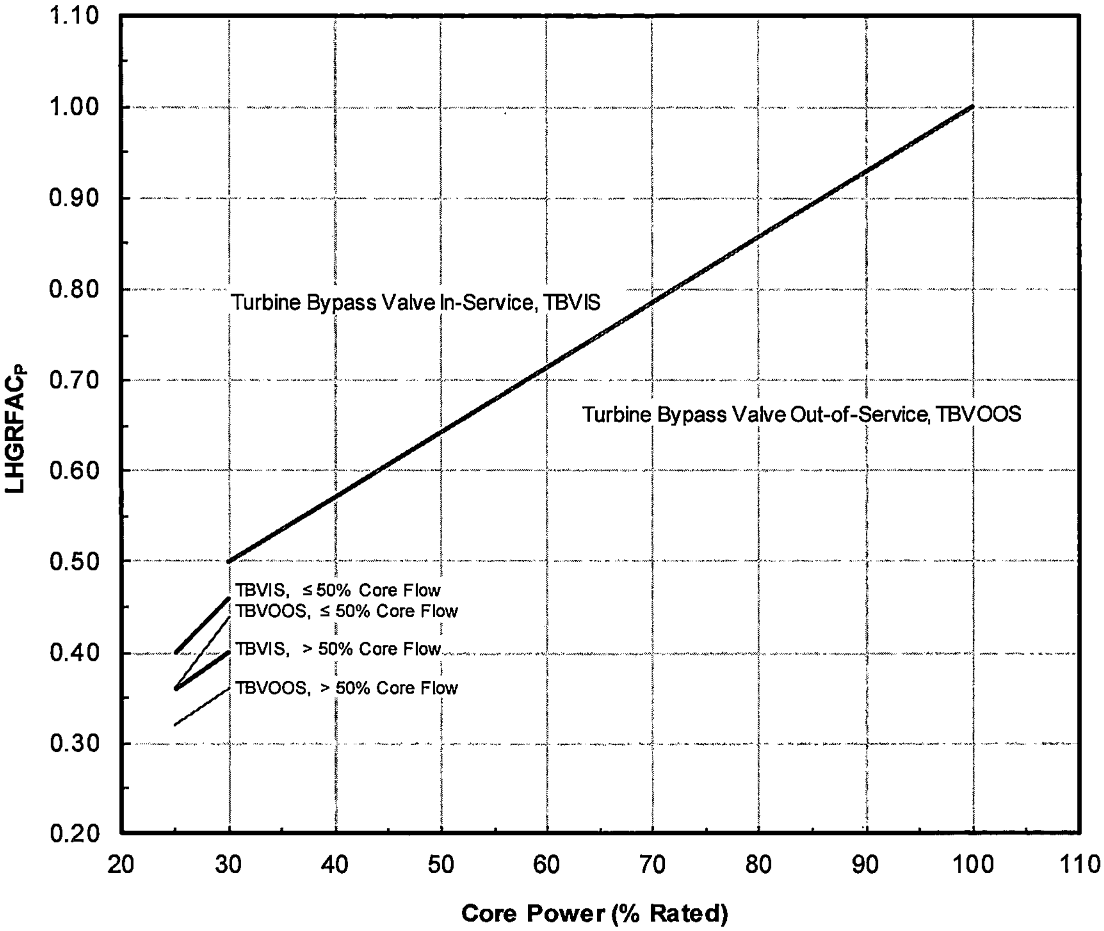
Turbine Bypass In-Service		*Turbine Bypass Out-of-Service*	
Core Power (% Rated)	**LHGRFAC$_P$**	**Core Power (% Rated)**	**LHGRFAC$_P$**
100.0	1.00	100.0	1.00
30.0	0.48	30.0	0.48
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.36	30.0	0.34
25.0	0.34	25.0	0.30
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.42	30.0	0.42
25.0	0.36	25.0	0.32

Figure 3.13 Startup Operation LHGRFAC$_P$ for ATRIUM-10XM
Fuel: Table 3.1 Temperature Range 2
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



Turbine Bypass In-Service Core Power	LHGRFAC$_P$	*Turbine Bypass Out-of-Service* Core Power	LHGRFAC$_P$
(% Rated)		(% Rated)	
100.0	1.00	100.0	1.00
30.0	0.50	30.0	0.50
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.40	30.0	0.36
25.0	0.36	25.0	0.34
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.46	30.0	0.44
25.0	0.40	25.0	0.36

Figure 3.14 Startup Operation LHGRFAC$_P$ for ATRIUM-11 Fuel:
Table 3.1 Temperature Range 1
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



Turbine Bypass In-Service	
Core Power (% Rated)	**LHGRFAC$_P$**
100.0	1.00
30.0	0.50
Core Flow > 50% Rated	
30.0	0.40
25.0	0.36
Core Flow ≤ 50% Rated	
30.0	0.46
25.0	0.40

Turbine Bypass Out-of-Service	
Core Power (% Rated)	**LHGRFAC$_P$**
100.0	1.00
30.0	0.50
Core Flow > 50% Rated	
30.0	0.36
25.0	0.32
Core Flow ≤ 50% Rated	
30.0	0.44
25.0	0.36

Figure 3.15 Startup Operation LHGRFAC$_P$ for ATRIUM-11 Fuel:
Table 3.1 Temperature Range 2
(no Feedwater heating during startup)
(Limits valid at and below 50% power)

4 OLMCPR Limits

(Technical Specification 3.2.2, 3.3.4.1, & 3.7.5)

OLMCPR is calculated to be the most limiting of the flow or power dependent values

$$\text{OLMCPR limit} = \text{MAX}(\text{MCPR}_F, \text{MCPR}_P)$$

where:

$MCPR_F$	core flow-dependent MCPR limit
$MCPR_P$	power-dependent MCPR limit

4.1 Flow Dependent MCPR Limit: $MCPR_F$

$MCPR_F$ limits are dependent upon core flow (% of Rated), and the max core flow limit, (Rated or Increased Core Flow, ICF). $MCPR_F$ limits are shown in Figure 4.1, per Reference 1. Limits are valid for all EOOS combinations. No adjustment is required for SLO conditions.

4.2 Power Dependent MCPR Limit: $MCPR_P$

$MCPR_P$ limits are dependent upon:

- Core Power Level (% of Rated)
- Technical Specification Scram Speed (TSSS), Nominal Scram Speed (NSS), or Optimum Scram Speed (OSS)
- Cycle Operating Exposure (NEOC, EOC, and CD - as defined in this section)
- Equipment Out-Of-Service Options
- Two or Single recirculation Loop Operation (TLO vs. SLO)

The $MCPR_P$ limits are provided in Table 4.2 through Table 4.4, where each table contains the limits for all fuel types and EOOS options (for a specified scram speed and exposure range). The CMSS determines $MCPR_P$ limits, from these tables, based on linear interpolation between the specified powers.

4.2.1 Startup without Feedwater Heaters

There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. Additional power dependent limits are shown in Table 4.5 through Table 4.8 based on temperature conditions identified in Table 3.1.

4.2.2 Scram Speed Dependent Limits (TSSS vs. NSS vs. OSS)

$MCPR_P$ limits are provided for three different sets of assumed scram speeds. The Technical Specification Scram Speed (TSSS) $MCPR_P$ limits are applicable at all times, as long as the scram time surveillance demonstrates the times in Technical Specification Table 3.1.4-1 are met. Both Nominal Scram Speeds (NSS) and/or Optimum Scram Speeds (OSS) may be used, as long as the scram time surveillance demonstrates Table 4.1 times are applicable.*†

Table 4.1 Nominal Scram Time Basis

Notch Position (index)	Nominal Scram Timing (seconds)	Optimum Scram Timing (seconds)
46	0.420	0.380
36	0.980	0.875
26	1.600	1.465
6	2.900	2.900

In demonstrating compliance with the NSS and/or OSS scram time basis, surveillance requirements from Technical Specification 3.1.4 apply; accepting the definition of SLOW rods should conform to scram speeds shown in Table 4.1. If conformance is not demonstrated, TSSS based $MCPR_P$ limits are applied.

On initial cycle startup, TSSS limits are used until the successful completion of scram timing confirms NSS and/or OSS based limits are applicable.

4.2.3 Exposure Dependent Limits

Exposures are tracked on a Core Average Exposure basis (CAVEX, not Cycle Exposure). Higher exposure $MCPR_P$ limits are always more limiting and may be used for any Core Average Exposure up to the ending exposure. Per Reference 1, $MCPR_P$ limits are provided for the following exposure ranges:

BOC to NEOC	NEOC corresponds to	**30,910.3 MWd / MTU**
BOC to EOCLB	EOCLB corresponds to	**33,210.3 MWd / MTU**
BOC to End of Coast	End of Coast	**34,624.4 MWd / MTU**

NEOC refers to a Near EOC exposure point.

* Reference 1 analysis results are based on information identified in Reference 4.

† Drop out times consistent with method used to perform actual timing measurements (i.e., including pickup/dropout effects).

The EOCLB exposure point is not the true End-Of-Cycle exposure. Instead it corresponds to a licensing exposure window exceeding expected end-of-full-power-life.

The End of Coast exposure point represents a licensing exposure point exceeding the expected end-of-cycle exposure including cycle extension options.

4.2.4 Equipment Out-Of-Service (EOOS) Options

EOOS options* covered by $MCPR_P$ limits are given by the following:

In-Service	All equipment In-Service
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
RPTOOS+TBVOOS	Combined RPTOOS and TBVOOS
PLUOOS	Power Load Unbalance Out-Of-Service
PLUOOS+RPTOOS	Combined PLUOOS and RPTOOS
PLUOOS+TBVOOS	Combined PLUOOS and TBVOOS
PLUOOS+TBVOOS+RPTOOS	Combined PLUOOS, RPTOOS, and TBVOOS
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service (or Final Feedwater Temperature Reduction)
RCPOOS	One Recirculation Pump Out-Of-Service

For exposure ranges up to NEOC and EOCLB, additional combinations of $MCPR_P$ limits are also provided including FHOOS. The coast down exposure range assumes application of FFWTR. FHOOS based $MCPR_P$ limits for the coast down exposure are redundant because the temperature setdown assumption is identical with FFWTR.

4.2.5 Single-Loop-Operation (SLO) Limits

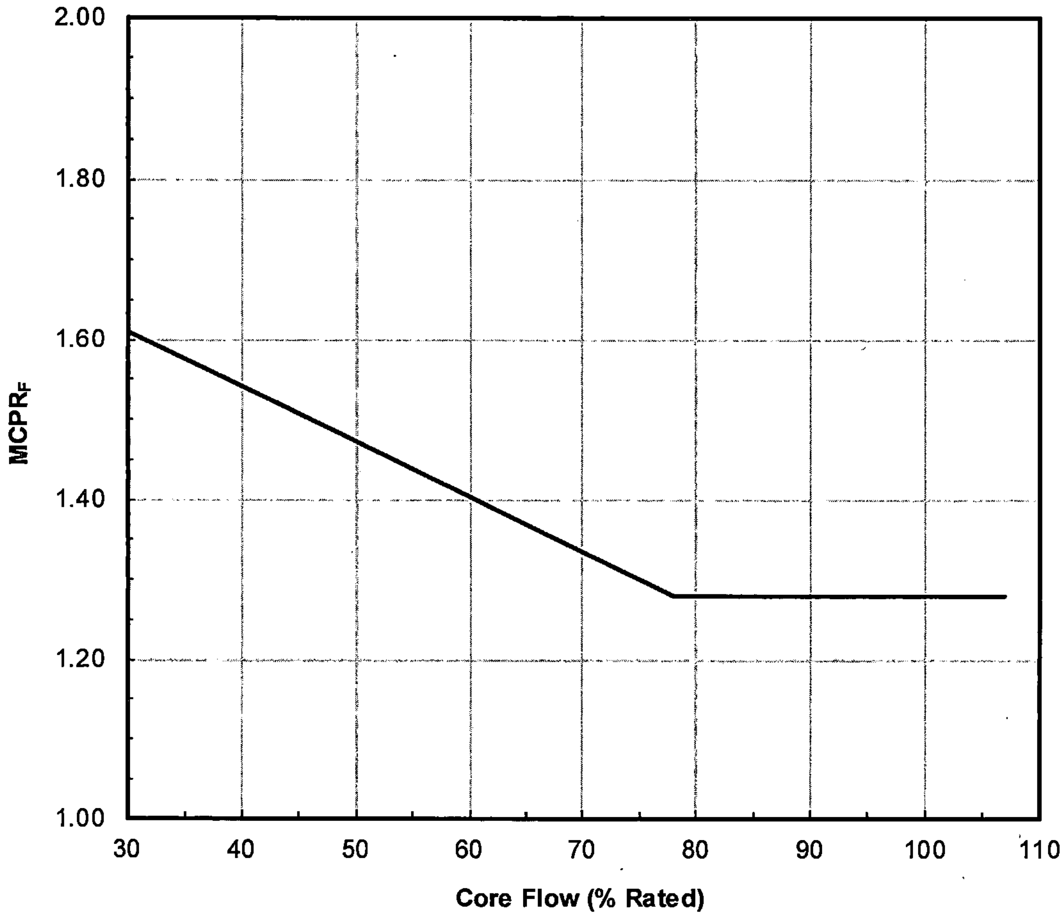
When operating in RCPOOS conditions, MCPRp limits are constructed differently from the normal operating RCP conditions. The limiting event for RCPOOS is a pump seizure scenario, which sets the upper bound for allowed core power and flow†. This event is not impacted by scram time assumptions. Specific $MCPR_P$ limits are shown in Table 4.9.

4.2.6 Below Pbypass Limits

Below Pbypass (30% rated power), $MCPR_P$ limits depend upon core flow. One set of $MCPR_P$ limits applies for core flow above 50% of rated; a second set applies if the core flow is less than or equal to 50% rated.

* All equipment service conditions assume 1 SRVOOS.

† RCPOOS limits are only valid up to 50% rated core power, 50% rated core flow, and an active recirculation drive flow of 17.73 Mlb_m/hr.



Core Flow (% Rated)	$MCPR_F$
30.0	1.61
78.0	1.28
107.0	1.28

Figure 4.1 $MCPR_F$ for All Fuel Types
(*Values bound all EOOS conditions*)

(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)

Table 4.2 $MCPR_P$ Limits for All Fuel Types: Optimum Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10			ATRIUM-10XM			ATRIUM-11		
		BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
Base Case	100	1.43	1.46	1.48	1.40	1.41	1.43	1.38	1.40	1.42
	75	1.59	1.60	1.64	1.53	1.53	1.56	1.51	1.52	1.55
	65	1.68	1.69	1.75	1.61	1.61	1.66	1.60	1.61	1.69
	50	1.85	1.86	---	1.76	1.76	---	1.83	1.84	---
	50	1.93	1.94	1.96	1.81	1.81	1.84	1.85	1.86	1.95
	40	2.14	2.15	2.26	1.99	1.99	2.09	2.10	2.11	2.24
	30	2.51	2.52	2.67	2.28	2.28	2.42	2.44	2.45	2.62
	30 at > 50%F	2.65	2.66	2.79	2.59	2.59	2.71	2.62	2.63	2.74
	25 at > 50%F	2.93	2.94	3.07	2.87	2.87	3.01	2.88	2.89	3.04
	30 at ≤ 50%F	2.58	2.59	2.69	2.50	2.50	2.60	2.57	2.58	2.69
	25 at ≤ 50%F	2.77	2.78	2.92	2.73	2.73	2.87	2.81	2.82	2.98
FHOOS	100	1.47	1.48	---	1.43	1.43	---	1.41	1.42	---
	75	1.62	1.64	---	1.56	1.56	---	1.54	1.55	---
	65	1.74	1.75	---	1.66	1.66	---	1.68	1.69	---
	50	---	---	---	---	---	---	---	---	---
	50	1.95	1.96	---	1.84	1.84	---	1.94	1.95	---
	40	2.25	2.26	---	2.09	2.09	---	2.23	2.24	---
	30	2.66	2.67	---	2.42	2.42	---	2.61	2.62	---
	30 at > 50%F	2.78	2.79	---	2.71	2.71	---	2.73	2.74	---
	25 at > 50%F	3.06	3.07	---	3.01	3.01	---	3.03	3.04	---
	30 at ≤ 50%F	2.68	2.69	---	2.60	2.60	---	2.68	2.69	---
	25 at ≤ 50%F	2.91	2.92	---	2.87	2.87	---	2.97	2.98	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR/FHOOS is supported for the BOC to End of Coast limits.

Table 4.3 MCPR$_P$ Limits for All Fuel Types: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10			ATRIUM-10XM			ATRIUM-11		
		BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
	100	1.44	1.47	1.50	1.41	1.42	1.45	1.40	1.42	1.44
	75	1.60	1.61	1.64	1.54	1.55	1.57	1.53	1.54	1.59
	65	1.69	1.70	1.76	1.62	1.62	1.67	1.64	1.65	1.73
	50	1.90	1.91	---	1.79	1.79	---	---	---	---
	50	1.94	1.95	2.01	1.81	1.82	1.87	1.87	1.88	2.00
Base Case	40	2.19	2.20	2.32	2.03	2.03	2.12	2.15	2.16	2.29
	30	2.57	2.58	2.74	2.33	2.33	2.47	2.50	2.51	2.68
	30 at > 50%F	2.65	2.66	2.79	2.59	2.59	2.71	2.62	2.63	2.74
	25 at > 50%F	2.93	2.94	3.07	2.87	2.87	3.01	2.88	2.89	3.04
	30 at ≤ 50%F	2.58	2.59	2.74	2.50	2.50	2.60	2.57	2.58	2.69
	25 at ≤ 50%F	2.77	2.78	2.92	2.73	2.73	2.87	2.81	2.82	2.98
	100	1.50	1.52	1.54	1.45	1.46	1.48	1.44	1.46	1.47
	75	1.65	1.66	1.69	1.58	1.58	1.60	1.57	1.58	1.60
	65	1.75	1.76	1.81	1.66	1.66	1.70	1.65	1.66	1.74
	50	1.91	1.92	---	1.80	1.80	---	---	---	---
	50	1.94	1.95	2.01	1.81	1.82	1.88	1.88	1.89	2.00
TBVOOS	40	2.19	2.20	2.32	2.03	2.03	2.12	2.15	2.16	2.29
	30	2.57	2.58	2.74	2.33	2.33	2.47	2.50	2.51	2.68
	30 at > 50%F	3.26	3.27	3.41	3.09	3.09	3.22	3.23	3.24	3.38
	25 at > 50%F	3.67	3.68	3.80	3.52	3.52	3.63	3.61	3.62	3.76
	30 at ≤ 50%F	3.02	3.03	3.18	2.85	2.85	2.99	3.04	3.05	3.20
	25 at ≤ 50%F	3.45	3.46	3.67	3.32	3.32	3.50	3.45	3.46	3.65
	100	1.48	1.50	---	1.44	1.45	---	1.43	1.44	---
	75	1.64	1.64	---	1.57	1.57	---	1.58	1.59	---
	65	1.75	1.76	---	1.67	1.67	---	1.72	1.73	---
	50	---	---	---	---	---	---	---	---	---
	50	2.00	2.01	---	1.87	1.87	---	1.99	2.00	---
FHOOS	40	2.31	2.32	---	2.12	2.12	---	2.28	2.29	---
	30	2.73	2.74	---	2.47	2.47	---	2.67	2.68	---
	30 at > 50%F	2.78	2.79	---	2.71	2.71	---	2.73	2.74	---
	25 at > 50%F	3.06	3.07	---	3.01	3.01	---	3.03	3.04	---
	30 at ≤ 50%F	2.73	2.74	---	2.60	2.60	---	2.68	2.69	---
	25 at ≤ 50%F	2.91	2.92	---	2.87	2.87	---	2.97	2.98	---
	100	1.44	1.47	1.50	1.41	1.42	1.45	1.40	1.42	1.44
	75	1.60	1.61	1.64	1.54	1.55	1.57	1.53	1.54	1.59
	65	1.85	1.87	1.87	1.73	1.74	1.75	1.76	1.78	1.79
	50	---	---	---	---	---	---	---	---	---
	50	1.94	1.95	2.01	1.81	1.82	1.87	1.87	1.88	2.00
PLUOOS	40	2.19	2.20	2.32	2.03	2.03	2.12	2.15	2.16	2.29
	30	2.57	2.58	2.74	2.33	2.33	2.47	2.50	2.51	2.68
	30 at > 50%F	2.65	2.66	2.79	2.59	2.59	2.71	2.62	2.63	2.74
	25 at > 50%F	2.93	2.94	3.07	2.87	2.87	3.01	2.88	2.89	3.04
	30 at ≤ 50%F	2.58	2.59	2.74	2.50	2.50	2.60	2.57	2.58	2.69
	25 at ≤ 50%F	2.77	2.78	2.92	2.73	2.73	2.87	2.81	2.82	2.98

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.3 MCPR$_P$ Limits for All Fuel Types: Nominal Scram Time Basis (*continued*)*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
TBVOOS FHOOS	100	1.53	1.54	---	1.47	1.48	---	1.46	1.47	---
	75	1.68	1.69	---	1.60	1.60	---	1.59	1.60	---
	65	1.80	1.81	---	1.70	1.70	---	1.73	1.74	---
	50	---	---	---	---	---	---	---	---	---
	50	2.00	2.01	---	1.88	1.88	---	1.99	2.00	---
	40	2.31	2.32	---	2.12	2.12	---	2.28	2.29	---
	30	2.73	2.74	---	2.47	2.47	---	2.67	2.68	---
	30 at > 50%F	3.40	3.41	---	3.22	3.22	---	3.37	3.38	---
	25 at > 50%F	3.79	3.80	---	3.63	3.63	---	3.75	3.76	---
	30 at ≤ 50%F	3.17	3.18	---	2.99	2.99	---	3.19	3.20	---
	25 at ≤ 50%F	3.66	3.67	---	3.50	3.50	---	3.64	3.65	---
TBVOOS PLUOOS	100	1.50	1.52	1.54	1.45	1.46	1.48	1.44	1.46	1.47
	75	1.65	1.66	1.69	1.58	1.58	1.60	1.57	1.58	1.60
	65	1.85	1.87	1.87	1.73	1.74	1.75	1.76	1.78	1.79
	50	---	---	---	---	---	---	---	---	---
	50	1.94	1.95	2.01	1.81	1.82	1.88	1.88	1.89	2.00
	40	2.19	2.20	2.32	2.03	2.03	2.12	2.15	2.16	2.29
	30	2.57	2.58	2.74	2.33	2.33	2.47	2.50	2.51	2.68
	30 at > 50%F	3.26	3.27	3.41	3.09	3.09	3.22	3.23	3.24	3.38
	25 at > 50%F	3.67	3.68	3.80	3.52	3.52	3.63	3.61	3.62	3.76
	30 at ≤ 50%F	3.02	3.03	3.18	2.85	2.85	2.99	3.04	3.05	3.20
	25 at ≤ 50%F	3.45	3.46	3.67	3.32	3.32	3.50	3.45	3.46	3.65
FHOOS PLUOOS	100	1.48	1.50	---	1.44	1.45	---	1.43	1.44	---
	75	1.64	1.64	---	1.57	1.57	---	1.58	1.59	---
	65	1.85	1.87	---	1.73	1.74	---	1.76	1.78	---
	50	---	---	---	---	---	---	---	---	---
	50	2.00	2.01	---	1.87	1.87	---	1.99	2.00	---
	40	2.31	2.32	---	2.12	2.12	---	2.28	2.29	---
	30	2.73	2.74	---	2.47	2.47	---	2.67	2.68	---
	30 at > 50%F	2.78	2.79	---	2.71	2.71	---	2.73	2.74	---
	25 at > 50%F	3.06	3.07	---	3.01	3.01	---	3.03	3.04	---
	30 at ≤ 50%F	2.73	2.74	---	2.60	2.60	---	2.68	2.69	---
	25 at ≤ 50%F	2.91	2.92	---	2.87	2.87	---	2.97	2.98	---
TBVOOS FHOOS PLUOOS	100	1.53	1.54	---	1.47	1.48	---	1.46	1.47	---
	75	1.68	1.69	---	1.60	1.60	---	1.59	1.60	---
	65	1.85	1.87	---	1.73	1.74	---	1.76	1.78	---
	50	---	---	---	---	---	---	---	---	---
	50	2.00	2.01	---	1.88	1.88	---	1.99	2.00	---
	40	2.31	2.32	---	2.12	2.12	---	2.28	2.29	---
	30	2.73	2.74	---	2.47	2.47	---	2.67	2.68	---
	30 at > 50%F	3.40	3.41	---	3.22	3.22	---	3.37	3.38	---
	25 at > 50%F	3.79	3.80	---	3.63	3.63	---	3.75	3.76	---
	30 at ≤ 50%F	3.17	3.18	---	2.99	2.99	---	3.19	3.20	---
	25 at ≤ 50%F	3.66	3.67	---	3.50	3.50	---	3.64	3.65	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.4 $MCPR_P$ Limits for All Fuel Types: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10			ATRIUM-10XM			ATRIUM-11		
		BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
Base Case	100	1.47	1.48	1.51	1.43	1.43	1.46	1.42	1.43	1.45
	75	1.61	1.62	1.65	1.55	1.55	1.58	1.55	1.56	1.63
	65	1.70	1.71	1.79	1.63	1.63	1.68	1.69	1.70	1.77
	50	---	---	---	---	1.82	---	---	---	---
	50	1.95	1.96	2.07	1.82	1.83	1.90	1.92	1.93	2.05
	40	2.25	2.26	2.37	2.06	2.06	2.16	2.20	2.21	2.34
	30	2.64	2.65	2.81	2.38	2.38	2.52	2.55	2.56	2.74
	30 at > 50%F	2.65	2.66	2.81	2.59	2.59	2.71	2.62	2.63	2.74
	25 at > 50%F	2.93	2.94	3.07	2.87	2.87	3.01	2.88	2.89	3.04
	30 at ≤ 50%F	2.64	2.65	2.81	2.50	2.50	2.60	2.57	2.58	2.74
	25 at ≤ 50%F	2.77	2.78	2.92	2.73	2.73	2.87	2.81	2.82	2.98
TBVOOS	100	1.52	1.54	1.56	1.47	1.47	1.48	1.46	1.47	1.49
	75	1.66	1.67	1.70	1.59	1.59	1.61	1.58	1.59	1.64
	65	1.76	1.77	1.82	1.67	1.67	1.71	1.70	1.71	1.78
	50	---	---	---	---	---	---	---	---	---
	50	1.96	1.97	2.07	1.83	1.83	1.91	1.93	1.94	2.06
	40	2.26	2.27	2.38	2.07	2.07	2.16	2.20	2.21	2.34
	30	2.64	2.65	2.81	2.38	2.38	2.52	2.55	2.56	2.74
	30 at > 50%F	3.26	3.27	3.41	3.09	3.09	3.22	3.23	3.24	3.38
	25 at > 50%F	3.67	3.68	3.80	3.52	3.52	3.63	3.61	3.62	3.76
	30 at ≤ 50%F	3.02	3.03	3.18	2.85	2.85	2.99	3.04	3.05	3.20
	25 at ≤ 50%F	3.45	3.46	3.67	3.32	3.32	3.50	3.45	3.46	3.65
FHOOS	100	1.50	1.51	---	1.46	1.46	---	1.44	1.45	---
	75	1.64	1.65	---	1.58	1.58	---	1.62	1.63	---
	65	1.78	1.79	---	1.68	1.68	---	1.76	1.77	---
	50	---	---	---	---	---	---	---	---	---
	50	2.06	2.07	---	1.90	1.90	---	2.04	2.05	---
	40	2.36	2.37	---	2.16	2.16	---	2.33	2.34	---
	30	2.80	2.81	---	2.52	2.52	---	2.73	2.74	---
	30 at > 50%F	2.80	2.81	---	2.71	2.71	---	2.73	2.74	---
	25 at > 50%F	3.06	3.07	---	3.01	3.01	---	3.03	3.04	---
	30 at ≤ 50%F	2.80	2.81	---	2.60	2.60	---	2.73	2.74	---
	25 at ≤ 50%F	2.91	2.92	---	2.87	2.87	---	2.97	2.98	---
PLUOOS	100	1.47	1.48	1.51	1.43	1.43	1.46	1.42	1.43	1.45
	75	1.61	1.62	1.65	1.55	1.55	1.58	1.55	1.56	1.63
	65	1.87	1.89	1.89	1.75	1.76	1.78	1.78	1.81	1.81
	50	---	---	---	---	---	---	---	---	---
	50	1.95	1.96	2.07	1.82	1.83	1.90	1.92	1.93	2.05
	40	2.25	2.26	2.37	2.06	2.06	2.16	2.20	2.21	2.34
	30	2.64	2.65	2.81	2.38	2.38	2.52	2.55	2.56	2.74
	30 at > 50%F	2.65	2.66	2.81	2.59	2.59	2.71	2.62	2.63	2.74
	25 at > 50%F	2.93	2.94	3.07	2.87	2.87	3.01	2.88	2.89	3.04
	30 at ≤ 50%F	2.64	2.65	2.81	2.50	2.50	2.60	2.57	2.58	2.74
	25 at ≤ 50%F	2.77	2.78	2.92	2.73	2.73	2.87	2.81	2.82	2.98

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.4 MCPR$_P$ Limits for All Fuel Types: Technical Specification Scram Time Basis (*continued*)*

Operating Condition	Power (% of rated)	ATRIUM-10			ATRIUM-10XM			ATRIUM-11		
		BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVOOS FHOOS	100	1.54	1.56	---	1.48	1.48	---	1.47	1.49	---
	75	1.69	1.70	---	1.61	1.61	---	1.63	1.64	---
	65	1.81	1.82	---	1.71	1.71	---	1.77	1.78	---
	50	---	---	---	---	---	---	---	---	---
	50	2.06	2.07	---	1.91	1.91	---	2.05	2.06	---
	40	2.37	2.38	---	2.16	2.16	---	2.33	2.34	---
	30	2.80	2.81	---	2.52	2.52	---	2.73	2.74	---
	30 at > 50%F	3.40	3.41	---	3.22	3.22	---	3.37	3.38	---
	25 at > 50%F	3.79	3.80	---	3.63	3.63	---	3.75	3.76	---
	30 at ≤ 50%F	3.17	3.18	---	2.99	2.99	---	3.19	3.20	---
	25 at ≤ 50%F	3.66	3.67	---	3.50	3.50	---	3.64	3.65	---
TBVOOS PLUOOS	100	1.52	1.54	1.56	1.47	1.47	1.48	1.46	1.47	1.49
	75	1.66	1.67	1.70	1.59	1.59	1.61	1.58	1.59	1.64
	65	1.87	1.89	1.89	1.75	1.76	1.78	1.78	1.81	1.81
	50	---	---	---	---	---	---	---	---	---
	50	1.96	1.97	2.07	1.83	1.83	1.91	1.93	1.94	2.06
	40	2.26	2.27	2.38	2.07	2.07	2.16	2.20	2.21	2.34
	30	2.64	2.65	2.81	2.38	2.38	2.52	2.55	2.56	2.74
	30 at > 50%F	3.26	3.27	3.41	3.09	3.09	3.22	3.23	3.24	3.38
	25 at > 50%F	3.67	3.68	3.80	3.52	3.52	3.63	3.61	3.62	3.76
	30 at ≤ 50%F	3.02	3.03	3.18	2.85	2.85	2.99	3.04	3.05	3.20
	25 at ≤ 50%F	3.45	3.46	3.67	3.32	3.32	3.50	3.45	3.46	3.65
FHOOS PLUOOS	100	1.50	1.51	---	1.46	1.46	---	1.44	1.45	---
	75	1.64	1.65	---	1.58	1.58	---	1.62	1.63	---
	65	1.87	1.89	---	1.75	1.76	---	1.78	1.81	---
	50	---	---	---	---	---	---	---	---	---
	50	2.06	2.07	---	1.90	1.90	---	2.04	2.05	---
	40	2.36	2.37	---	2.16	2.16	---	2.33	2.34	---
	30	2.80	2.81	---	2.52	2.52	---	2.73	2.74	---
	30 at > 50%F	2.80	2.81	---	2.71	2.71	---	2.73	2.74	---
	25 at > 50%F	3.06	3.07	---	3.01	3.01	---	3.03	3.04	---
	30 at ≤ 50%F	2.80	2.81	---	2.60	2.60	---	2.73	2.74	---
	25 at ≤ 50%F	2.91	2.92	---	2.87	2.87	---	2.97	2.98	---
TBVOOS FHOOS PLUOOS	100	1.54	1.56	---	1.48	1.48	---	1.47	1.49	---
	75	1.69	1.70	---	1.61	1.61	---	1.63	1.64	---
	65	1.87	1.89	---	1.75	1.76	---	1.78	1.81	---
	50	---	---	---	---	---	---	---	---	---
	50	2.06	2.07	---	1.91	1.91	---	2.05	2.06	---
	40	2.37	2.38	---	2.16	2.16	---	2.33	2.34	---
	30	2.80	2.81	---	2.52	2.52	---	2.73	2.74	---
	30 at > 50%F	3.40	3.41	---	3.22	3.22	---	3.37	3.38	---
	25 at > 50%F	3.79	3.80	---	3.63	3.63	---	3.75	3.76	---
	30 at ≤ 50%F	3.17	3.18	---	2.99	2.99	---	3.19	3.20	---
	25 at ≤ 50%F	3.66	3.67	---	3.50	3.50	---	3.64	3.65	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.5 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 1 for All Fuel Types: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
TBVIS	100	1.48	1.50	1.50	1.44	1.45	1.45	1.43	1.44	1.44
	75	1.64	1.64	1.64	1.57	1.57	1.57	1.58	1.59	1.59
	65	1.85	1.87	1.87	1.73	1.74	1.75	1.76	1.78	1.79
	50	---	---	---	---	---	---	---	---	---
	50	2.23	2.24	2.24	2.07	2.07	2.07	2.26	2.27	2.27
	40	2.60	2.61	2.61	2.38	2.38	2.38	2.62	2.63	2.63
	30	3.10	3.11	3.11	2.79	2.79	2.79	3.05	3.06	3.06
	30 at > 50%F	3.10	3.11	3.11	2.98	2.98	2.98	3.05	3.06	3.06
	25 at > 50%F	3.42	3.43	3.43	3.30	3.30	3.30	3.39	3.40	3.40
	30 at ≤ 50%F	3.10	3.11	3.11	2.86	2.86	2.86	3.05	3.06	3.06
	25 at ≤ 50%F	3.38	3.39	3.39	3.18	3.18	3.18	3.30	3.31	3.31
TBVOOS	100	1.53	1.54	1.54	1.47	1.48	1.48	1.46	1.47	1.47
	75	1.68	1.69	1.69	1.60	1.60	1.60	1.59	1.60	1.60
	65	1.85	1.87	1.87	1.73	1.74	1.75	1.76	1.78	1.79
	50	---	---	---	---	---	---	---	---	---
	50	2.23	2.24	2.24	2.07	2.07	2.07	2.26	2.27	2.27
	40	2.60	2.61	2.61	2.38	2.38	2.38	2.62	2.63	2.63
	30	3.10	3.11	3.11	2.79	2.79	2.79	3.05	3.06	3.06
	30 at > 50%F	3.60	3.61	3.61	3.42	3.42	3.42	3.62	3.63	3.63
	25 at > 50%F	4.03	4.04	4.04	3.86	3.86	3.86	4.03	4.04	4.04
	30 at ≤ 50%F	3.43	3.44	3.44	3.24	3.24	3.24	3.43	3.44	3.44
	25 at ≤ 50%F	3.93	3.94	3.94	3.78	3.78	3.78	3.94	3.95	3.95

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.6 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 2 for All Fuel Types: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10			ATRIUM-10XM			ATRIUM-11		
		BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVIS	100	1.48	1.50	1.50	1.44	1.45	1.45	1.43	1.44	1.44
	75	1.64	1.64	1.64	1.57	1.57	1.57	1.58	1.59	1.59
	65	1.85	1.87	1.87	1.73	1.74	1.75	1.76	1.78	1.79
	50	---	---	---	---	---	---	---	---	---
	50	2.25	2.26	2.26	2.08	2.08	2.08	2.28	2.29	2.29
	40	2.62	2.63	2.63	2.39	2.39	2.39	2.64	2.65	2.65
	30	3.12	3.13	3.13	2.82	2.82	2.82	3.08	3.09	3.09
	30 at > 50%F	3.12	3.13	3.13	2.99	2.99	2.99	3.08	3.09	3.09
	25 at > 50%F	3.44	3.45	3.45	3.32	3.32	3.32	3.41	3.42	3.42
	30 at ≤ 50%F	3.12	3.13	3.13	2.88	2.88	2.88	3.08	3.09	3.09
	25 at ≤ 50%F	3.40	3.41	3.41	3.20	3.20	3.20	3.32	3.33	3.33
TBVOOS	100	1.53	1.54	1.54	1.47	1.48	1.48	1.46	1.47	1.47
	75	1.68	1.69	1.69	1.60	1.60	1.60	1.59	1.60	1.60
	65	1.85	1.87	1.87	1.73	1.74	1.75	1.76	1.78	1.79
	50	---	---	---	---	---	---	---	---	---
	50	2.25	2.26	2.26	2.08	2.08	2.08	2.28	2.29	2.29
	40	2.62	2.63	2.63	2.39	2.39	2.39	2.64	2.65	2.65
	30	3.12	3.13	3.13	2.82	2.82	2.82	3.08	3.09	3.09
	30 at > 50%F	3.62	3.63	3.63	3.43	3.43	3.43	3.64	3.65	3.65
	25 at > 50%F	4.04	4.05	4.05	3.88	3.88	3.88	4.05	4.06	4.06
	30 at ≤ 50%F	3.45	3.46	3.46	3.26	3.26	3.26	3.45	3.46	3.46
	25 at ≤ 50%F	3.95	3.96	3.96	3.80	3.80	3.80	3.96	3.97	3.97

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.7 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 1 for All Fuel Types: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
TBVIS	100	1.50	1.51	1.51	1.46	1.46	1.46	1.44	1.45	1.45
	75	1.64	1.65	1.65	1.58	1.58	1.58	1.62	1.63	1.63
	65	1.87	1.89	1.89	1.75	1.76	1.78	1.78	1.81	1.81
	50	---	---	---	---	---	---	---	---	---
	50	2.29	2.30	2.30	2.10	2.10	2.10	2.32	2.33	2.33
	40	2.66	2.67	2.67	2.42	2.42	2.42	2.68	2.69	2.69
	30	3.17	3.18	3.18	2.85	2.85	2.85	3.12	3.13	3.13
	30 at > 50%F	3.17	3.18	3.18	2.98	2.98	2.98	3.12	3.13	3.13
	25 at > 50%F	3.42	3.43	3.43	3.30	3.30	3.30	3.39	3.40	3.40
	30 at ≤ 50%F	3.17	3.18	3.18	2.86	2.86	2.86	3.12	3.13	3.13
	25 at ≤ 50%F	3.38	3.39	3.39	3.18	3.18	3.18	3.30	3.31	3.31
TBVOOS	100	1.54	1.56	1.56	1.48	1.48	1.48	1.47	1.49	1.49
	75	1.69	1.70	1.70	1.61	1.61	1.61	1.63	1.64	1.64
	65	1.87	1.89	1.89	1.75	1.76	1.78	1.78	1.81	1.81
	50	---	---	---	---	---	---	---	---	---
	50	2.29	2.30	2.30	2.11	2.11	2.11	2.32	2.33	2.33
	40	2.66	2.67	2.67	2.42	2.42	2.42	2.68	2.69	2.69
	30	3.17	3.18	3.18	2.85	2.85	2.85	3.12	3.13	3.13
	30 at > 50%F	3.60	3.61	3.61	3.42	3.42	3.42	3.62	3.63	3.63
	25 at > 50%F	4.03	4.04	4.04	3.86	3.86	3.86	4.03	4.04	4.04
	30 at ≤ 50%F	3.43	3.44	3.44	3.24	3.24	3.24	3.43	3.44	3.44
	25 at ≤ 50%F	3.93	3.94	3.94	3.78	3.78	3.78	3.94	3.95	3.95

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.8 Startup Operation MCPR$_P$ Limits for Table 3.1 Temperature Range 2 for All Fuel Types: Technical Specification Scram Time Basis*

		ATRIUM-10			ATRIUM-10XM			ATRIUM-11		
Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVIS	100	1.50	1.51	1.51	1.46	1.46	1.46	1.44	1.45	1.45
	75	1.64	1.65	1.65	1.58	1.58	1.58	1.62	1.63	1.63
	65	1.87	1.89	1.89	1.75	1.76	1.78	1.78	1.81	1.81
	50	---	---	---	---	---	---	---	---	---
	50	2.31	2.32	2.32	2.11	2.11	2.11	2.33	2.34	2.34
	40	2.68	2.69	2.69	2.43	2.43	2.43	2.70	2.71	2.71
	30	3.20	3.21	3.21	2.88	2.88	2.88	3.15	3.16	3.16
	30 at > 50%F	3.20	3.21	3.21	2.99	2.99	2.99	3.15	3.16	3.16
	25 at > 50%F	3.44	3.45	3.45	3.32	3.32	3.32	3.41	3.42	3.42
	30 at ≤ 50%F	3.20	3.21	3.21	2.88	2.88	2.88	3.15	3.16	3.16
	25 at ≤ 50%F	3.40	3.41	3.41	3.20	3.20	3.20	3.32	3.33	3.33
TBVOOS	100	1.54	1.56	1.56	1.48	1.48	1.48	1.47	1.49	1.49
	75	1.69	1.70	1.70	1.61	1.61	1.61	1.63	1.64	1.64
	65	1.87	1.89	1.89	1.75	1.76	1.78	1.78	1.81	1.81
	50	---	---	---	---	---	---	---	---	---
	50	2.31	2.32	2.32	2.12	2.12	2.12	2.33	2.34	2.34
	40	2.68	2.69	2.69	2.43	2.43	2.43	2.70	2.71	2.71
	30	3.20	3.21	3.21	2.88	2.88	2.88	3.15	3.16	3.16
	30 at > 50%F	3.62	3.63	3.63	3.43	3.43	3.43	3.64	3.65	3.65
	25 at > 50%F	4.04	4.05	4.05	3.88	3.88	3.88	4.05	4.06	4.06
	30 at ≤ 50%F	3.45	3.46	3.46	3.26	3.26	3.26	3.45	3.46	3.46
	25 at ≤ 50%F	3.95	3.96	3.96	3.80	3.80	3.80	3.96	3.97	3.97

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.9 $MCPR_P$ Limits for All Fuel Types: Single Loop Operation for All Scram Times*

Operating Condition	Power (% of rated)	BOC to End of COAST		
		ATRIUM-10	ATRIUM-10XM	ATRIUM-11
RCPOOS FHOOS	100	2.09	2.01	2.07
	50	2.09	2.01	2.07
	40	2.39	2.18	2.36
	30	2.83	2.54	2.76
	30 at > 50%F	2.83	2.73	2.76
	25 at > 50%F	3.09	3.03	3.06
	30 at ≤ 50%F	2.83	2.62	2.76
	25 at ≤ 50%F	2.94	2.89	3.00
RCPOOS TBVOOS PLUOOS FHOOS	100	2.09	2.01	2.08
	50	2.09	2.01	2.08
	40	2.40	2.18	2.36
	30	2.83	2.54	2.76
	30 at > 50%F	3.43	3.24	3.40
	25 at > 50%F	3.82	3.65	3.78
	30 at ≤ 50%F	3.20	3.01	3.22
	25 at ≤ 50%F	3.69	3.52	3.67
RCPOOS TBVOOS FHOOS1	100	2.32	2.13	2.35
	50	2.32	2.13	2.35
	40	2.69	2.44	2.71
	30	3.20	2.87	3.15
	30 at > 50%F	3.63	3.44	3.65
	25 at > 50%F	4.06	3.88	4.06
	30 at ≤ 50%F	3.46	3.26	3.46
	25 at ≤ 50%F	3.96	3.80	3.97
RCPOOS TBVOOS FHOOS2	100	2.34	2.14	2.36
	50	2.34	2.14	2.36
	40	2.71	2.45	2.73
	30	3.23	2.90	3.18
	30 at > 50%F	3.65	3.45	3.67
	25 at > 50%F	4.07	3.90	4.08
	30 at ≤ 50%F	3.48	3.28	3.48
	25 at ≤ 50%F	3.98	3.82	3.99

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop.

RCPOOS limits are only valid up to 50% rated core power, 50% rated core flow, and an active recirculation drive flow of 17.73 Mlbm/hr.

5 Oscillation Power Range Monitor (OPRM) Setpoint

(Technical Specification 3.3.1.1)

Technical Specification Table 3.3.1.1-1, Function 2f, identifies the OPRM upscale function.

Instrument setpoints are established, such that the reactor will be tripped before an oscillation can grow to the point where the SLMCPR is exceeded. An Option III stability analysis is performed for each reload core to determine allowable OLMCPR's as a function of OPRM setpoint. Analyses consider both steady state startup operation, and the case of a two recirculation pump trip from rated power.

The resulting stability based OLMCPR's are reported in Reference 1. The OPRM setpoint (*sometimes referred to as the Amplitude Trip, S_p*) is selected, such that required margin to the SLMCPR is provided without stability being a limiting event. Analyses are based on cycle specific DIVOM analyses performed per Reference 22. The calculated OLMCPR's are shown in Table 5.1. Review of results shown in COLR Table 4.2 indicates an OPRM setpoint of **1.14** may be used. The successive confirmation count (*sometimes referred to as N_p*) is provided in Table 5.2, per Reference 30.

Table 5.1 OPRM Setpoint Range*

OPRM Setpoint	OLMCPR (SS)	OLMCPR (2PT)
1.05	1.15	1.20
1.06	1.17	1.21
1.07	1.19	1.23
1.08	1.20	1.25
1.09	1.22	1.27
1.10	1.24	1.29
1.11	1.26	1.31
1.12	1.28	1.34
1.13	1.30	1.36
1.14	1.33	1.38
1.15	1.35	1.40

Table 5.2 OPRM Successive Confirmation Count Setpoint

Count	OPRM Setpoint
6	≥ 1.04
8	≥ 1.05
10	≥ 1.07
12	≥ 1.09
14	≥ 1.11
16	≥ 1.14
18	≥ 1.18
20	≥ 1.24

* Extrapolation beyond a setpoint of 1.15 is not allowed

6 APRM Flow Biased Rod Block Trip Settings

(Technical Requirements Manual Section 5.3.1 and Table 3.3.4-1)

The APRM rod block trip setting is based upon References 26 & 27, and is defined by the following:

SRB ≤	(0.66(W-ΔW) + 61%)	Allowable Value
SRB ≤	(0.66(W-ΔW) + 59%)	Nominal Trip Setpoint (NTSP)

where:

SRB	=	Rod Block setting in percent of rated thermal power (3458 MW_t)
W	=	Loop recirculation flow rate in percent of rated
ΔW	=	Difference between two-loop and single-loop effective recirculation flow at the same core flow (ΔW=0.0 for two-loop operation)

The APRM rod block trip setting is clamped at a maximum allowable value of 115% (corresponding to a NTSP of 113%).

7 Rod Block Monitor (RBM) Trip Setpoints and Operability

(Technical Specification Table 3.3.2.1-1)

The RBM trip setpoints and applicable power ranges, based on References 26 & 27, are shown in Table 7.1. Setpoints are based on an HTSP, unfiltered analytical limit of 114%. Unfiltered setpoints are consistent with a nominal RBM filter setting of 0.0 seconds; filtered setpoints are consistent with a nominal RBM filter setting less than 0.5 seconds. Cycle specific CRWE analyses of OLMCPR are documented in Reference 1, superseding values reported in References 26, 27, and 29.

Table 7.1 Analytical RBM Trip Setpoints*

RBM Trip Setpoint		Allowable Value (AV)	Nominal Trip Setpoint (NTSP)
LPSP		27%	25%
IPSP		62%	60%
HPSP		82%	80%
LTSP	- unfiltered	121.7%	120.0%
	- filtered	120.7%	119.0%
ITSP	- unfiltered	116.7%	115.0%
	- filtered	115.7%	114.0%
HTSP	- unfiltered	111.7%	110.0%
	- filtered	110.9%	109.2%
DTSP		90%	92%

As a result of cycle specific CRWE analyses, RBM setpoints in Technical Specification Table 3.3.2.1-1 are applicable as shown in Table 7.2. Cycle specific setpoint analysis results are shown in Table 7.3, per Reference 1.

Table 7.2 RBM Setpoint Applicability

Thermal Power (% Rated)	Applicable MCPR†	Notes from Table 3.3.2.1-1	Comment
> 27% and < 90%	< 1.70	(a), (b), (f), (h)	two loop operation
	< 1.74	(a), (b), (f), (h)	single loop operation
≥ 90%	< 1.39	(g)	two loop operation‡

* Values are considered maximums. Using lower values, due to RBM system hardware/software limitations, is conservative, and acceptable.

† MCPR values shown correspond with, (support), SLMPCR values identified in Reference 1.

‡ Greater than 90% rated power is not attainable in single loop operation.

Table 7.3 Control Rod Withdrawal Error Results

RBM HTSP Analytical Limit Unfiltered	CRWE OLMCPR
107	1.20
111	1.26
114	1.37
117	1.37

Results, compared against the base case OLMCPR results of Table 4.2, indicate SLMCPR remains protected for RBM inoperable conditions (i.e., 114% unblocked).

8 Shutdown Margin Limit

(Technical Specification 3.1.1)

Assuming the strongest OPERABLE control blade is fully withdrawn, and all other OPERABLE control blades are fully inserted, the core shall be sub-critical and meet the following minimum shutdown margin:

SDM > 0.38% dk/k

EDMS L32 160809 800
QA Document
BFE-3843, Revision 5

TVA NPG

Reactor Engineering and Fuels - BWRFE
1101 Market Street, Chattanooga, TN 37402

Browns Ferry Unit 2 Cycle 19
Core Operating Limits Report, (105% OLTP)

TVA-COLR-BF2C19 **Revision 5 (Final)**
(Revision Log, Page v)

August 2016

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Total Number of Pages = 54 (*including review cover sheet*)

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Revision Log

Number	Page	Description
0-R5		Revision Driven by CR 1145408: feedwater runout flow
1-R5	ix	Updated References 1, 4
2-R5	1	Revised Section 1.1.1; identify new basis of revised reload safety analysis report.
3-R5	13, 19, 20,	Updated Figures 3.4, 3.10, and 3.11 due to Reference 1 changes.
4-R5	29-38	Updated Tables 4.2 through 4.9 due to Reference 1 changes.
0-R4		Revision driven by CRs 1090560 and 1097540: code errors
1-R4	ix	Updated Reference 1.
2-R4	1	Revised Section 1.1.1; identify new basis of revised reload safety analysis report.
3-R4	39	Updated Table 5.1 2 pump trip column of results due to change in Reference 1
0-R3		No Changes to Section 2; Sections 5 through 8.
1-R3	ix	Updated References 1 & 4.
2-R3	xi	Removed References 31 & 32 as they are no longer applicable due to Reference 1 update.
3-R3	1	Revised Section 1.1.1; identify new basis of revised reload safety analysis report.
4-R3	8	Revised Section 3.2; removed cross references coincident with change 2-R3. Also, revised Section 3.2.1; removed last sentence as items no longer applicable. Updated pointers to Figure numbers.
5-R3	9	Revised Section 3.4; updated Figure number cross references coincident with change 8-R3.
6-R3	13-15, 19-24	Revised Section 3, Figures 3.4 -3.6; 3.10 - 3.15 based on new reload safety analysis report.
7-R3	old 25-26	Removed Figures 3.16 - 3.17 as they are no longer applicable.
8-R3	25	Revised Section 4.2.1; removed last sentence as items no longer applicable. Update pointers to Table numbers.
9-R3	27	Section 4.2.5 updated Figure number cross reference.
10-R3	29-38	Updated Tables 4.2 through 4.9 with the results from revised reload safety analysis report.
11-R3	old 40-43	Removed old Tables 4.9 - 4.12 as they are no longer applicable.
1-R2	x	Added Reference 32 supplement to BFN Reload Analysis Report
2-R2	1	Revised Section 1.1.1 to identify the new range of applicability for startup with feedwater heating out of service

3-R2	8	Revised last sentence of Section 3.2.1 to reflect the updated cycle exposure range of applicability (3000 MWd/MTU); also added Reference 32
4-R2	25-26	Revised Section 3, Figure 3.16-17 titles to reflect the updated cycle exposure range of applicability (3000 MWd/MTU). No limits were changed, only titles.
5-R2	27	Revised last sentence of Section 4.2.1 to reflect the updated cycle exposure range of applicability (3000 MWd/MTU); also added Reference 32.
6-R2	40-43	Revised Section 4, Tables 4.9 - 4.12 to reflect the updated cycle exposure range of applicability (3000 MWd/MTU). No limits were changed, only titles
1-R1	x	Added Reference 31 supplement to BFN Unit 2 Cycle 19 Reload Analysis
2-R1	1-49	Corrected page numbering. Revision 0 did not reset the page numbering at Section 1 to start at 1, after the roman numeral sections at the front of the document. The new Arabic numbered pages for Sections 1 through 8 begin with page 1 and go through page 49.
3-R1	1	Added Section 1.1.1 to describe Revision 1 purpose
4-R1	8	Added discussion of Reference 31 for $LHGRFAC_p$
5-R1	9	Figure number range increased to Figure 3.17
6-R1	25-26	Added Figure 3.16 & Figure 3.17, $LHGRFAC_p$ for ATRIUM-10 fuel BOC - 250 MWd/MTU
7-R1	27	Added discussion of Reference 31 for $MCPR_p$
8-R1	29	Renumbered Table 4.9 as Table 4.13
9-R1	40-43	Inserted Table 4.9 through Table 4.12, BOC – 250 MWd/MTU Startup Operation $MCPR_p$ Limits
10-R1	44	Renumbered Table 4.9 as Table 4.13
0-R0	All	New document.

Nomenclature

APLHGR	Average Planar LHGR
APRM	Average Power Range Monitor
AREVA NP	Vendor (Framatome, Siemens)
BOC	Beginning of Cycle
BSP	Backup Stability Protection
BWR	Boiling Water Reactor
CAVEX	Core Average Exposure
CD	Coast Down
CMSS	Core Monitoring System Software
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRWE	Control Rod Withdrawal Error
CSDM	Cold SDM
DIVOM	Delta CPR over Initial CPR vs. Oscillation Magnitude
EOC	End of Cycle
EOCLB	End-of-Cycle Licensing Basis
EOOS	Equipment OOS
FFTR	Final Feedwater Temperature Reduction
FFWTR	Final Feedwater Temperature Reduction
FHOOS	Feedwater Heaters OOS
ft	Foot: english unit of measure for length
GNF	Vendor (General Electric, Global Nuclear Fuels)
GWd	Giga Watt Day
HTSP	High TSP
ICA	Interim Corrective Action
ICF	Increased Core Flow (beyond rated)
IS	In-Service
kW	kilo watt: SI unit of measure for power.
LCO	License Condition of Operation
LFWH	Loss of Feedwater Heating
LHGRFAC	LHGR Multiplier (Power or Flow dependent)
LPRM	Low Power Range Monitor
LRNB	Generator Load Reject, No Bypass
MAPFAC	MAPLHGR multiplier (Power or Flow dependent)
MCPR	Minimum CPR

MSRV	Moisture Separator Reheater Valve
MSRVOOS	MSRV OOS
MTU	Metric Ton Uranium
MWd/MTU	Mega Watt Day per Metric Ton Uranium
NEOC	Near EOC
NRC	United States Nuclear Regulatory Commission
NSS	Nominal Scram Speed
NTSP	Nominal TSP
OLMCPR	MCPR Operating Limit
OOS	Out-Of-Service
OPRM	Oscillation Power Range Monitor
OSS	Optimum Scram Speed
PBDA	Period Based Detection Algorithm
Pbypass	Power, below which TSV Position and TCV Fast Closure Scrams are Bypassed
PLU	Power Load Unbalance
PLUOOS	PLU OOS
PRNM	Power Range Neutron Monitor
RBM	Rod Block Monitor
RCPOOS	Recirculation Pump OOS (*SLO*)
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPTOOS	RPT OOS
SDM	Shutdown Margin
SLMCPR	MCPR Safety Limit
SLO	Single Loop Operation
TBV	Turbine Bypass Valve
TBVIS	TBV IS
TBVOOS	Turbine Bypass Valves OOS
TIP	Transversing In-core Probe
TIPOOS	TIP OOS
TLO	Two Loop Operation
TSP	Trip Setpoint
TSSS	Technical Specification Scram Speed
TVA	Tennessee Valley Authority

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PRNM Setpoint References

26. Filtered Setpoints - EDE-28-0990 Rev. 3 Supplement E, "PRNM (APRM, RBM, and RFM) Setpoint Calculations [ARTS/MELLL (NUMAC) - Power-Uprate Condition] for Tennessee Valley Authority Browns Ferry Nuclear Plant", October 1997.

27. Unfiltered Setpoints - EDE-28-0990 Rev. 2 Supplement E, "PRNM (APRM, RBM, and RFM) Setpoint Calculations [ARTS/MELLL (NUMAC) - Power-Uprate Condition] for Tennessee Valley Authority Browns Ferry Nuclear Plant", October 1997.

28. GE Letter LB#: 262-97-133, Browns Ferry Nuclear Plant Rod Block Monitor Setpoint Clarification - GE Proprietary Information, September 12, 1997.

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

30. NEDO-32465-A, **Licensing Topical Report – Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications**, GE Nuclear Energy, August 1996.

1 Introduction

In anticipation of cycle startup, it is necessary to describe the expected limits of operation.

1.1 Purpose

The primary purpose of this document is to satisfy requirements identified by unit technical specification section 5.6.5. This document may be provided, upon final approval, to the NRC.

1.1.1 Revision 5

The purpose of this revision is a consequence of the Reference 1 reload safety analyses revision. The Reference 1 revision is in response to condition report (CR) 1145408 feedwater runout flow.

1.2 Scope

This document will discuss the following areas:

- Average Planar Linear Heat Generation Rate (APLHGR) Limit
  (Technical Specifications 3.2.1 and 3.7.5)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Linear Heat Generation Rate (LHGR) Limit
  (Technical Specification 3.2.3, 3.3.4.1, and 3.7.5)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Minimum Critical Power Ratio Operating Limit (OLMCPR)
  (Technical Specifications 3.2.2, 3.3.4.1, 3.7.5 and Table 3.3.2.1-1)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Oscillation Power Range Monitor (OPRM) Setpoint
  (Technical Specification Table 3.3.1.1)
  Applicability: Mode 1, ≥ (as specified in Technical Specifications Table 3.3.1.1-1)
- Average Power Range Monitor (APRM) Flow Biased Rod Block Trip Setting
  (Technical Requirements Manual Section 5.3.1 and Table 3.3.4-1)
  Applicability: Mode 1, ≥ (as specified in Technical Requirements Manuals Table 3.3.4-1)
- Rod Block Monitor (RBM) Trip Setpoints and Operability
  (Technical Specification Table 3.3.2.1-1)
  Applicability: Mode 1, ≥ % RTP as specified in Table 3.3.2.1-1 (TS definition of RTP)
- Shutdown Margin (SDM) Limit
  (Technical Specification 3.1.1)
  Applicability: All Modes

1.3 Fuel Loading

The core will contain previously exposed AREVA NP, Inc., ATRIUM-10 fuel, along with fresh ATRIUM-10XM and ATRIUM-11 fuel. Nuclear fuel types used in the core loading are shown in Table 1.1. The core shuffle and final loading were explicitly evaluated for BOC cold shutdown margin performance as documented per Reference 5.

Table 1.1 Nuclear Fuel Types*

Fuel Description	Original Cycle	Number of Assemblies	Nuclear Fuel Type (NFT)	Fuel Names (Range)
ATRIUM-10 A10-3799B-14GV80-FBD	17	73	10	FBD001-FBD136
ATRIUM-10 A10-4004B-15GV80-FBD	17	115	11	FBD137-FBD272
ATRIUM-10 A10-4165B-15GV75-FBE	18	176	12	FBE001-FBE176
ATRIUM-10 A10-4107B-13GV75-FBE	18	68	13	FBE177-FBE244
ATRIUM-10 A10-4176B-10GV75-FBE	18	72	14	FBE245-FBE316
ATRIUM 10XM XMLC-3904B-15GV80-FBF	19	172	15	FBF401-FBF572
ATRIUM 10XM XMLC-4035B-13GV80-FBF	19	80	16	FBF573-FBF652
ATRIUM 11 A11-3693B-13GV80-FBF	19	8	17	FBF653-FBF660

1.4 Acceptability

Limits discussed in this document were generated based on NRC approved methodologies per References 6 through 25.

* The table identifies the expected fuel type breakdown in anticipation of final core loading. The final composition of the core depends upon uncertainties during the outage such as discovering a failed fuel bundle, or other bundle damage. Minor core loading changes, due to unforeseen events, will conform to the safety and monitoring requirements identified in this document.

2 APLHGR Limits

(Technical Specifications 3.2.1 & 3.7.5)

The APLHGR limit is determined by adjusting the rated power APLHGR limit for off-rated power, off-rated flow, and SLO conditions. The most limiting of these is then used as follows:

$$\text{APLHGR limit} = \text{MIN}\ (\text{APLHGR}_P\ ,\ \text{APLHGR}_F,\ \text{APLHGR}_{SLO}\ )$$

where:

$APLHGR_P$	off-rated power APLHGR limit	[$APLHGR_{RATED}$ * $MAPFAC_P$]
$APLHGR_F$	off-rated flow APLHGR limit	[$APLHGR_{RATED}$ * $MAPFAC_F$]
$APLHGR_{SLO}$	SLO APLHGR limit	[$APLHGR_{RATED}$ * SLO Multiplier]

2.1 Rated Power and Flow Limit: $APLHGR_{RATED}$

The rated conditions APLHGR for ATRIUM-10 fuel is identified in Reference 1 and shown in Figure 2.1. The rated conditions APLHGR for ATRIUM-10XM are shown in Figure 2.2. The rated conditions APLHGR for ATRIUM-11 are shown in Figure 2.3.

2.2 Off-Rated Power Dependent Limit: $APLHGR_P$

Reference 1 does not specify a power dependent APLHGR. Therefore, $MAPFAC_P$ is set to a value of **1.0**.

2.2.1 Startup without Feedwater Heaters

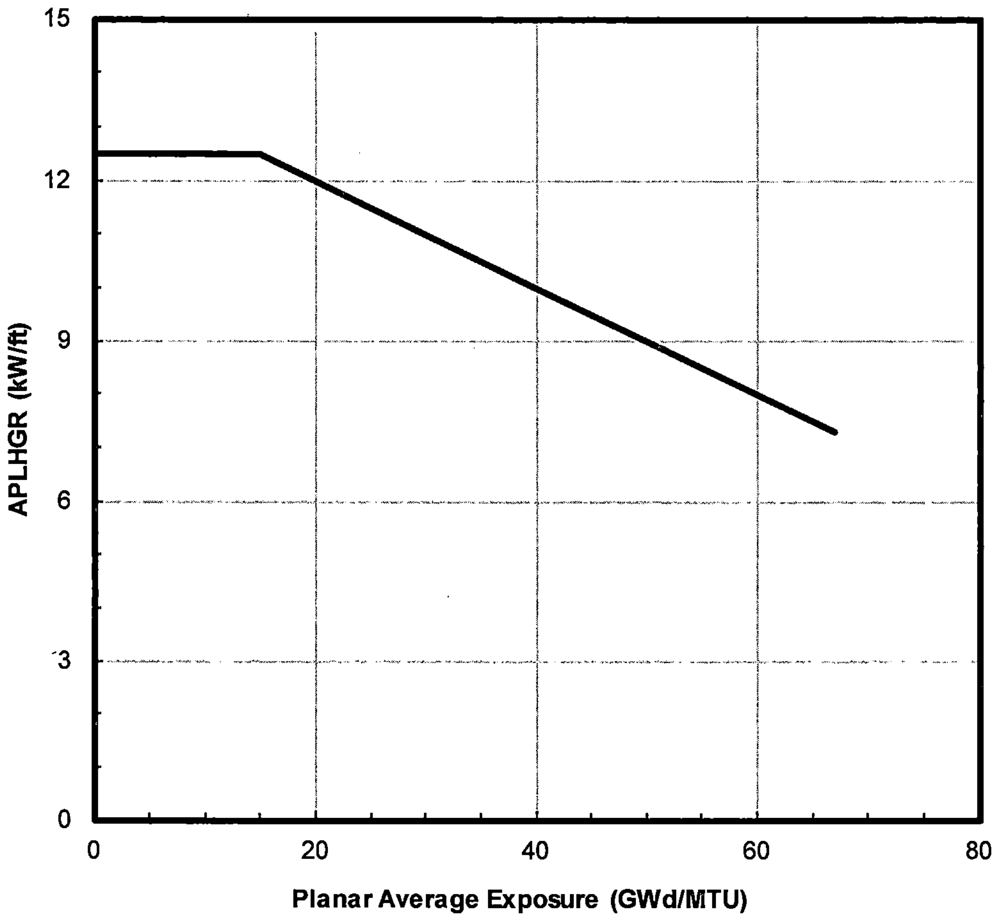
There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. No additional power dependent limitation is required.

2.3 Off-Rated Flow Dependent Limit: $APLHGR_F$

Reference 1 does not specify a flow dependent APLHGR. Therefore, $MAPFAC_F$ is set to a value of **1.0**.

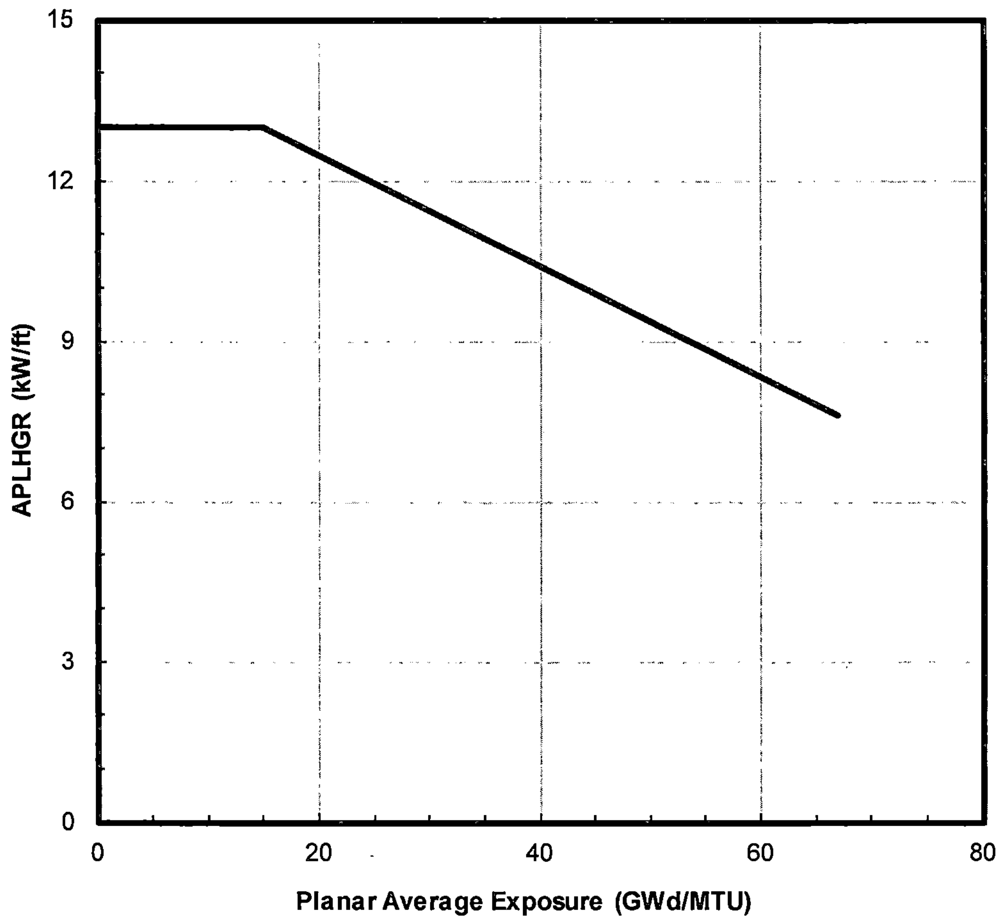
2.4 Single Loop Operation Limit: $APLHGR_{SLO}$

The single loop operation multiplier for ATRIUM-10, ATRIUM-10XM, and ATRIUM-11 fuel is **0.85**, per Reference 1.



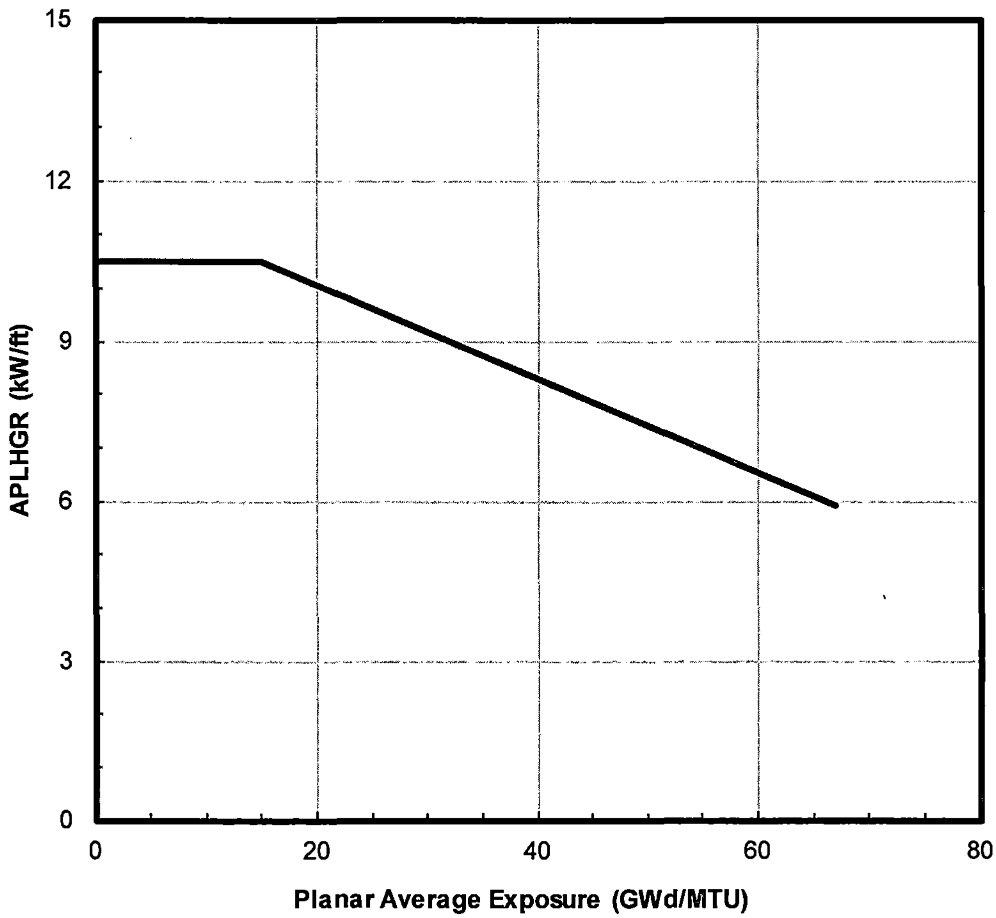
Planar Avg. Exposure (GWd/MTU)	APLHGR Limit (kW/ft)
0.0	12.5
15.0	12.5
67.0	7.3

Figure 2.1 $APLHGR_{RATED}$ for ATRIUM-10 Fuel



Planar Avg. Exposure (GWd/MTU)	APLHGR Limit (kW/ft)
0.0	13.0
15.0	13.0
67.0	7.6

Figure 2.2 APLHGR$_{RATED}$ for ATRIUM-10XM Fuel



Planar Avg. Exposure (GWd/MTU)	APLHGR Limit (kW/ft)
0.0	10.5
15.0	10.5
67.0	5.9

Figure 2.3 APLHGR$_{RATED}$ for ATRIUM-11 Fuel

2.5 Equipment Out-Of-Service Corrections

The limits shown in Figure 2.1, Figure 2.2, and Figure 2.3 are applicable for operation with all equipment In-Service as well as the following Equipment Out-Of-Service (EOOS) options; including combinations of the options.

In-Service	All equipment In-Service*
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
PLUOOS	Power Load Unbalance Out-Of-Service
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service or Final Feedwater Temperature Reduction
RCPOOS	One Recirculation Pump Out-Of-Service

* All equipment service conditions assume 1 SRVOOS.

3 LHGR Limits

(Technical Specification 3.2.3, 3.3.4.1, & 3.7.5)

The LHGR limit is determined by adjusting the rated power LHGR limit for off-rated power and off-rated flow conditions. The most limiting of these is then used as follows:

LHGR limit = MIN ($LHGR_P$, $LHGR_F$)

where:

$LHGR_P$	off-rated power LHGR limit	[$LHGR_{RATED}$ * $LHGRFAC_P$]
$LHGR_F$	off-rated flow LHGR limit	[$LHGR_{RATED}$ * $LHGRFAC_F$]

3.1 Rated Power and Flow Limit: $LHGR_{RATED}$

The rated conditions LHGR for ATRIUM-10 fuel is identified in Reference 1 and shown in Figure 3.1. The rated conditions LHGR for ATRIUM-10XM fuel is shown in Figure 3.2. The rated conditions LHGR for ATRIUM-11 fuel is shown in Figure 3.3. The LHGR limit is consistent with References 2 and 3.

3.2 Off-Rated Power Dependent Limit: $LHGR_P$

LHGR limits are adjusted for off-rated power conditions using the $LHGRFAC_P$ multiplier provided in Reference 1. The multiplier is split into two sub cases: turbine bypass valves in and out-of-service. The base case multipliers are shown in Figure 3.4 through Figure 3.6.

3.2.1 <u>Startup without Feedwater Heaters</u>

There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. Additional limits are shown in Figure 3.10 through Figure 3.15, based on temperature conditions identified in Table 3.1.

Table 3.1 Startup Feedwater Temperature Basis

Power	Temperature Range 1	Temperature Range 2
(% Rated)	(°F)	(°F)
25	160.0	155.0
30	165.0	160.0
40	175.0	170.0
50	185.0	180.0

3.3 Off-Rated Flow Dependent Limit: $LHGR_F$

LHGR limits are adjusted for off-rated flow conditions using the $LHGRFAC_F$ multiplier provided in Reference 1. Multipliers are shown in Figure 3.7 through Figure 3.9.

3.4 Equipment Out-Of-Service Corrections

The limits shown in Figure 3.1 through Figure 3.3 are applicable for operation with all equipment In-Service as well as the following Equipment Out-Of-Service (EOOS) options; including combinations of the options.*

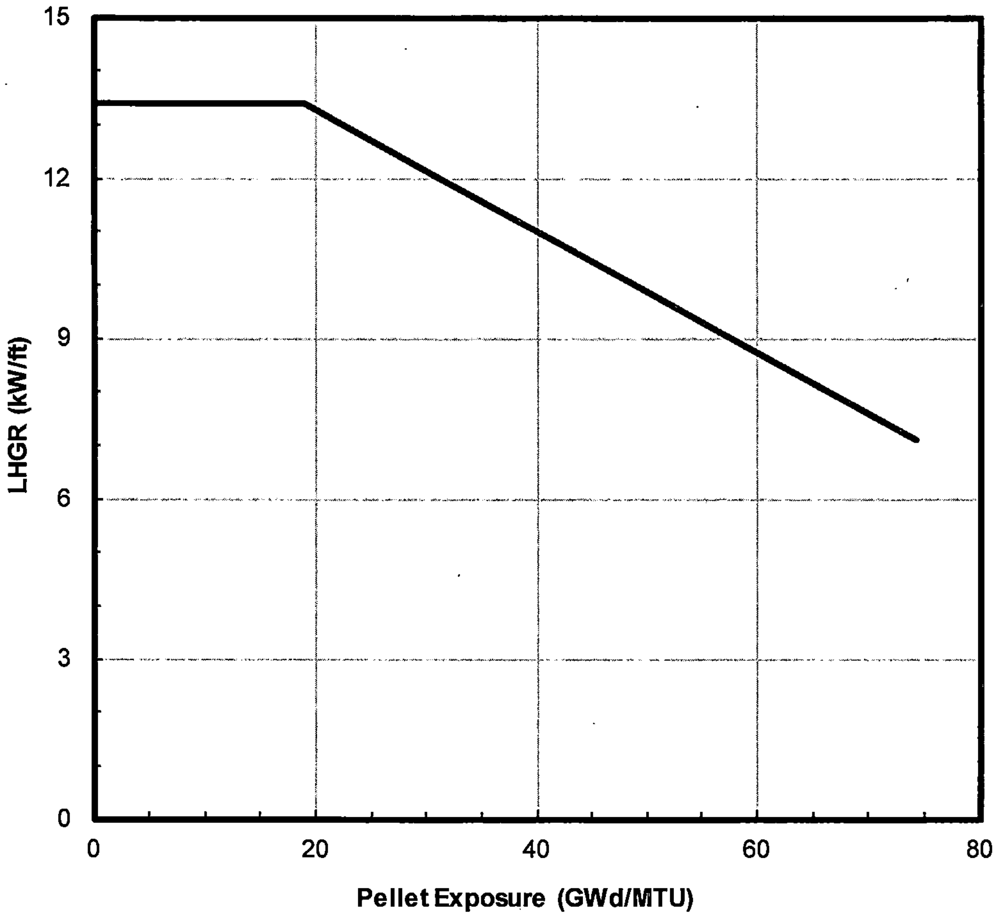
In-Service	All equipment In-Service
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
PLUOOS	Power Load Unbalance Out-Of-Service
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service or Final Feedwater Temperature Reduction
RCPOOS	One Recirculation Pump Out-Of-Service

Off-rated power corrections shown in Figure 3.4 through Figure 3.6 are dependent on operation of the Turbine Bypass Valve system. For this reason, separate limits are to be applied for TBVIS or TBVOOS operation. The limits have no dependency on RPTOOS, PLUOOS, FHOOS/FFWTR, or SLO.

Off-rated flow corrections shown in Figure 3.7 through Figure 3.9 are bounding for all EOOS conditions.

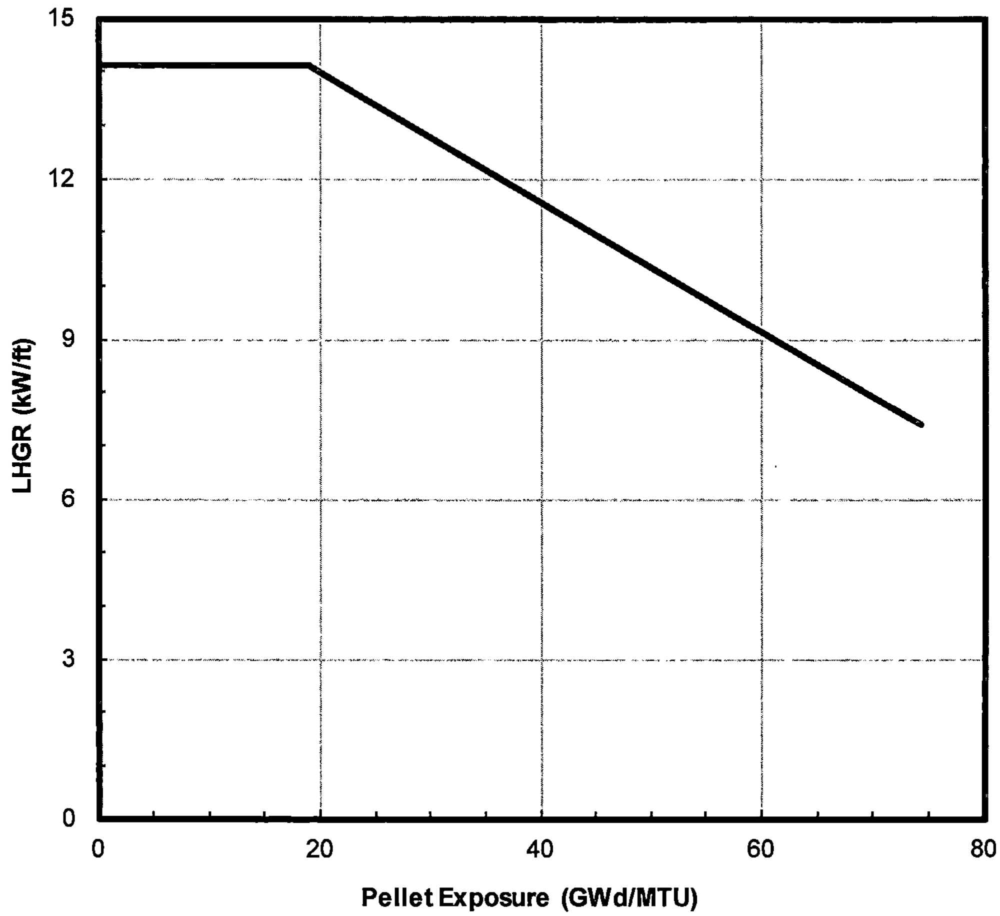
Off-rated power corrections shown in Figure 3.10 through Figure 3.15 are also dependent on operation of the Turbine Bypass Valve system. In this case, limits support FHOOS operation during startup. These limits have no dependency on RPTOOS, PLUOOS, or SLO.

* All equipment service conditions assume 1 SRVOOS.



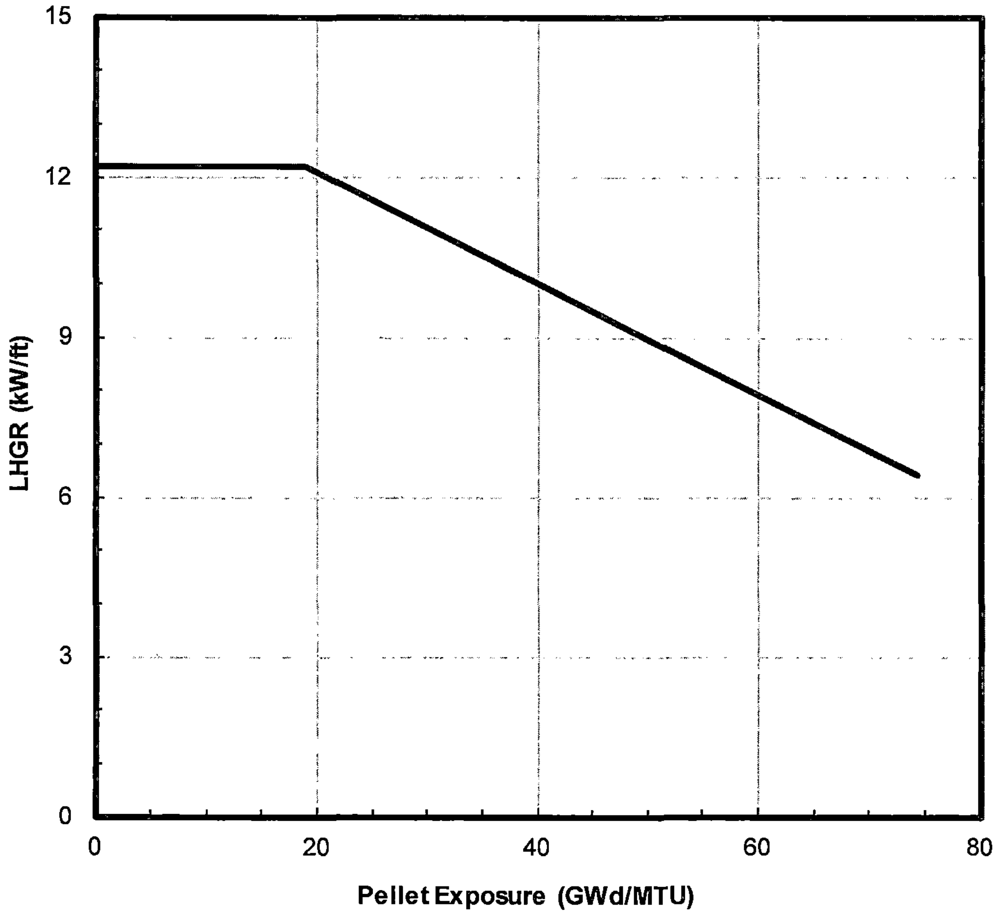
Pellet Exposure (GWd/MTU)	LHGR Limit (kW/ft)
0.0	13.4
18.9	13.4
74.4	7.1

Figure 3.1 LHGR$_{RATED}$ for ATRIUM-10 Fuel



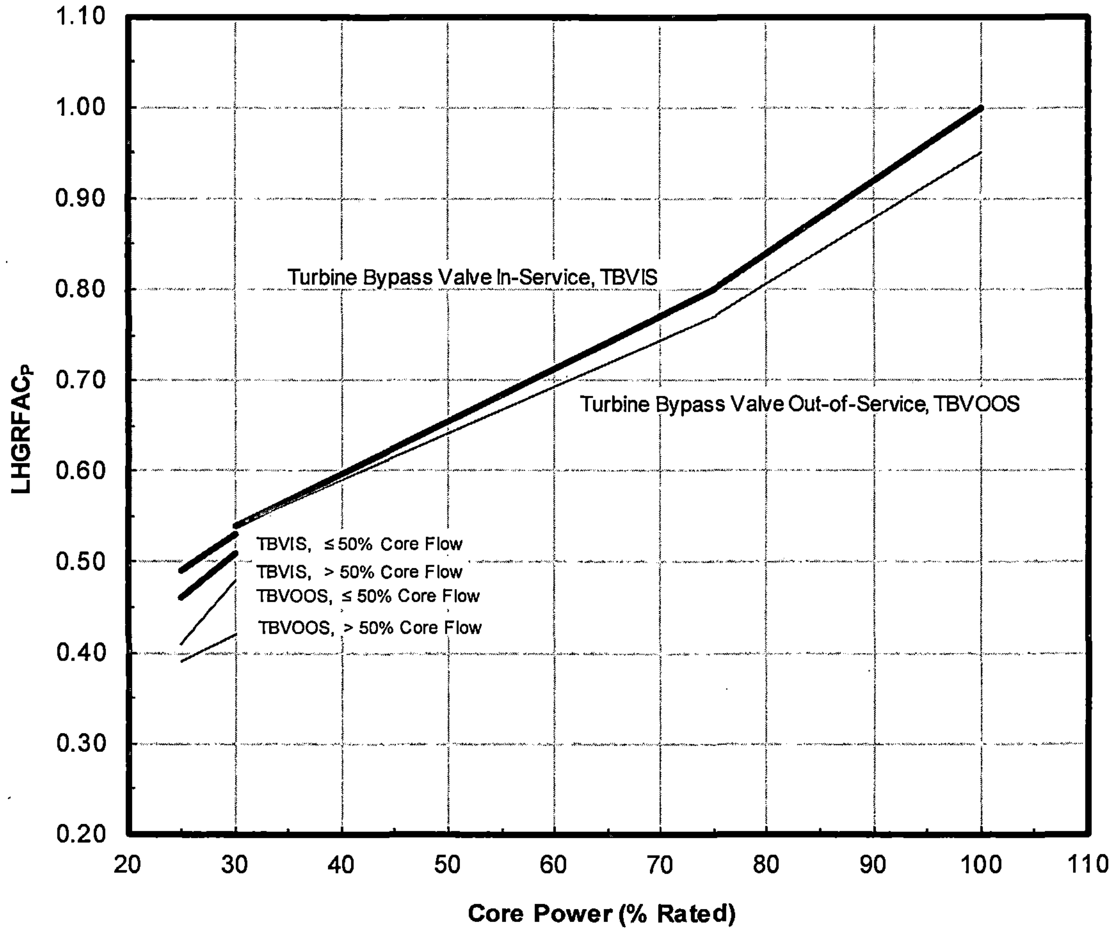
Pellet Exposure (GWd/MTU)	LHGR Limit (kW/ft)
0.0	14.1
18.9	14.1
74.4	7.4

Figure 3.2 $LHGR_{RATED}$ for ATRIUM-10XM Fuel



Pellet Exposure (GWd/MTU)	LHGR Limit (kW/ft)
0.0	12.2
18.9	12.2
74.4	6.4

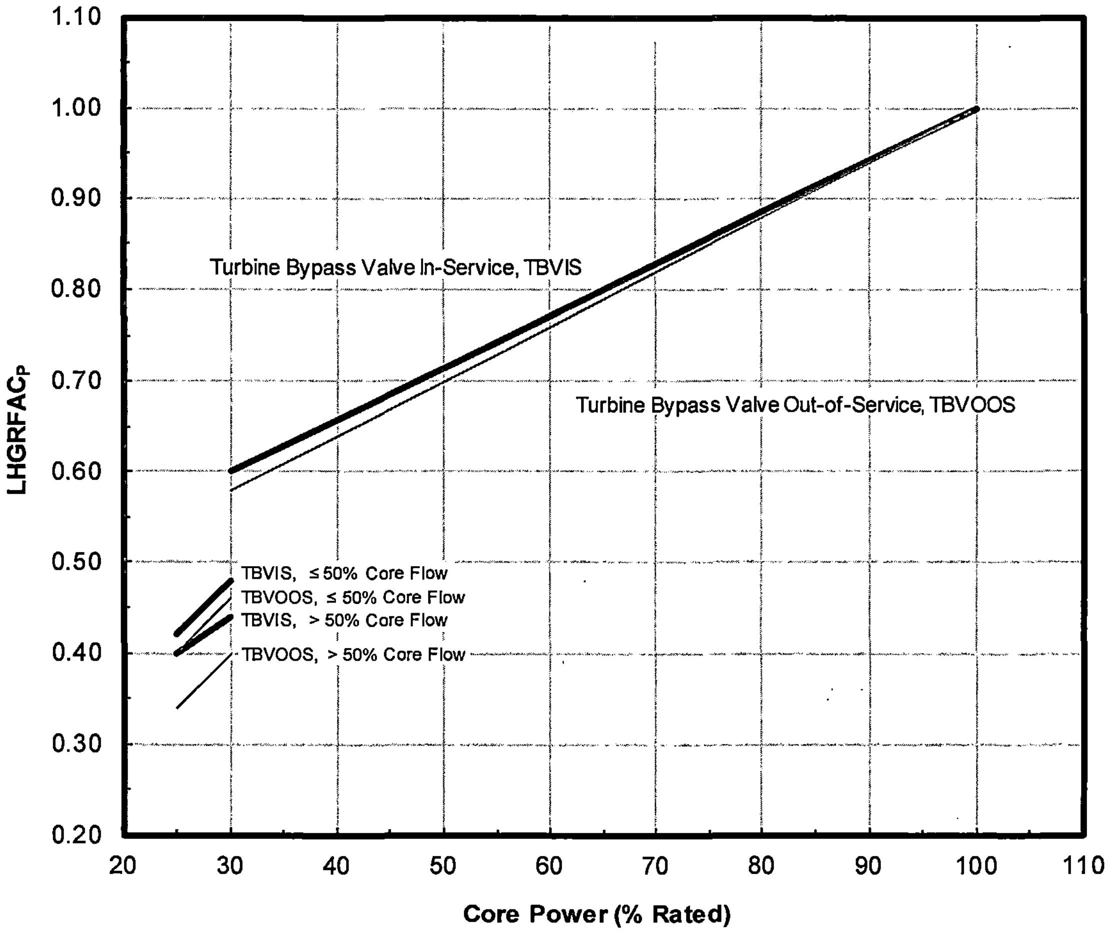
Figure 3.3 $LHGR_{RATED}$ for ATRIUM-11 Fuel



Turbine Bypass In-Service		*Turbine Bypass Out-of-Service*	
Core Power (% Rated)	**$LHGRFAC_P$**	**Core Power (% Rated)**	**$LHGRFAC_P$**
100.0	1.00	100.0	0.95
75.0	0.80	75.0	0.77
30.0	0.54	30.0	0.54
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.51	30.0	0.42
25.0	0.46	25.0	0.39
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.53	30.0	0.48
25.0	0.49	25.0	0.41

Figure 3.4 Base Operation $LHGRFAC_P$ for ATRIUM-10 Fuel

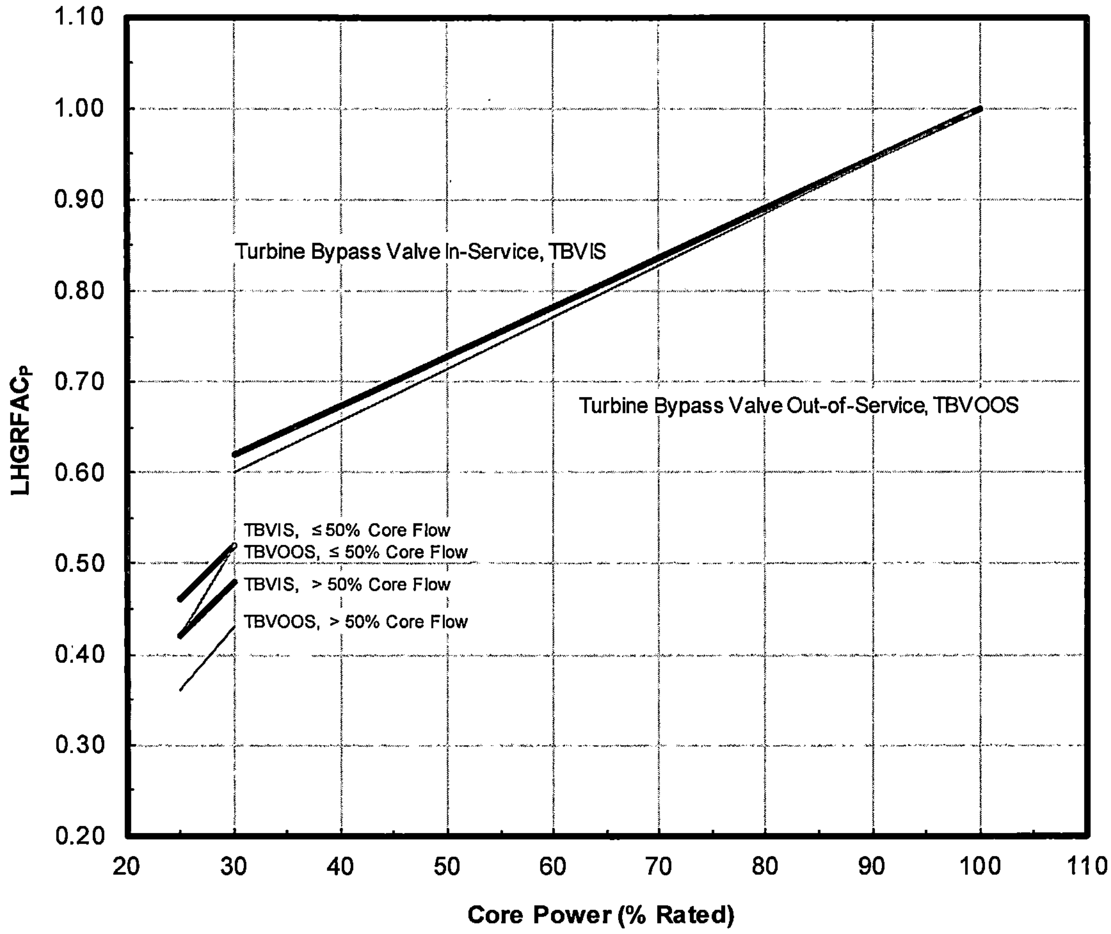
(*Independent of other EOOS conditions*)



Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$	*Turbine Bypass Out-of-Service* Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00	100.0	1.00
30.0	0.60	30.0	0.58
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.44	30.0	0.40
25.0	0.40	25.0	0.34
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.48	30.0	0.46
25.0	0.42	25.0	0.40

Figure 3.5 Base Operation LHGRFAC$_P$ for ATRIUM-10XM Fuel

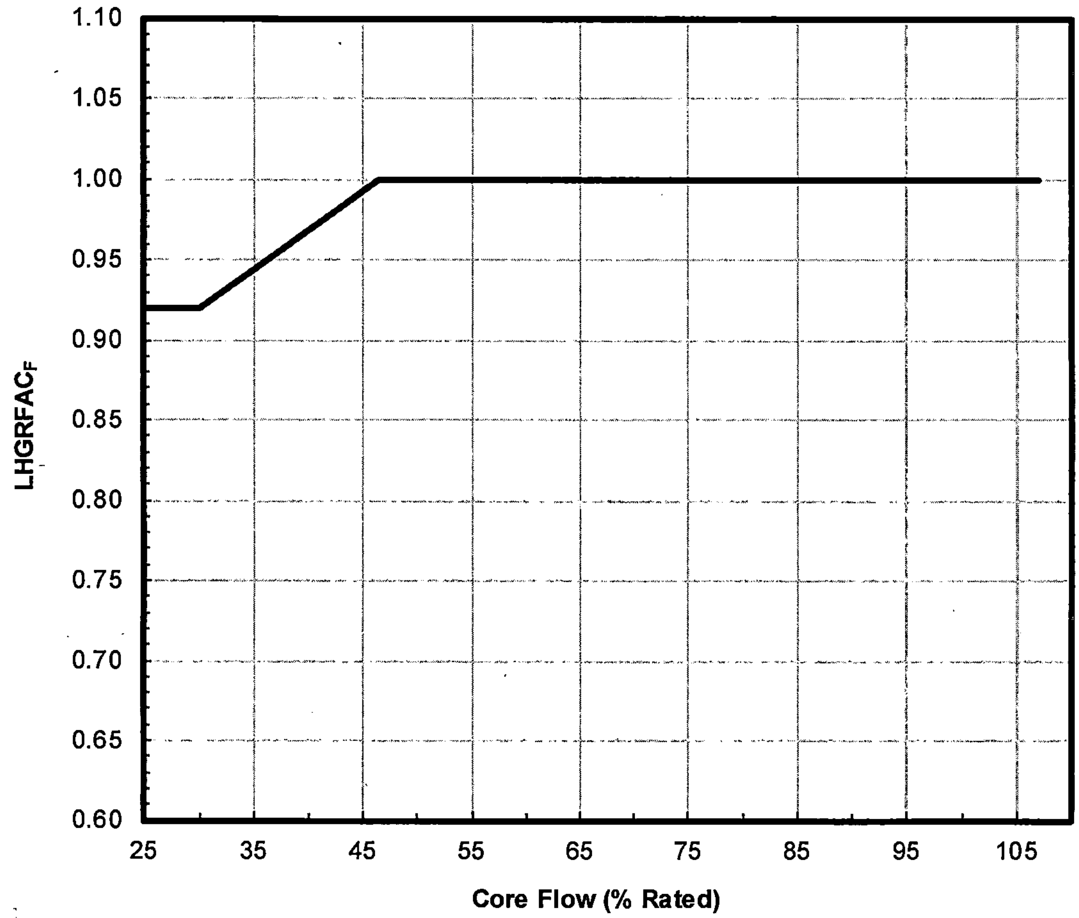
(*Independent of other EOOS conditions*)



Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$	*Turbine Bypass Out-of-Service* Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00	100.0	1.00
30.0	0.62	30.0	0.60
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.48	30.0	0.43
25.0	0.42	25.0	0.36
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.52	30.0	0.52
25.0	0.46	25.0	0.42

Figure 3.6 Base Operation LHGRFAC$_P$ for ATRIUM-11 Fuel

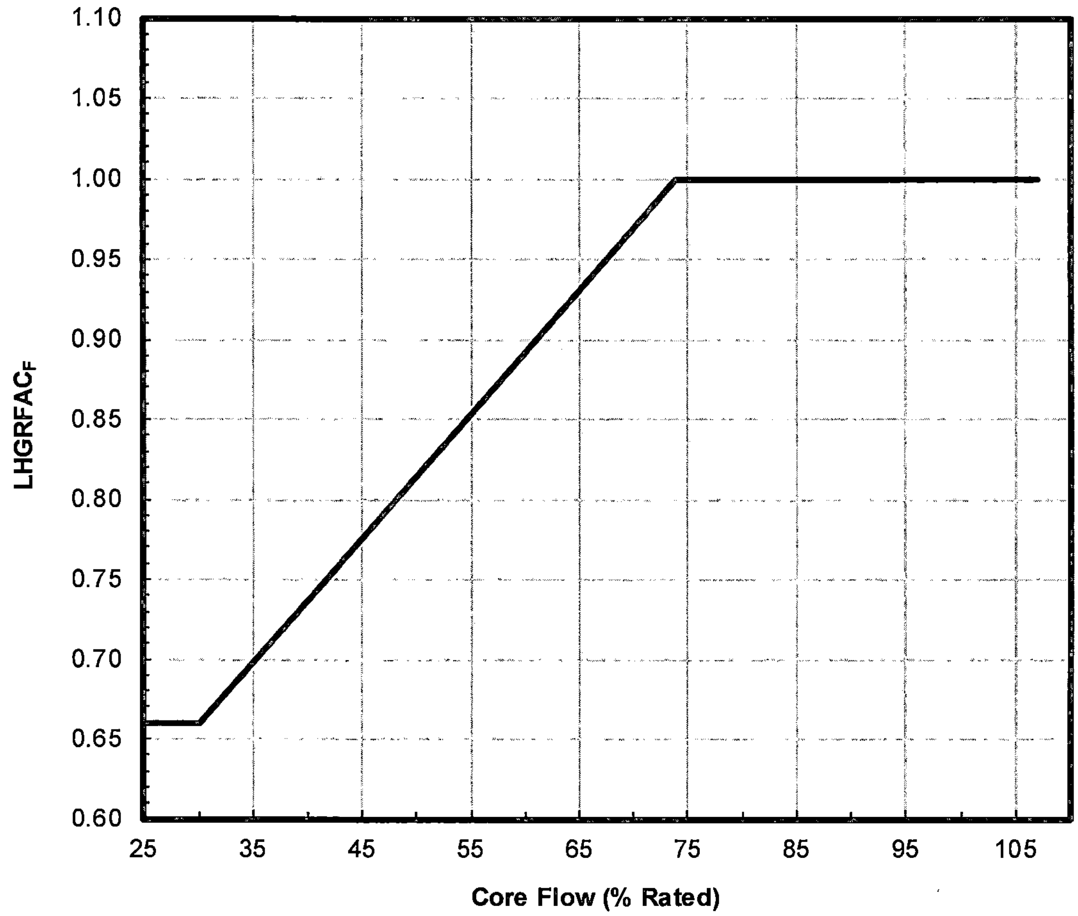
(*Independent of other EOOS conditions*)



Core Flow (% Rated)	LHGRFAC$_F$
0.0	0.92
30.0	0.92
46.4	1.00
107.0	1.00

Figure 3.7 LHGRFAC$_F$ for ATRIUM-10 Fuel
(*Values bound all EOOS conditions*)

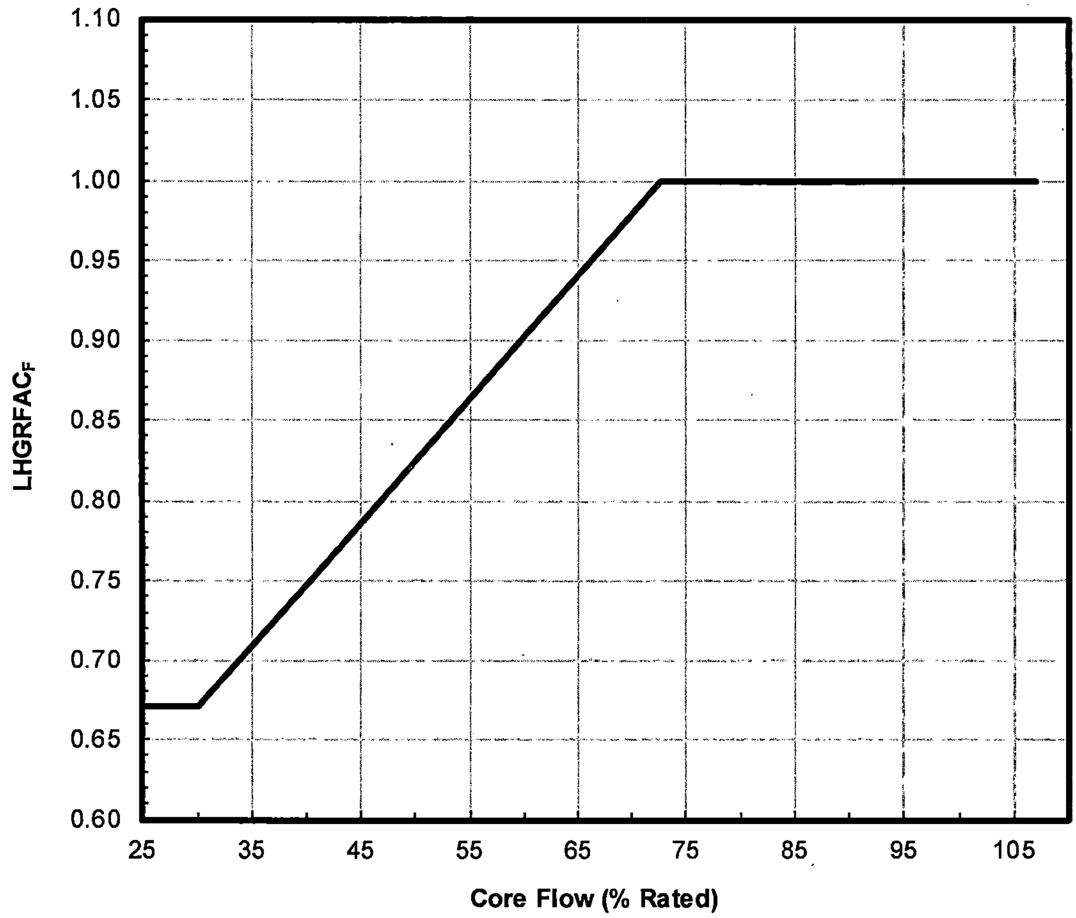
(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)



Core Flow (% Rated)	$LHGRFAC_F$
0.0	0.66
30.0	0.66
73.9	1
107.0	1

Figure 3.8 $LHGRFAC_F$ for ATRIUM-10XM Fuel
(*Values bound all EOOS conditions*)

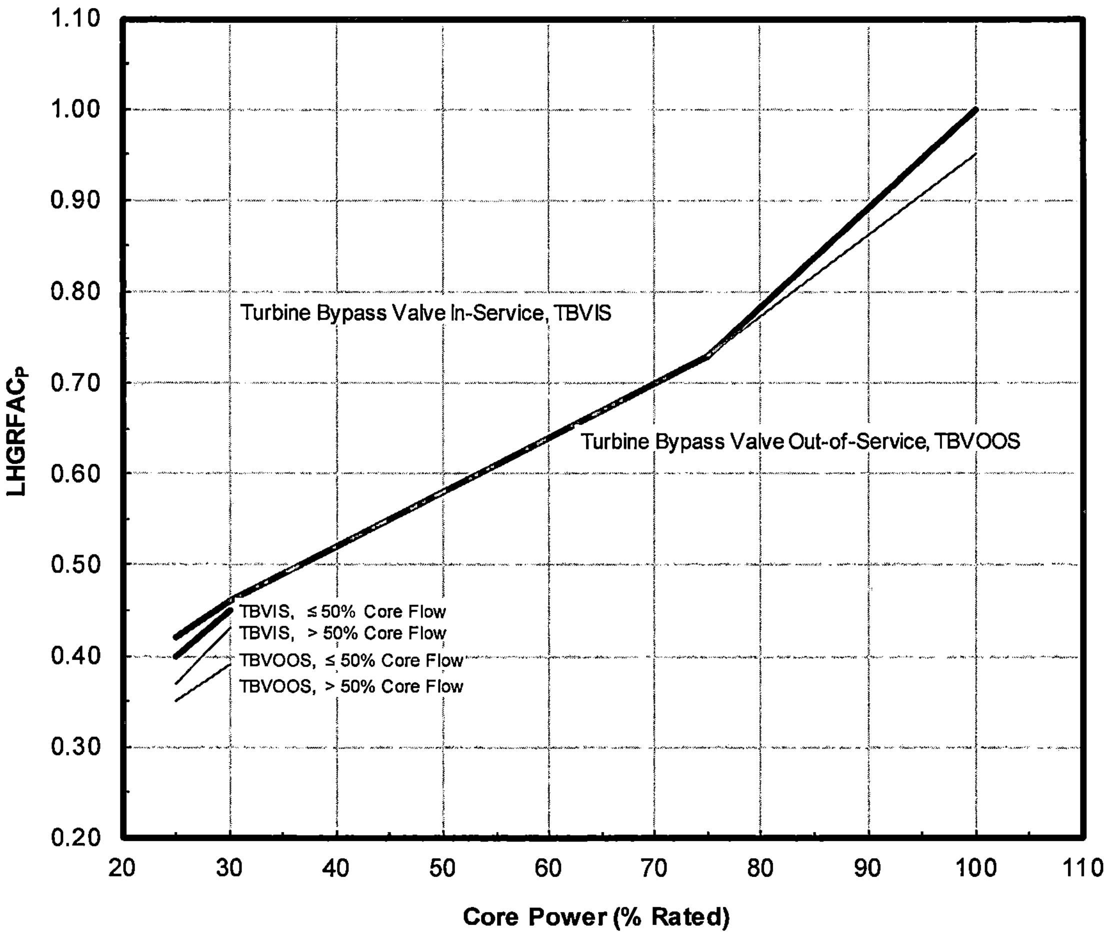
(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)



Core Flow (% Rated)	LHGRFAC$_F$
0.0	0.67
30.0	0.67
72.7	1
107.0	1

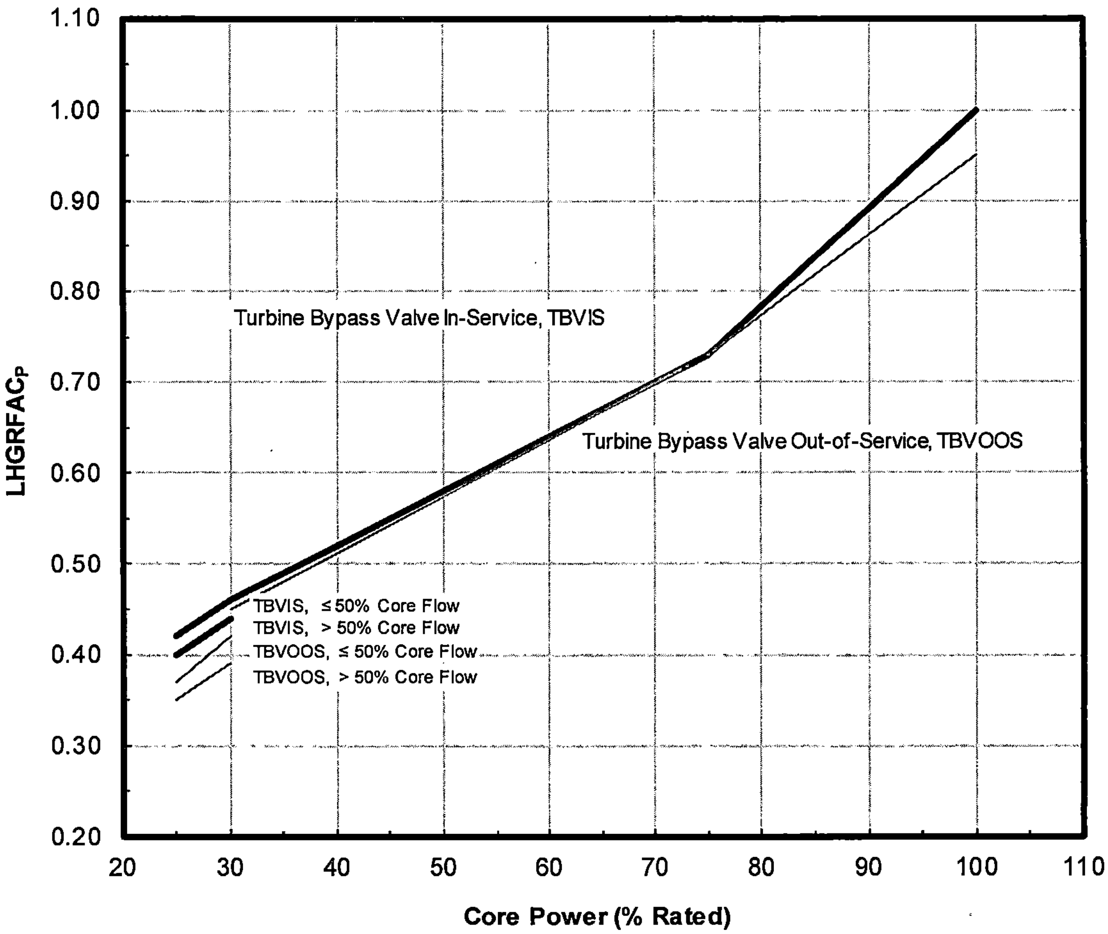
Figure 3.9 LHGRFAC$_F$ for ATRIUM-11 Fuel
(*Values bound all EOOS conditions*)

(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)



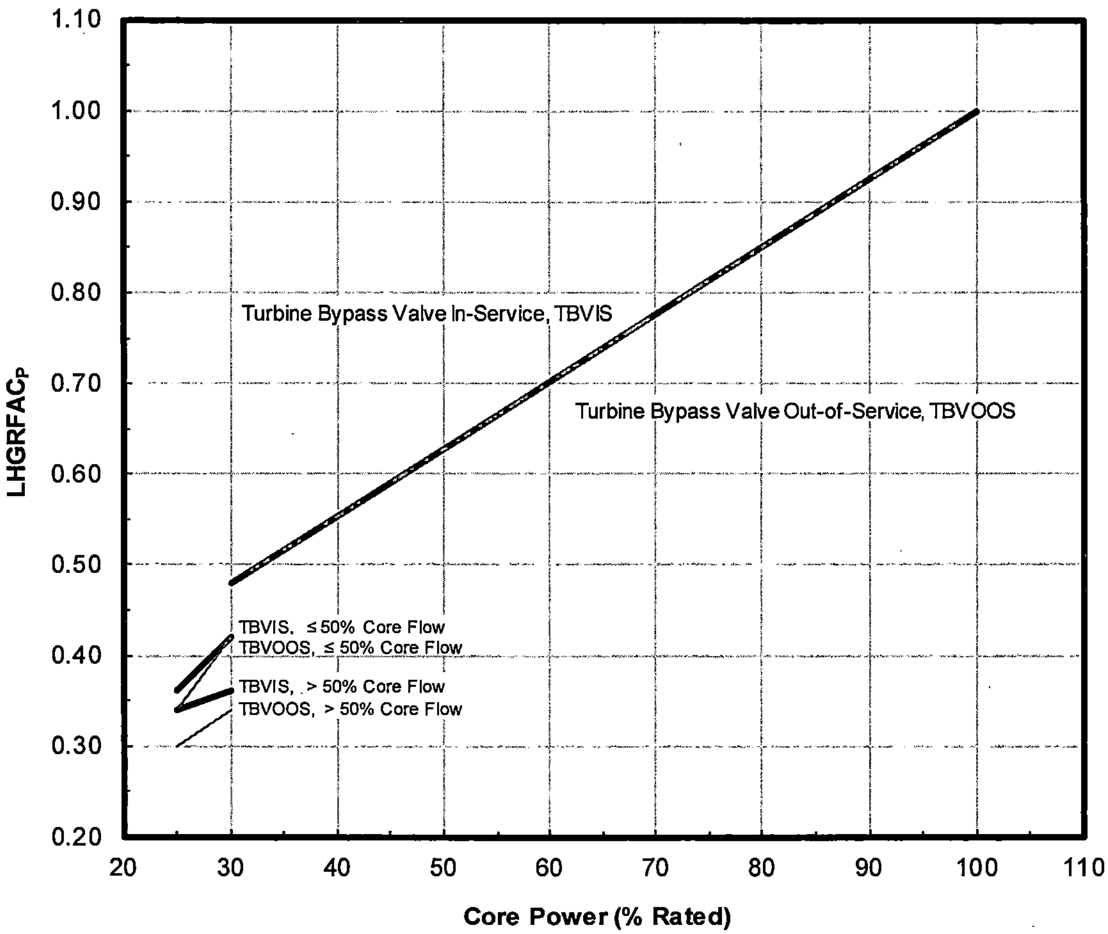
Turbine Bypass In-Service		*Turbine Bypass Out-of-Service*	
Core Power	**LHGRFAC$_P$**	**Core Power**	**LHGRFAC$_P$**
(% Rated)		**(% Rated)**	
100.0	1.00	100.0	0.95
75.0	0.73	75.0	0.73
30.0	0.46	30.0	0.46
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.45	30.0	0.39
25.0	0.40	25.0	0.35
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.46	30.0	0.43
25.0	0.42	25.0	0.37

Figure 3.10 Startup Operation LHGRFAC$_P$ for ATRIUM-10 Fuel:
Table 3.1 Temperature Range 1
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



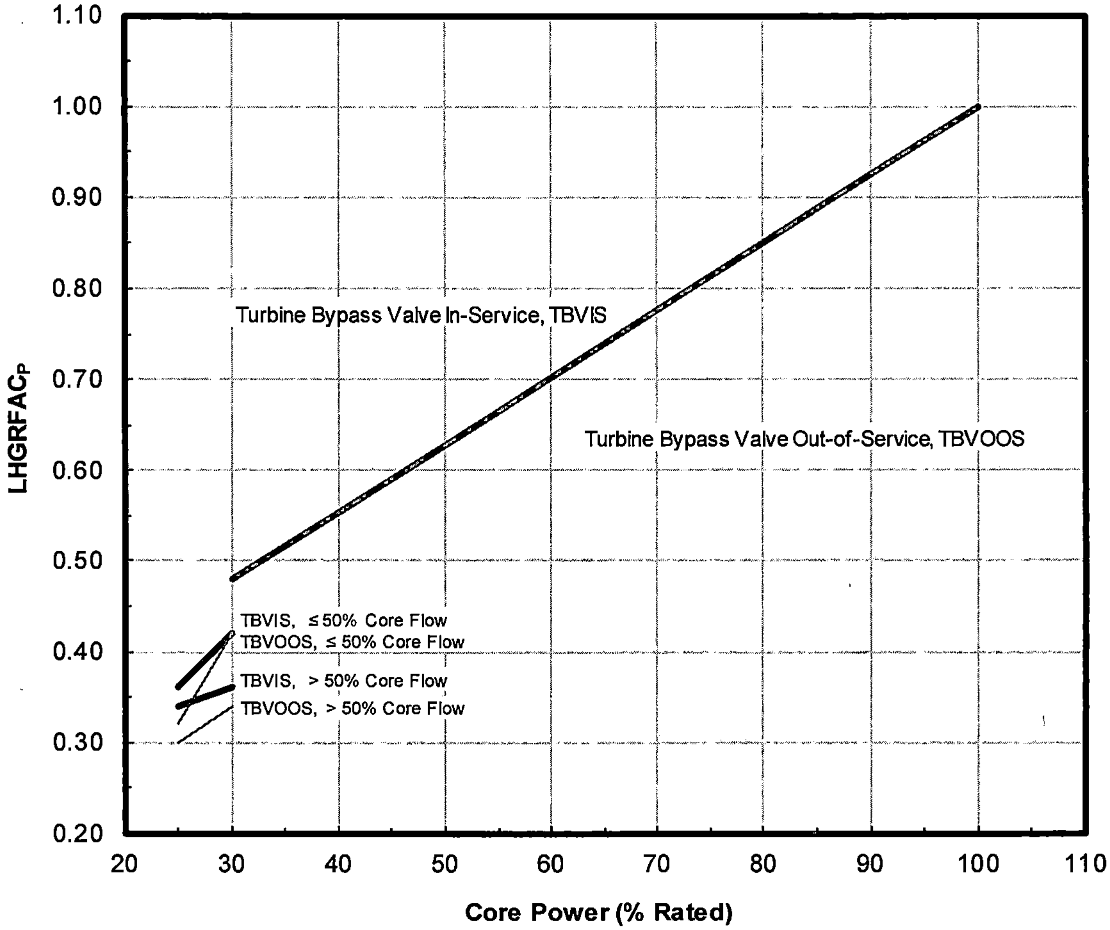
Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$	*Turbine Bypass Out-of-Service* Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00	100.0	0.95
75.0	0.73	75.0	0.73
30.0	0.46	30.0	0.45
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.44	30.0	0.39
25.0	0.40	25.0	0.35
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.46	30.0	0.42
25.0	0.42	25.0	0.37

Figure 3.11 Startup Operation LHGRFAC$_P$ for ATRIUM-10 Fuel: Table 3.1 Temperature Range 2
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



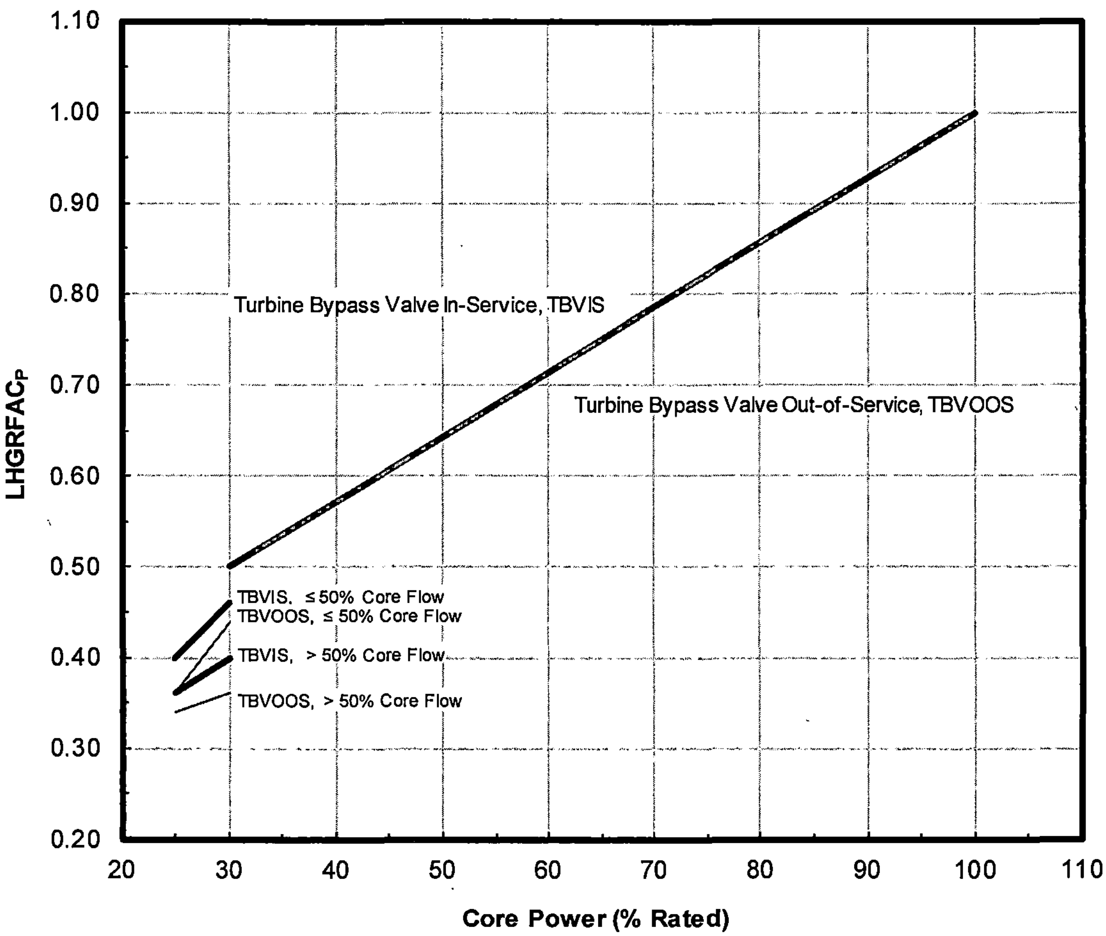
Turbine Bypass In-Service		*Turbine Bypass Out-of-Service*	
Core Power (% Rated)	**$LHGRFAC_P$**	**Core Power (% Rated)**	**$LHGRFAC_P$**
100.0	1.00	100.0	1.00
30.0	0.48	30.0	0.48
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.36	30.0	0.34
25.0	0.34	25.0	0.30
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.42	30.0	0.42
25.0	0.36	25.0	0.34

Figure 3.12 Startup Operation $LHGRFAC_P$ for ATRIUM-10XM
Fuel: Table 3.1 Temperature Range 1
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



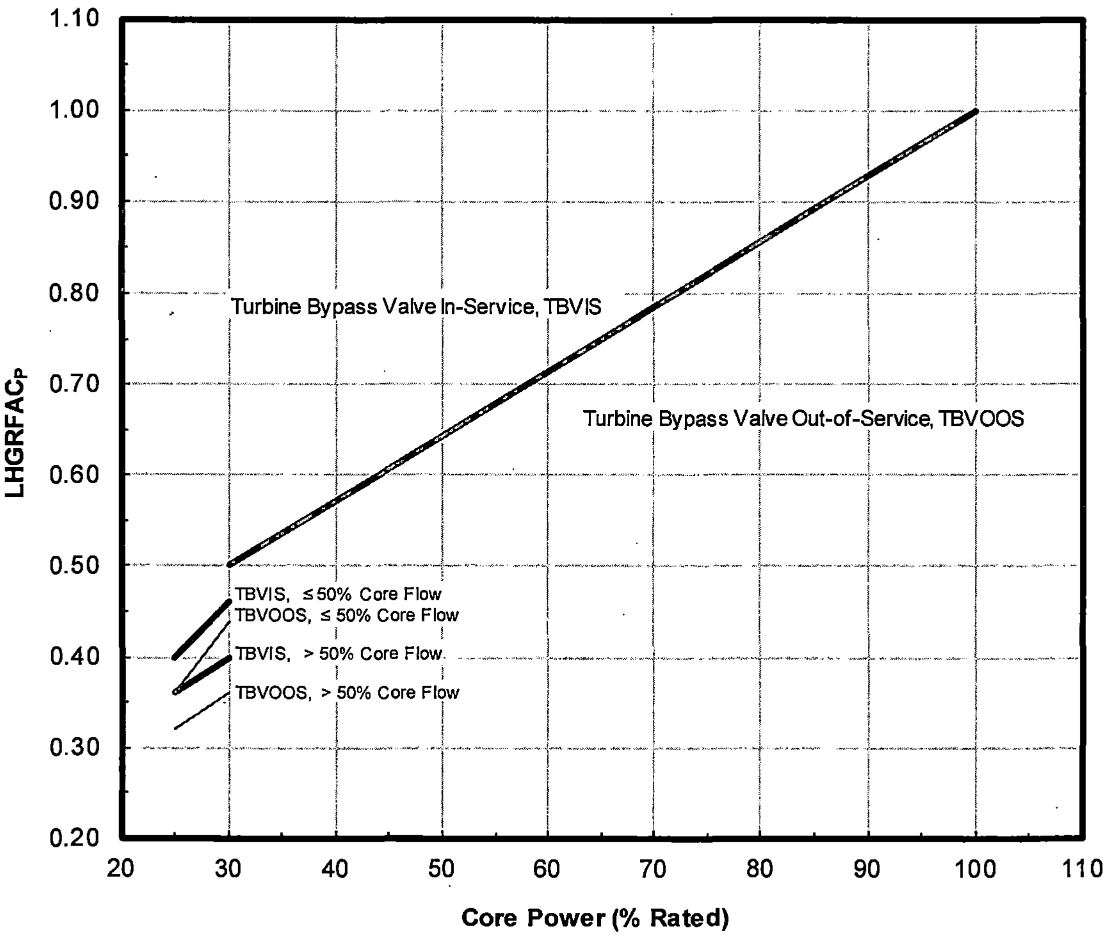
Turbine Bypass In-Service		*Turbine Bypass Out-of-Service*	
Core Power	**LHGRFAC$_P$**	**Core Power**	**LHGRFAC$_P$**
(% Rated)		**(% Rated)**	
100.0	1.00	100.0	1.00
30.0	0.48	30.0	0.48
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.36	30.0	0.34
25.0	0.34	25.0	0.30
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.42	30.0	0.42
25.0	0.36	25.0	0.32

Figure 3.13 Startup Operation LHGRFAC$_P$ for ATRIUM-10XM
Fuel: Table 3.1 Temperature Range 2
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



Turbine Bypass In-Service		*Turbine Bypass Out-of-Service*	
Core Power (% Rated)	**$LHGRFAC_P$**	**Core Power (% Rated)**	**$LHGRFAC_P$**
100.0	1.00	100.0	1.00
30.0	0.50	30.0	0.50
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.40	30.0	0.36
25.0	0.36	25.0	0.34
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.46	30.0	0.44
25.0	0.40	25.0	0.36

Figure 3.14 Startup Operation $LHGRFAC_P$ for ATRIUM-11 Fuel:
Table 3.1 Temperature Range 1
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



Turbine Bypass In-Service Core Power (% Rated)	$LHGRFAC_P$	*Turbine Bypass Out-of-Service* Core Power (% Rated)	$LHGRFAC_P$
100.0	1.00	100.0	1.00
30.0	0.50	30.0	0.50
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.40	30.0	0.36
25.0	0.36	25.0	0.32
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.46	30.0	0.44
25.0	0.40	25.0	0.36

Figure 3.15 Startup Operation $LHGRFAC_P$ for ATRIUM-11 Fuel:
Table 3.1 Temperature Range 2
(no Feedwater heating during startup)
(Limits valid at and below 50% power)

4 OLMCPR Limits

(Technical Specification 3.2.2, 3.3.4.1, & 3.7.5)

OLMCPR is calculated to be the most limiting of the flow or power dependent values

$$\text{OLMCPR limit} = \text{MAX}\ (\ MCPR_F\ ,\ MCPR_P\ )$$

where:

$MCPR_F$	core flow-dependent MCPR limit
$MCPR_P$	power-dependent MCPR limit

4.1 Flow Dependent MCPR Limit: $MCPR_F$

$MCPR_F$ limits are dependent upon core flow (% of Rated), and the max core flow limit, (Rated or Increased Core Flow, ICF). $MCPR_F$ limits are shown in Figure 4.1, per Reference 1. Limits are valid for all EOOS combinations. No adjustment is required for SLO conditions.

4.2 Power Dependent MCPR Limit: $MCPR_P$

$MCPR_P$ limits are dependent upon:

- Core Power Level (% of Rated)
- Technical Specification Scram Speed (TSSS), Nominal Scram Speed (NSS), or Optimum Scram Speed (OSS)
- Cycle Operating Exposure (NEOC, EOC, and CD - as defined in this section)
- Equipment Out-Of-Service Options
- Two or Single recirculation Loop Operation (TLO vs. SLO)

The $MCPR_P$ limits are provided in Table 4.2 through Table 4.4, where each table contains the limits for all fuel types and EOOS options (for a specified scram speed and exposure range). The CMSS determines $MCPR_P$ limits, from these tables, based on linear interpolation between the specified powers.

4.2.1 <u>Startup without Feedwater Heaters</u>

There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. Additional power dependent limits are shown in Table 4.5 through Table 4.8 based on temperature conditions identified in Table 3.1.

4.2.2 Scram Speed Dependent Limits (TSSS vs. NSS vs. OSS)

$MCPR_P$ limits are provided for three different sets of assumed scram speeds. The Technical Specification Scram Speed (TSSS) $MCPR_P$ limits are applicable at all times, as long as the scram time surveillance demonstrates the times in Technical Specification Table 3.1.4-1 are met. Both Nominal Scram Speeds (NSS) and/or Optimum Scram Speeds (OSS) may be used, as long as the scram time surveillance demonstrates Table 4.1 times are applicable.*†

Table 4.1 Nominal Scram Time Basis

Notch Position (index)	Nominal Scram Timing (seconds)	Optimum Scram Timing (seconds)
46	0.420	0.380
36	0.980	0.875
26	1.600	1.465
6	2.900	2.900

In demonstrating compliance with the NSS and/or OSS scram time basis, surveillance requirements from Technical Specification 3.1.4 apply; accepting the definition of SLOW rods should conform to scram speeds shown in Table 4.1. If conformance is not demonstrated, TSSS based $MCPR_P$ limits are applied.

On initial cycle startup, TSSS limits are used until the successful completion of scram timing confirms NSS and/or OSS based limits are applicable.

4.2.3 Exposure Dependent Limits

Exposures are tracked on a Core Average Exposure basis (CAVEX, not Cycle Exposure). Higher exposure $MCPR_P$ limits are always more limiting and may be used for any Core Average Exposure up to the ending exposure. Per Reference 1, $MCPR_P$ limits are provided for the following exposure ranges:

BOC to NEOC	NEOC corresponds to	**30,910.3 MWd / MTU**
BOC to EOCLB	EOCLB corresponds to	**33,210.3 MWd / MTU**
BOC to End of Coast	End of Coast	**34,624.4 MWd / MTU**

NEOC refers to a Near EOC exposure point.

* Reference 1 analysis results are based on information identified in Reference 4.

† Drop out times consistent with method used to perform actual timing measurements (i.e., including pickup/dropout effects).

The EOCLB exposure point is not the true End-Of-Cycle exposure. Instead it corresponds to a licensing exposure window exceeding expected end-of-full-power-life.

The End of Coast exposure point represents a licensing exposure point exceeding the expected end-of-cycle exposure including cycle extension options.

4.2.4 Equipment Out-Of-Service (EOOS) Options

EOOS options* covered by $MCPR_P$ limits are given by the following:

In-Service	All equipment In-Service
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
RPTOOS+TBVOOS	Combined RPTOOS and TBVOOS
PLUOOS	Power Load Unbalance Out-Of-Service
PLUOOS+RPTOOS	Combined PLUOOS and RPTOOS
PLUOOS+TBVOOS	Combined PLUOOS and TBVOOS
PLUOOS+TBVOOS+RPTOOS	Combined PLUOOS, RPTOOS, and TBVOOS
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service (or Final Feedwater Temperature Reduction)
RCPOOS	One Recirculation Pump Out-Of-Service

For exposure ranges up to NEOC and EOCLB, additional combinations of $MCPR_P$ limits are also provided including FHOOS. The coast down exposure range assumes application of FFWTR. FHOOS based $MCPR_P$ limits for the coast down exposure are redundant because the temperature setdown assumption is identical with FFWTR.

4.2.5 Single-Loop-Operation (SLO) Limits

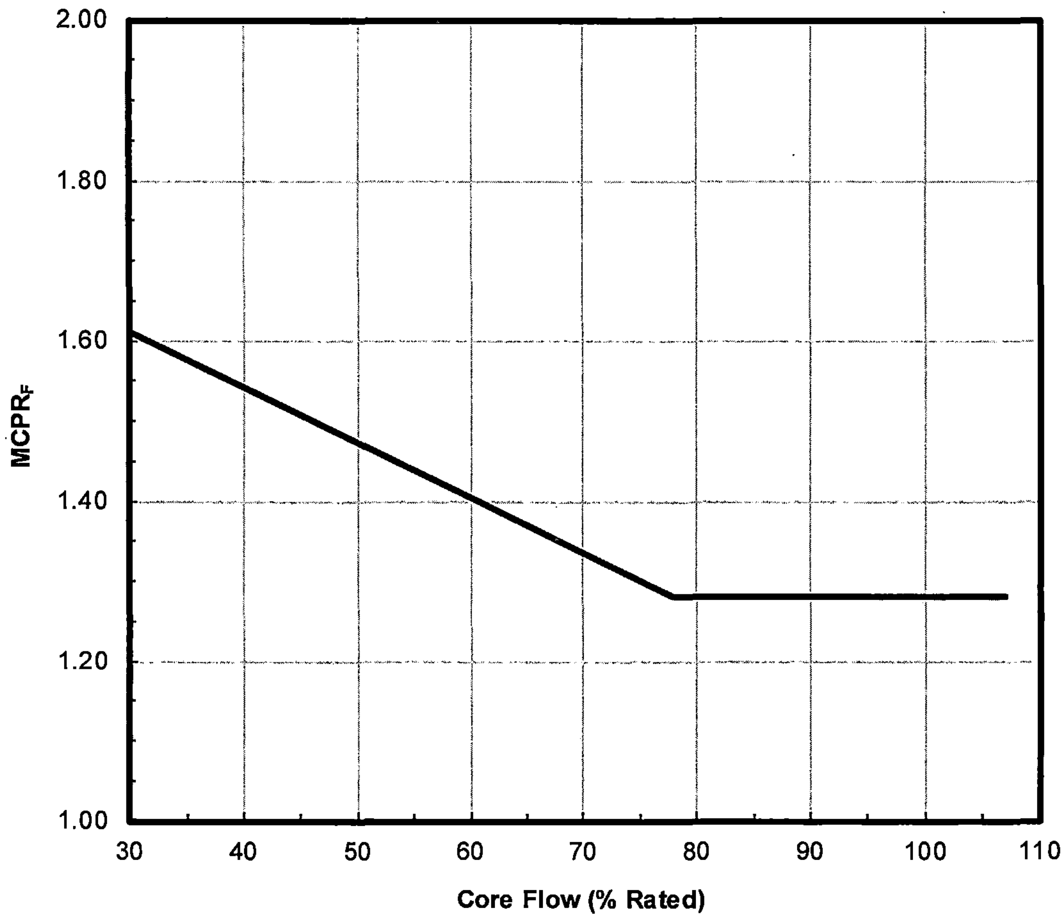
When operating in RCPOOS conditions, MCPRp limits are constructed differently from the normal operating RCP conditions. The limiting event for RCPOOS is a pump seizure scenario, which sets the upper bound for allowed core power and flow†. This event is not impacted by scram time assumptions. Specific $MCPR_P$ limits are shown in Table 4.9.

4.2.6 Below Pbypass Limits

Below Pbypass (30% rated power), $MCPR_P$ limits depend upon core flow. One set of $MCPR_P$ limits applies for core flow above 50% of rated; a second set applies if the core flow is less than or equal to 50% rated.

* All equipment service conditions assume 1 SRVOOS.

† RCPOOS limits are only valid up to 50% rated core power, 50% rated core flow, and an active recirculation drive flow of 17.73 Mlb_m/hr.



Core Flow (% Rated)	$MCPR_F$
30.0	1.61
78.0	1.28
107.0	1.28

Figure 4.1 $MCPR_F$ for All Fuel Types
(Values bound all EOOS conditions)

(107.0% maximum core flow line is used to support 105% rated flow operation, ICF)

Table 4.2 MCPR$_P$ Limits for All Fuel Types: Optimum Scram Time Basis*

		ATRIUM-10			ATRIUM-10XM			ATRIUM-11		
Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
Base Case	100	1.43	1.46	1.49	1.40	1.41	1.44	1.39	1.41	1.43
	75	1.59	1.60	1.63	1.53	1.54	1.56	1.52	1.53	1.56
	65	1.68	1.69	1.75	1.61	1.61	1.66	1.61	1.62	1.69
	50	1.86	1.87	---	1.76	1.76	---	1.83	1.84	---
	50	1.93	1.94	1.96	1.81	1.81	1.85	1.85	1.86	1.96
	40	2.14	2.15	2.27	2.00	2.00	2.10	2.10	2.11	2.24
	30	2.51	2.52	2.68	2.28	2.28	2.42	2.44	2.45	2.62
	30 at > 50%F	2.68	2.69	2.81	2.62	2.62	2.73	2.64	2.65	2.77
	25 at > 50%F	2.96	2.97	3.11	2.89	2.89	3.03	2.91	2.92	3.07
	30 at ≤ 50%F	2.61	2.62	2.71	2.52	2.52	2.62	2.59	2.60	2.72
	25 at ≤ 50%F	2.80	2.81	2.94	2.76	2.76	2.90	2.84	2.85	3.01
FHOOS	100	1.47	1.49	---	1.43	1.44	---	1.41	1.43	---
	75	1.62	1.63	---	1.56	1.56	---	1.55	1.56	---
	65	1.74	1.75	---	1.66	1.66	---	1.68	1.69	---
	50	--	--	---	--	--	---	--	--	---
	50	1.95	1.96	---	1.85	1.85	---	1.95	1.96	---
	40	2.26	2.27	---	2.10	2.10	---	2.23	2.24	---
	30	2.67	2.68	---	2.42	2.42	---	2.61	2.62	---
	30 at > 50%F	2.80	2.81	---	2.73	2.73	---	2.76	2.77	---
	25 at > 50%F	3.10	3.11	---	3.03	3.03	---	3.06	3.07	---
	30 at ≤ 50%F	2.70	2.71	---	2.62	2.62	---	2.71	2.72	---
	25 at ≤ 50%F	2.93	2.94	---	2.90	2.90	---	3.00	3.01	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR/FHOOS is supported for the BOC to End of Coast limits.

Table 4.3 MCPRP Limits for All Fuel Types: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
Base Case	100	1.45	1.47	1.50	1.42	1.43	1.45	1.40	1.42	1.44
	75	1.60	1.61	1.65	1.55	1.55	1.57	1.53	1.54	1.59
	65	1.70	1.71	1.76	1.63	1.63	1.67	1.65	1.66	1.74
	50	1.91	1.92	---	1.79	1.79	---	---	---	---
	50	1.94	1.95	2.02	1.81	1.82	1.88	1.88	1.89	2.01
	40	2.20	2.21	2.32	2.03	2.03	2.13	2.16	2.17	2.30
	30	2.58	2.59	2.75	2.33	2.33	2.48	2.50	2.51	2.69
	30 at > 50%F	2.68	2.69	2.81	2.62	2.62	2.73	2.64	2.65	2.77
	25 at > 50%F	2.96	2.97	3.11	2.89	2.89	3.03	2.91	2.92	3.07
	30 at ≤ 50%F	2.61	2.62	2.75	2.52	2.52	2.62	2.59	2.60	2.72
	25 at ≤ 50%F	2.80	2.81	2.94	2.76	2.76	2.90	2.84	2.85	3.01
TBVOOS	100	1.50	1.52	1.54	1.45	1.46	1.48	1.44	1.46	1.47
	75	1.65	1.67	1.69	1.58	1.58	1.60	1.57	1.58	1.61
	65	1.75	1.76	1.82	1.66	1.66	1.71	1.66	1.67	1.74
	50	1.91	1.92	---	1.80	1.80	---	---	---	---
	50	1.94	1.95	2.02	1.81	1.82	1.88	1.88	1.89	2.01
	40	2.20	2.21	2.32	2.03	2.03	2.13	2.16	2.17	2.30
	30	2.58	2.59	2.75	2.33	2.33	2.48	2.50	2.51	2.69
	30 at > 50%F	3.28	3.29	3.43	3.11	3.11	3.24	3.25	3.26	3.41
	25 at > 50%F	3.70	3.71	3.83	3.54	3.54	3.66	3.64	3.65	3.80
	30 at ≤ 50%F	3.03	3.04	3.19	2.86	2.86	3.00	3.05	3.06	3.21
	25 at ≤ 50%F	3.47	3.48	3.69	3.33	3.33	3.52	3.46	3.47	3.67
FHOOS	100	1.49	1.50	---	1.44	1.45	---	1.43	1.44	---
	75	1.64	1.65	---	1.57	1.57	---	1.58	1.59	---
	65	1.75	1.76	---	1.67	1.67	---	1.73	1.74	---
	50	---	---	---	---	---	---	---	---	---
	50	2.01	2.02	---	1.88	1.88	---	2.00	2.01	---
	40	2.31	2.32	---	2.13	2.13	---	2.29	2.30	---
	30	2.74	2.75	---	2.48	2.48	---	2.68	2.69	---
	30 at > 50%F	2.80	2.81	---	2.73	2.73	---	2.76	2.77	---
	25 at > 50%F	3.10	3.11	---	3.03	3.03	---	3.06	3.07	---
	30 at ≤ 50%F	2.74	2.75	---	2.62	2.62	---	2.71	2.72	---
	25 at ≤ 50%F	2.93	2.94	---	2.90	2.90	---	3.00	3.01	---
PLUOOS	100	1.45	1.47	1.50	1.42	1.43	1.45	1.40	1.42	1.44
	75	1.60	1.61	1.65	1.55	1.55	1.57	1.53	1.54	1.59
	65	1.85	1.87	1.87	1.73	1.74	1.75	1.76	1.78	1.79
	50	---	---	---	---	---	---	---	---	---
	50	1.94	1.95	2.02	1.81	1.82	1.88	1.88	1.89	2.01
	40	2.20	2.21	2.32	2.03	2.03	2.13	2.16	2.17	2.30
	30	2.58	2.59	2.75	2.33	2.33	2.48	2.50	2.51	2.69
	30 at > 50%F	2.68	2.69	2.81	2.62	2.62	2.73	2.64	2.65	2.77
	25 at > 50%F	2.96	2.97	3.11	2.89	2.89	3.03	2.91	2.92	3.07
	30 at ≤ 50%F	2.61	2.62	2.75	2.52	2.52	2.62	2.59	2.60	2.72
	25 at ≤ 50%F	2.80	2.81	2.94	2.76	2.76	2.90	2.84	2.85	3.01

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.3 MCPR$_P$ Limits for All Fuel Types: Nominal Scram Time Basis (*continued*)*

Operating Condition	Power (% of rated)	ATRIUM-10			ATRIUM-10XM			ATRIUM-11		
		BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVOOS FHOOS	100	1.53	1.54	---	1.47	1.48	---	1.46	1.47	---
	75	1.68	1.69	---	1.60	1.60	---	1.60	1.61	---
	65	1.81	1.82	---	1.71	1.71	---	1.73	1.74	---
	50	---	---	---	---	---	---	---	---	---
	50	2.01	2.02	---	1.88	1.88	---	2.00	2.01	---
	40	2.31	2.32	---	2.13	2.13	---	2.29	2.30	---
	30	2.74	2.75	---	2.48	2.48	---	2.68	2.69	---
	30 at > 50%F	3.42	3.43	---	3.24	3.24	---	3.40	3.41	---
	25 at > 50%F	3.82	3.83	---	3.66	3.66	---	3.79	3.80	---
	30 at ≤ 50%F	3.18	3.19	---	3.00	3.00	---	3.20	3.21	---
	25 at ≤ 50%F	3.68	3.69	---	3.52	3.52	---	3.66	3.67	---
TBVOOS PLUOOS	100	1.50	1.52	1.54	1.45	1.46	1.48	1.44	1.46	1.47
	75	1.65	1.67	1.69	1.58	1.58	1.60	1.57	1.58	1.61
	65	1.85	1.87	1.87	1.73	1.74	1.75	1.76	1.78	1.79
	50	---	---	---	---	---	---	---	---	---
	50	1.94	1.95	2.02	1.81	1.82	1.88	1.88	1.89	2.01
	40	2.20	2.21	2.32	2.03	2.03	2.13	2.16	2.17	2.30
	30	2.58	2.59	2.75	2.33	2.33	2.48	2.50	2.51	2.69
	30 at > 50%F	3.28	3.29	3.43	3.11	3.11	3.24	3.25	3.26	3.41
	25 at > 50%F	3.70	3.71	3.83	3.54	3.54	3.66	3.64	3.65	3.80
	30 at ≤ 50%F	3.03	3.04	3.19	2.86	2.86	3.00	3.05	3.06	3.21
	25 at ≤ 50%F	3.47	3.48	3.69	3.33	3.33	3.52	3.46	3.47	3.67
FHOOS PLUOOS	100	1.49	1.50	---	1.44	1.45	---	1.43	1.44	---
	75	1.64	1.65	---	1.57	1.57	---	1.58	1.59	---
	65	1.85	1.87	---	1.73	1.74	---	1.76	1.78	---
	50	---	---	---	---	---	---	---	---	---
	50	2.01	2.02	---	1.88	1.88	---	2.00	2.01	---
	40	2.31	2.32	---	2.13	2.13	---	2.29	2.30	---
	30	2.74	2.75	---	2.48	2.48	---	2.68	2.69	---
	30 at > 50%F	2.80	2.81	---	2.73	2.73	---	2.76	2.77	---
	25 at > 50%F	3.10	3.11	---	3.03	3.03	---	3.06	3.07	---
	30 at ≤ 50%F	2.74	2.75	---	2.62	2.62	---	2.71	2.72	---
	25 at ≤ 50%F	2.93	2.94	---	2.90	2.90	---	3.00	3.01	---
TBVOOS FHOOS PLUOOS	100	1.53	1.54	---	1.47	1.48	---	1.46	1.47	---
	75	1.68	1.69	---	1.60	1.60	---	1.60	1.61	---
	65	1.85	1.87	---	1.73	1.74	---	1.76	1.78	---
	50	---	---	---	---	---	---	---	---	---
	50	2.01	2.02	---	1.88	1.88	---	2.00	2.01	---
	40	2.31	2.32	---	2.13	2.13	---	2.29	2.30	---
	30	2.74	2.75	---	2.48	2.48	---	2.68	2.69	---
	30 at > 50%F	3.42	3.43	---	3.24	3.24	---	3.40	3.41	---
	25 at > 50%F	3.82	3.83	---	3.66	3.66	---	3.79	3.80	---
	30 at ≤ 50%F	3.18	3.19	---	3.00	3.00	---	3.20	3.21	---
	25 at ≤ 50%F	3.68	3.69	---	3.52	3.52	---	3.66	3.67	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.4 MCPRP Limits for All Fuel Types: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
Base Case	100	1.47	1.48	1.51	1.43	1.44	1.46	1.42	1.43	1.45
	75	1.61	1.62	1.66	1.56	1.56	1.58	1.56	1.57	1.63
	65	1.71	1.72	1.79	1.63	1.63	1.69	1.69	1.70	1.78
	50	--	---	---	--	1.82	---	---	---	---
	50	1.96	1.97	2.07	1.82	1.83	1.91	1.93	1.94	2.06
	40	2.25	2.26	2.38	2.06	2.06	2.16	2.21	2.22	2.35
	30	2.64	2.65	2.82	2.38	2.38	2.53	2.56	2.57	2.75
	30 at > 50%F	2.68	2.69	2.82	2.62	2.62	2.73	2.64	2.65	2.77
	25 at > 50%F	2.96	2.97	3.11	2.89	2.89	3.03	2.91	2.92	3.07
	30 at ≤ 50%F	2.64	2.65	2.82	2.52	2.52	2.62	2.59	2.60	2.75
	25 at ≤ 50%F	2.80	2.81	2.94	2.76	2.76	2.90	2.84	2.85	3.01
TBVOOS	100	1.52	1.55	1.56	1.47	1.47	1.49	1.46	1.47	1.49
	75	1.67	1.68	1.71	1.59	1.59	1.61	1.58	1.59	1.64
	65	1.76	1.77	1.83	1.67	1.67	1.72	1.70	1.71	1.79
	50	--	---	---	--	---	---	---	---	---
	50	1.97	1.98	2.08	1.83	1.83	1.92	1.93	1.94	2.06
	40	2.26	2.27	2.38	2.07	2.07	2.17	2.21	2.22	2.35
	30	2.64	2.65	2.82	2.38	2.38	2.53	2.56	2.57	2.75
	30 at > 50%F	3.28	3.29	3.43	3.11	3.11	3.24	3.25	3.26	3.41
	25 at > 50%F	3.70	3.71	3.83	3.54	3.54	3.66	3.64	3.65	3.80
	30 at ≤ 50%F	3.03	3.04	3.19	2.86	2.86	3.00	3.05	3.06	3.21
	25 at ≤ 50%F	3.47	3.48	3.69	3.33	3.33	3.52	3.46	3.47	3.67
FHOOS	100	1.50	1.51	---	1.46	1.46	---	1.44	1.45	---
	75	1.65	1.66	---	1.58	1.58	---	1.62	1.63	---
	65	1.78	1.79	---	1.69	1.69	---	1.77	1.78	---
	50	---	---	---	---	---	---	---	---	---
	50	2.06	2.07	---	1.91	1.91	---	2.05	2.06	---
	40	2.37	2.38	---	2.16	2.16	---	2.34	2.35	---
	30	2.81	2.82	---	2.53	2.53	---	2.74	2.75	---
	30 at > 50%F	2.81	2.82	---	2.73	2.73	---	2.76	2.77	---
	25 at > 50%F	3.10	3.11	---	3.03	3.03	---	3.06	3.07	---
	30 at ≤ 50%F	2.81	2.82	---	2.62	2.62	---	2.74	2.75	---
	25 at ≤ 50%F	2.93	2.94	---	2.90	2.90	---	3.00	3.01	---
PLUOOS	100	1.47	1.48	1.51	1.43	1.44	1.46	1.42	1.43	1.45
	75	1.61	1.62	1.66	1.56	1.56	1.58	1.56	1.57	1.63
	65	1.87	1.89	1.89	1.75	1.76	1.78	1.78	1.81	1.81
	50	---	---	---	---	---	---	---	---	---
	50	1.96	1.97	2.07	1.82	1.83	1.91	1.93	1.94	2.06
	40	2.25	2.26	2.38	2.06	2.06	2.16	2.21	2.22	2.35
	30	2.64	2.65	2.82	2.38	2.38	2.53	2.56	2.57	2.75
	30 at > 50%F	2.68	2.69	2.82	2.62	2.62	2.73	2.64	2.65	2.77
	25 at > 50%F	2.96	2.97	3.11	2.89	2.89	3.03	2.91	2.92	3.07
	30 at ≤ 50%F	2.64	2.65	2.82	2.52	2.52	2.62	2.59	2.60	2.75
	25 at ≤ 50%F	2.80	2.81	2.94	2.76	2.76	2.90	2.84	2.85	3.01

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.4 MCPRp Limits for All Fuel Types: Technical Specification Scram Time Basis (*continued*)*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
TBVOOS FHOOS	100	1.54	1.56	---	1.49	1.49	---	1.48	1.49	---
	75	1.70	1.71	---	1.61	1.61	---	1.63	1.64	---
	65	1.82	1.83	---	1.72	1.72	---	1.78	1.79	---
	50	---	---	---	---	---	---	---	---	---
	50	2.07	2.08	---	1.92	1.92	---	2.05	2.06	---
	40	2.37	2.38	---	2.17	2.17	---	2.34	2.35	---
	30	2.81	2.82	---	2.53	2.53	---	2.74	2.75	---
	30 at > 50%F	3.42	3.43	---	3.24	3.24	---	3.40	3.41	---
	25 at > 50%F	3.82	3.83	---	3.66	3.66	---	3.79	3.80	---
	30 at ≤ 50%F	3.18	3.19	---	3.00	3.00	---	3.20	3.21	---
	25 at ≤ 50%F	3.68	3.69	---	3.52	3.52	---	3.66	3.67	---
TBVOOS PLUOOS	100	1.52	1.55	1.56	1.47	1.47	1.49	1.46	1.47	1.49
	75	1.67	1.68	1.71	1.59	1.59	1.61	1.58	1.59	1.64
	65	1.87	1.89	1.89	1.75	1.76	1.78	1.78	1.81	1.81
	50	---	---	---	---	---	---	---	---	---
	50	1.97	1.98	2.08	1.83	1.83	1.92	1.93	1.94	2.06
	40	2.26	2.27	2.38	2.07	2.07	2.17	2.21	2.22	2.35
	30	2.64	2.65	2.82	2.38	2.38	2.53	2.56	2.57	2.75
	30 at > 50%F	3.28	3.29	3.43	3.11	3.11	3.24	3.25	3.26	3.41
	25 at > 50%F	3.70	3.71	3.83	3.54	3.54	3.66	3.64	3.65	3.80
	30 at ≤ 50%F	3.03	3.04	3.19	2.86	2.86	3.00	3.05	3.06	3.21
	25 at ≤ 50%F	3.47	3.48	3.69	3.33	3.33	3.52	3.46	3.47	3.67
FHOOS PLUOOS	100	1.50	1.51	---	1.46	1.46	---	1.44	1.45	---
	75	1.65	1.66	---	1.58	1.58	---	1.62	1.63	---
	65	1.87	1.89	---	1.75	1.76	---	1.78	1.81	---
	50	---	---	---	---	---	---	---	---	---
	50	2.06	2.07	---	1.91	1.91	---	2.05	2.06	---
	40	2.37	2.38	---	2.16	2.16	---	2.34	2.35	---
	30	2.81	2.82	---	2.53	2.53	---	2.74	2.75	---
	30 at > 50%F	2.81	2.82	---	2.73	2.73	---	2.76	2.77	---
	25 at > 50%F	3.10	3.11	---	3.03	3.03	---	3.06	3.07	---
	30 at ≤ 50%F	2.81	2.82	---	2.62	2.62	---	2.74	2.75	---
	25 at ≤ 50%F	2.93	2.94	---	2.90	2.90	---	3.00	3.01	---
TBVOOS FHOOS PLUOOS	100	1.54	1.56	---	1.49	1.49	---	1.48	1.49	---
	75	1.70	1.71	---	1.61	1.61	---	1.63	1.64	---
	65	1.87	1.89	---	1.75	1.76	---	1.78	1.81	---
	50	---	---	---	---	---	---	---	---	---
	50	2.07	2.08	---	1.92	1.92	---	2.05	2.06	---
	40	2.37	2.38	---	2.17	2.17	---	2.34	2.35	---
	30	2.81	2.82	---	2.53	2.53	---	2.74	2.75	---
	30 at > 50%F	3.42	3.43	---	3.24	3.24	---	3.40	3.41	---
	25 at > 50%F	3.82	3.83	---	3.66	3.66	---	3.79	3.80	---
	30 at ≤ 50%F	3.18	3.19	---	3.00	3.00	---	3.20	3.21	---
	25 at ≤ 50%F	3.68	3.69	---	3.52	3.52	---	3.66	3.67	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.5 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 1 for All Fuel Types: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
TBVIS	100	1.49	1.50	1.50	1.44	1.45	1.45	1.43	1.44	1.44
	75	1.64	1.65	1.65	1.57	1.57	1.57	1.58	1.59	1.59
	65	1.85	1.87	1.87	1.73	1.74	1.74	1.76	1.78	1.78
	50	---	---	---	---	---	---	---	---	---
	50	2.25	2.25	2.25	2.07	2.07	2.07	2.28	2.28	2.28
	40	2.62	2.62	2.62	2.39	2.39	2.39	2.64	2.64	2.64
	30	3.12	3.12	3.12	2.80	2.80	2.80	3.07	3.07	3.07
	30 at > 50%F	3.12	3.12	3.12	3.00	3.00	3.00	3.08	3.08	3.08
	25 at > 50%F	3.46	3.46	3.46	3.33	3.33	3.33	3.44	3.44	3.44
	30 at ≤ 50%F	3.12	3.12	3.12	2.88	2.88	2.88	3.07	3.07	3.07
	25 at ≤ 50%F	3.42	3.42	3.42	3.21	3.21	3.21	3.34	3.34	3.34
TBVOOS	100	1.53	1.54	1.54	1.47	1.48	1.48	1.46	1.47	1.47
	75	1.68	1.69	1.69	1.60	1.60	1.60	1.60	1.61	1.61
	65	1.85	1.87	1.87	1.73	1.74	1.74	1.76	1.78	1.78
	50	---	---	---	---	---	---	---	---	---
	50	2.25	2.25	2.25	2.07	2.07	2.07	2.28	2.28	2.28
	40	2.62	2.62	2.62	2.39	2.39	2.39	2.64	2.64	2.64
	30	3.12	3.12	3.12	2.80	2.80	2.80	3.07	3.07	3.07
	30 at > 50%F	3.64	3.64	3.64	3.45	3.45	3.45	3.67	3.67	3.67
	25 at > 50%F	4.07	4.07	4.07	3.90	3.90	3.90	4.08	4.08	4.08
	30 at ≤ 50%F	3.46	3.46	3.46	3.26	3.26	3.26	3.46	3.46	3.46
	25 at ≤ 50%F	3.96	3.96	3.96	3.80	3.80	3.80	3.95	3.95	3.95

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.6 Startup Operation MCPR$_P$ Limits for Table 3.1 Temperature Range 2 for All Fuel Types: Nominal Scram Time Basis*

		ATRIUM-10			ATRIUM-10XM			ATRIUM-11		
Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVIS	100	1.49	1.50	1.50	1.44	1.45	1.45	1.43	1.44	1.44
	75	1.64	1.65	1.65	1.57	1.57	1.57	1.58	1.59	1.59
	65	1.85	1.87	1.87	1.73	1.74	1.74	1.76	1.78	1.78
	50	---	---	---	---	---	---	---	---	---
	50	2.27	2.27	2.27	2.09	2.09	2.09	2.30	2.30	2.30
	40	2.64	2.64	2.64	2.40	2.40	2.40	2.66	2.66	2.66
	30	3.14	3.14	3.14	2.83	2.83	2.83	3.10	3.10	3.10
	30 at > 50%F	3.17	3.17	3.17	3.01	3.01	3.01	3.10	3.10	3.10
	25 at > 50%F	3.50	3.50	3.50	3.35	3.35	3.35	3.46	3.46	3.46
	30 at ≤ 50%F	3.14	3.14	3.14	2.90	2.90	2.90	3.10	3.10	3.10
	25 at ≤ 50%F	3.44	3.44	3.44	3.23	3.23	3.23	3.37	3.37	3.37
TBVOOS	100	1.53	1.54	1.54	1.47	1.48	1.48	1.46	1.47	1.47
	75	1.68	1.69	1.69	1.60	1.60	1.60	1.60	1.61	1.61
	65	1.85	1.87	1.87	1.73	1.74	1.74	1.76	1.78	1.78
	50	---	---	---	---	---	---	---	---	---
	50	2.27	2.27	2.27	2.09	2.09	2.09	2.30	2.30	2.30
	40	2.64	2.64	2.64	2.40	2.40	2.40	2.66	2.66	2.66
	30	3.14	3.14	3.14	2.83	2.83	2.83	3.10	3.10	3.10
	30 at > 50%F	3.66	3.66	3.66	3.46	3.46	3.46	3.68	3.68	3.68
	25 at > 50%F	4.09	4.09	4.09	3.91	3.91	3.91	4.10	4.10	4.10
	30 at ≤ 50%F	3.48	3.48	3.48	3.28	3.28	3.28	3.48	3.48	3.48
	25 at ≤ 50%F	3.98	3.98	3.98	3.82	3.82	3.82	3.97	3.97	3.97

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.7 Startup Operation MCPRp Limits for Table 3.1 Temperature Range 1 for All Fuel Types: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
TBVIS	100	1.50	1.51	1.51	1.46	1.46	1.46	1.44	1.45	1.45
	75	1.65	1.66	1.66	1.58	1.58	1.58	1.62	1.63	1.63
	65	1.87	1.89	1.89	1.75	1.76	1.76	1.78	1.81	1.81
	50	---	---	---	---	---	---	---	---	---
	50	2.31	2.31	2.31	2.11	2.11	2.11	2.34	2.34	2.34
	40	2.68	2.68	2.68	2.43	2.43	2.43	2.70	2.70	2.70
	30	3.19	3.19	3.19	2.86	2.86	2.86	3.14	3.14	3.14
	30 at > 50%F	3.19	3.19	3.19	3.00	3.00	3.00	3.14	3.14	3.14
	25 at > 50%F	3.46	3.46	3.46	3.33	3.33	3.33	3.44	3.44	3.44
	30 at ≤ 50%F	3.19	3.19	3.19	2.88	2.88	2.88	3.14	3.14	3.14
	25 at ≤ 50%F	3.42	3.42	3.42	3.21	3.21	3.21	3.34	3.34	3.34
TBVOOS	100	1.54	1.56	1.56	1.49	1.49	1.49	1.48	1.49	1.49
	75	1.70	1.71	1.71	1.61	1.61	1.61	1.63	1.64	1.64
	65	1.87	1.89	1.89	1.75	1.76	1.76	1.78	1.81	1.81
	50	---	---	---	---	---	---	---	---	---
	50	2.31	2.31	2.31	2.11	2.11	2.11	2.34	2.34	2.34
	40	2.68	2.68	2.68	2.43	2.43	2.43	2.70	2.70	2.70
	30	3.19	3.19	3.19	2.86	2.86	2.86	3.14	3.14	3.14
	30 at > 50%F	3.64	3.64	3.64	3.45	3.45	3.45	3.67	3.67	3.67
	25 at > 50%F	4.07	4.07	4.07	3.90	3.90	3.90	4.08	4.08	4.08
	30 at ≤ 50%F	3.46	3.46	3.46	3.26	3.26	3.26	3.46	3.46	3.46
	25 at ≤ 50%F	3.96	3.96	3.96	3.80	3.80	3.80	3.95	3.95	3.95

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.8 Startup Operation MCPRP Limits for Table 3.1 Temperature Range 2 for All Fuel Types: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
TBVIS	100	1.50	1.51	1.51	1.46	1.46	1.46	1.44	1.45	1.45
	75	1.65	1.66	1.66	1.58	1.58	1.58	1.62	1.63	1.63
	65	1.87	1.89	1.89	1.75	1.76	1.76	1.78	1.81	1.81
	50	---	---	---	---	---	---	---	---	---
	50	2.33	2.33	2.33	2.12	2.12	2.12	2.35	2.35	2.35
	40	2.70	2.70	2.70	2.44	2.44	2.44	2.72	2.72	2.72
	30	3.22	3.22	3.22	2.89	2.89	2.89	3.17	3.17	3.17
	30 at > 50%F	3.22	3.22	3.22	3.01	3.01	3.01	3.17	3.17	3.17
	25 at > 50%F	3.50	3.50	3.50	3.35	3.35	3.35	3.46	3.46	3.46
	30 at ≤ 50%F	3.22	3.22	3.22	2.90	2.90	2.90	3.17	3.17	3.17
	25 at ≤ 50%F	3.44	3.44	3.44	3.23	3.23	3.23	3.37	3.37	3.37
TBVOOS	100	1.54	1.56	1.56	1.49	1.49	1.49	1.48	1.49	1.49
	75	1.70	1.71	1.71	1.61	1.61	1.61	1.63	1.64	1.64
	65	1.87	1.89	1.89	1.75	1.76	1.76	1.78	1.81	1.81
	50	---	---	---	---	---	---	---	---	---
	50	2.33	2.33	2.33	2.12	2.12	2.12	2.35	2.35	2.35
	40	2.70	2.70	2.70	2.44	2.44	2.44	2.72	2.72	2.72
	30	3.22	3.22	3.22	2.89	2.89	2.89	3.17	3.17	3.17
	30 at > 50%F	3.66	3.66	3.66	3.46	3.46	3.46	3.68	3.68	3.68
	25 at > 50%F	4.09	4.09	4.09	3.91	3.91	3.91	4.10	4.10	4.10
	30 at ≤ 50%F	3.48	3.48	3.48	3.28	3.28	3.28	3.48	3.48	3.48
	25 at ≤ 50%F	3.98	3.98	3.98	3.82	3.82	3.82	3.97	3.97	3.97

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.9 MCPR$_P$ Limits for All Fuel Types: Single Loop Operation for All Scram Times*

Operating Condition	Power (% of rated)	BOC to End of COAST		
		ATRIUM-10	ATRIUM-10XM	ATRIUM-11
RCPOOS FHOOS	100	2.09	2.01	2.08
	50	2.09	2.01	2.08
	40	2.40	2.18	2.37
	30	2.84	2.55	2.77
	30 at > 50%F	2.84	2.75	2.79
	25 at > 50%F	3.13	3.05	3.09
	30 at ≤ 50%F	2.84	2.64	2.77
	25 at ≤ 50%F	2.96	2.92	3.03
RCPOOS TBVOOS PLUOOS FHOOS	100	2.10	2.01	2.08
	50	2.10	2.01	2.08
	40	2.40	2.19	2.37
	30	2.84	2.55	2.77
	30 at > 50%F	3.45	3.26	3.43
	25 at > 50%F	3.85	3.68	3.82
	30 at ≤ 50%F	3.21	3.02	3.23
	25 at ≤ 50%F	3.71	3.54	3.69
RCPOOS TBVOOS FHOOS1	100	2.33	2.13	2.36
	50	2.33	2.13	2.36
	40	2.70	2.45	2.72
	30	3.21	2.88	3.16
	30 at > 50%F	3.66	3.47	3.69
	25 at > 50%F	4.09	3.92	4.10
	30 at ≤ 50%F	3.48	3.28	3.48
	25 at ≤ 50%F	3.98	3.82	3.97
RCPOOS TBVOOS FHOOS2	100	2.35	2.14	2.37
	50	2.35	2.14	2.37
	40	2.72	2.46	2.74
	30	3.24	2.91	3.19
	30 at > 50%F	3.68	3.48	3.70
	25 at > 50%F	4.11	3.93	4.12
	30 at ≤ 50%F	3.50	3.30	3.50
	25 at ≤ 50%F	4.00	3.84	3.99

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop.

RCPOOS limits are only valid up to 50% rated core power, 50% rated core flow, and an active recirculation drive flow of 17.73 Mlbm/hr.

5 Oscillation Power Range Monitor (OPRM) Setpoint

(Technical Specification 3.3.1.1)

Technical Specification Table 3.3.1.1-1, Function 2f, identifies the OPRM upscale function.

Instrument setpoints are established, such that the reactor will be tripped before an oscillation can grow to the point where the SLMCPR is exceeded. An Option III stability analysis is performed for each reload core to determine allowable OLMCPR's as a function of OPRM setpoint. Analyses consider both steady state startup operation, and the case of a two recirculation pump trip from rated power.

The resulting stability based OLMCPR's are reported in Reference 1. The OPRM setpoint (*sometimes referred to as the Amplitude Trip, S_p*) is selected, such that required margin to the SLMCPR is provided without stability being a limiting event. Analyses are based on cycle specific DIVOM analyses performed per Reference 22. The calculated OLMCPR's are shown in Table 5.1. Review of results shown in COLR Table 4.2 indicates an OPRM setpoint of **1.14** may be used. The successive confirmation count (*sometimes referred to as N_p*) is provided in Table 5.2, per Reference 30.

Table 5.1 OPRM Setpoint Range*

OPRM Setpoint	OLMCPR (SS)	OLMCPR (2PT)
1.05	1.15	1.20
1.06	1.17	1.21
1.07	1.19	1.23
1.08	1.20	1.25
1.09	1.22	1.27
1.10	1.24	1.29
1.11	1.26	1.31
1.12	1.28	1.34
1.13	1.30	1.36
1.14	1.33	1.38
1.15	1.35	1.40

Table 5.2 OPRM Successive Confirmation Count Setpoint

Count	OPRM Setpoint
6	≥ 1.04
8	≥ 1.05
10	≥ 1.07
12	≥ 1.09
14	≥ 1.11
16	≥ 1.14
18	≥ 1.18
20	≥ 1.24

* Extrapolation beyond a setpoint of 1.15 is not allowed

6 APRM Flow Biased Rod Block Trip Settings

(Technical Requirements Manual Section 5.3.1 and Table 3.3.4-1)

The APRM rod block trip setting is based upon References 26 & 27, and is defined by the following:

SRB $\leq$ (0.66(W-ΔW) + 61%)	Allowable Value
SRB $\leq$ (0.66(W-ΔW) + 59%)	Nominal Trip Setpoint (NTSP)

where:

SRB	=	Rod Block setting in percent of rated thermal power (3458 MW_t)
W	=	Loop recirculation flow rate in percent of rated
ΔW	=	Difference between two-loop and single-loop effective recirculation flow at the same core flow (ΔW=0.0 for two-loop operation)

The APRM rod block trip setting is clamped at a maximum allowable value of 115% (corresponding to a NTSP of 113%).

7 Rod Block Monitor (RBM) Trip Setpoints and Operability

(Technical Specification Table 3.3.2.1-1)

The RBM trip setpoints and applicable power ranges, based on References 26 & 27, are shown in Table 7.1. Setpoints are based on an HTSP, unfiltered analytical limit of 114%. Unfiltered setpoints are consistent with a nominal RBM filter setting of 0.0 seconds; filtered setpoints are consistent with a nominal RBM filter setting less than 0.5 seconds. Cycle specific CRWE analyses of OLMCPR are documented in Reference 1, superseding values reported in References 26, 27, and 29.

Table 7.1 Analytical RBM Trip Setpoints*

RBM Trip Setpoint		Allowable Value (AV)	Nominal Trip Setpoint (NTSP)
LPSP		27%	25%
IPSP		62%	60%
HPSP		82%	80%
LTSP	- unfiltered	121.7%	120.0%
	- filtered	120.7%	119.0%
ITSP	- unfiltered	116.7%	115.0%
	- filtered	115.7%	114.0%
HTSP	- unfiltered	111.7%	110.0%
	- filtered	110.9%	109.2%
DTSP		90%	92%

As a result of cycle specific CRWE analyses, RBM setpoints in Technical Specification Table 3.3.2.1-1 are applicable as shown in Table 7.2. Cycle specific setpoint analysis results are shown in Table 7.3, per Reference 1.

Table 7.2 RBM Setpoint Applicability

Thermal Power (% Rated)	Applicable MCPR†	Notes from Table 3.3.2.1-1	Comment
> 27% and < 90%	< 1.70	(a), (b), (f), (h)	two loop operation
	< 1.74	(a), (b), (f), (h)	single loop operation
≥ 90%	< 1.39	(g)	two loop operation ‡

* Values are considered maximums. Using lower values, due to RBM system hardware/software limitations, is conservative, and acceptable.

† MCPR values shown correspond with, (support), SLMPCR values identified in Reference 1.

‡ Greater than 90% rated power is not attainable in single loop operation.

Table 7.3 Control Rod Withdrawal Error Results

RBM HTSP Analytical Limit Unfiltered	CRWE OLMCPR
107	1.20
111	1.26
114	1.37
117	1.37

Results, compared against the base case OLMCPR results of Table 4.2, indicate SLMCPR remains protected for RBM inoperable conditions (i.e., 114% unblocked).

8 Shutdown Margin Limit

(Technical Specification 3.1.1)

Assuming the strongest OPERABLE control blade is fully withdrawn, and all other OPERABLE control blades are fully inserted, the core shall be sub-critical and meet the following minimum shutdown margin:

SDM > 0.38% dk/k

EDMS L32 170206 803
QA Document
BFE-4151, Revision 0

TVA NPG

Reactor Engineering and Fuels - BWRFE

1101 Market Street, Chattanooga, TN 37402

Browns Ferry Unit 2 Cycle 20

Core Operating Limits Report, (105% OLTP)

TVA-COLR-BF2C20 **Revision 0 (Final)**
(Revision Log, Page v)

February 2017

Prepared:	T. W. Eichenberg, Sr. Specialist	Date:	Feb 8, 2017
Verified:	B. C. Mitchell, Engineer	Date:	2/9/17
Approved:	W. B. Bird, Manager, BWR Fuel Engineering	Date:	2/10/2017
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Approved:	Chairman, PORC	Date:	2/15/17
Approved:	Plant Manager	Date:	2/15/17

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Revision Log

Number	Page	Description
0-R0	All	New document.

Nomenclature

APLHGR	Average Planar LHGR
APRM	Average Power Range Monitor
AREVA NP	Vendor (Framatome, Siemens)
BOC	Beginning of Cycle
BSP	Backup Stability Protection
BWR	Boiling Water Reactor
CAVEX	Core Average Exposure
CD	Coast Down
CMSS	Core Monitoring System Software
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRWE	Control Rod Withdrawal Error
CSDM	Cold SDM
DIVOM	Delta CPR over Initial CPR vs. Oscillation Magnitude
EOC	End of Cycle
EOCLB	End-of-Cycle Licensing Basis
EOOS	Equipment OOS
FFTR	Final Feedwater Temperature Reduction
FFWTR	Final Feedwater Temperature Reduction
FHOOS	Feedwater Heaters OOS
ft	Foot: English unit of measure for length
GNF	Vendor (General Electric, Global Nuclear Fuels)
GWd	Giga Watt Day
HTSP	High TSP
ICA	Interim Corrective Action
ICF	Increased Core Flow (beyond rated)
IS	In-Service
kW	kilo watt: SI unit of measure for power.
LCO	License Condition of Operation
LFWH	Loss of Feedwater Heating
LHGRFAC	LHGR Multiplier (Power or Flow dependent)
LPRM	Low Power Range Monitor
LRNB	Generator Load Reject, No Bypass
MAPFAC	MAPLHGR multiplier (Power or Flow dependent)
MCPR	Minimum CPR

MSRV	Moisture Separator Reheater Valve
MSRVOOS	MSRV OOS
MTU	Metric Ton Uranium
MWd/MTU	Mega Watt Day per Metric Ton Uranium
NEOC	Near EOC
NRC	United States Nuclear Regulatory Commission
NSS	Nominal Scram Speed
NTSP	Nominal TSP
OLMCPR	MCPR Operating Limit
OOS	Out-Of-Service
OPRM	Oscillation Power Range Monitor
OSS	Optimum Scram Speed
PBDA	Period Based Detection Algorithm
Pbypass	Power, below which TSV Position and TCV Fast Closure Scrams are Bypassed
PLU	Power Load Unbalance
PLUOOS	PLU OOS
PRNM	Power Range Neutron Monitor
RBM	Rod Block Monitor
RCPOOS	Recirculation Pump OOS (*SLO*)
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPTOOS	RPT OOS
SDM	Shutdown Margin
SLMCPR	MCPR Safety Limit
SLO	Single Loop Operation
TBV	Turbine Bypass Valve
TBVIS	TBV IS
TBVOOS	Turbine Bypass Valves OOS
TIP	Transversing In-core Probe
TIPOOS	TIP OOS
TLO	Two Loop Operation
TSP	Trip Setpoint
TSSS	Technical Specification Scram Speed
TVA	Tennessee Valley Authority

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<u>PRNM Setpoint References</u>

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28. GE Letter LB#: 262-97-133, Browns Ferry Nuclear Plant Rod Block Monitor Setpoint Clarification - GE Proprietary Information, September 12, 1997.

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1 Introduction

In anticipation of cycle startup, it is necessary to describe the expected limits of operation.

1.1 Purpose

The primary purpose of this document is to satisfy requirements identified by unit technical specification section 5.6.5. This document may be provided, upon final approval, to the NRC.

1.2 Scope

This document will discuss the following areas:

- Average Planar Linear Heat Generation Rate (APLHGR) Limit
  (Technical Specifications 3.2.1 and 3.7.5)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Linear Heat Generation Rate (LHGR) Limit
  (Technical Specification 3.2.3, 3.3.4.1, and 3.7.5)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Minimum Critical Power Ratio Operating Limit (OLMCPR)
  (Technical Specifications 3.2.2, 3.3.4.1, 3.7.5 and Table 3.3.2.1-1)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Oscillation Power Range Monitor (OPRM) Setpoint
  (Technical Specification Table 3.3.1.1)
  Applicability: Mode 1, ≥ (as specified in Technical Specifications Table 3.3.1.1-1)
- Average Power Range Monitor (APRM) Flow Biased Rod Block Trip Setting
  (Technical Requirements Manual Section 5.3.1 and Table 3.3.4-1)
  Applicability: Mode 1, ≥ (as specified in Technical Requirements Manuals Table 3.3.4-1)
- Rod Block Monitor (RBM) Trip Setpoints and Operability
  (Technical Specification Table 3.3.2.1-1)
  Applicability: Mode 1, ≥ % RTP as specified in Table 3.3.2.1-1 (TS definition of RTP)
- Shutdown Margin (SDM) Limit
  (Technical Specification 3.1.1)
  Applicability: All Modes

1.3 Fuel Loading

The core will contain previously exposed AREVA NP, Inc., ATRIUM-10 fuel, along with fresh and exposed ATRIUM-10XM; exposed ATRIUM-11 lead fuel. Nuclear fuel types used in the core loading are shown in Table 1.1. The core shuffle and final loading were explicitly evaluated for BOC cold shutdown margin performance as documented per Reference 5.

Table 1.1 Nuclear Fuel Types*

Fuel Description	Original Cycle	Number of Assemblies	Nuclear Fuel Type (NFT)	Fuel Names (Range)
ATRIUM-10 A10-4165B-15GV75-FBE	18	110	12	FBE001-FBE176
ATRIUM-10 A10-4107B-13GV75-FBE	18	44	13	FBE177-FBE244
ATRIUM-10 A10-4176B-10GV75-FBE	18	65	14	FBE245-FBE316
ATRIUM 10XM XMLC-3904B-15GV80-FBF	19	172	15	FBF401-FBF572
ATRIUM 10XM XMLC-4035B-13GV80-FBF	19	79	16	FBF573-FBF652
ATRIUM 11 A11-3693B-13GV80-FBF	19	6	17	FBF653-FBF660
ATRIUM 10XM XMLC-4102B-11GV70-FBG-B	20	48	18	FBG701-FBG748
ATRIUM 10XM XMLC-4062B-13GV80-FBG-C	20	152	19	FBG749-FBG900
ATRIUM 10XM XMLC-3948B-13GV70-FBG-B	20	88	20	FBG901-FBG988

1.4 Acceptability

Limits discussed in this document were generated based on NRC approved methodologies per References 6 through 25.

* The table identifies the expected fuel type breakdown in anticipation of final core loading. The final composition of the core depends upon uncertainties during the outage such as discovering a failed fuel bundle, or other bundle damage. Minor core loading changes, due to unforeseen events, will conform to the safety and monitoring requirements identified in this document.

2 APLHGR Limits

(Technical Specifications 3.2.1 & 3.7.5)

The APLHGR limit is determined by adjusting the rated power APLHGR limit for off-rated power, off-rated flow, and SLO conditions. The most limiting of these is then used as follows:

$$\text{APLHGR limit} = \text{MIN}\ (\text{APLHGR}_P,\ \text{APLHGR}_F,\ \text{APLHGR}_{SLO})$$

where:

APLHGR_P	off-rated power APLHGR limit	[APLHGR_{RATED} * MAPFAC_P]
APLHGR_F	off-rated flow APLHGR limit	[APLHGR_{RATED} * MAPFAC_F]
APLHGR_{SLO}	SLO APLHGR limit	[APLHGR_{RATED} * SLO Multiplier]

2.1 Rated Power and Flow Limit: APLHGR_{RATED}

The rated conditions APLHGR for all fuel are identified per Reference 1. The rated conditions APLHGR for ATRIUM-10 fuel are shown in Figure 2.1. The rated conditions APLHGR for ATRIUM-10XM are shown in Figure 2.2. The rated conditions APLHGR for ATRIUM-11 are shown in Figure 2.3.

2.2 Off-Rated Power Dependent Limit: APLHGR_P

Reference 1 does not specify a power dependent APLHGR. Therefore, MAPFAC_P is set to a value of **1.0**.

2.2.1 <u>Startup without Feedwater Heaters</u>

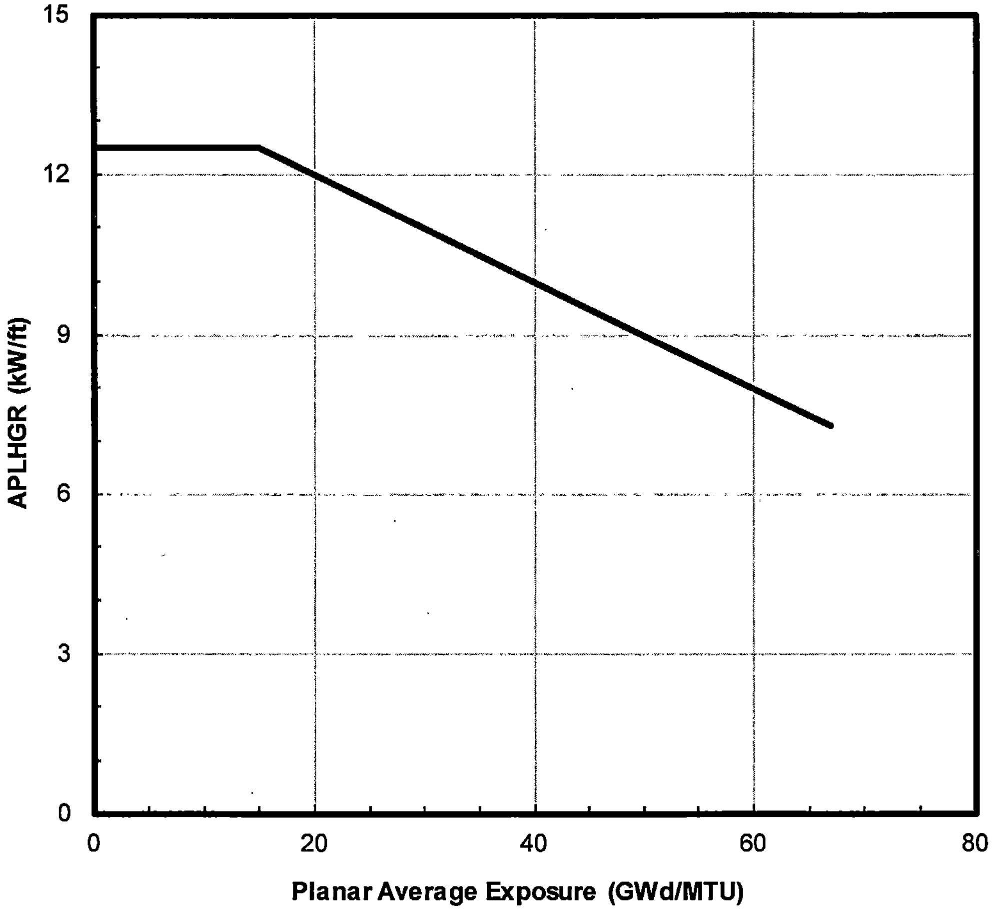
There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. No additional power dependent limitation is required.

2.3 Off-Rated Flow Dependent Limit: APLHGR_F

Reference 1 does not specify a flow dependent APLHGR. Therefore, MAPFAC_F is set to a value of **1.0**.

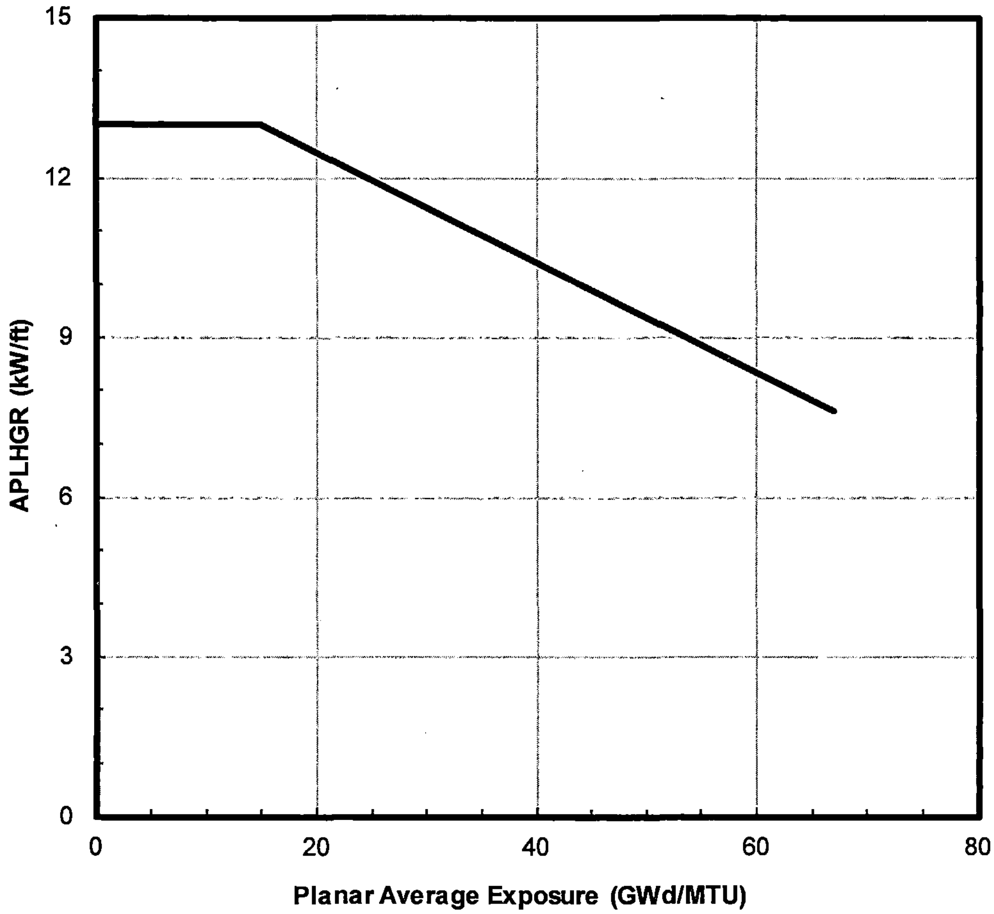
2.4 Single Loop Operation Limit: APLHGR_{SLO}

The single loop operation multiplier for ATRIUM-10, ATRIUM-10XM, and ATRIUM-11 fuel is **0.85**, per Reference 1.



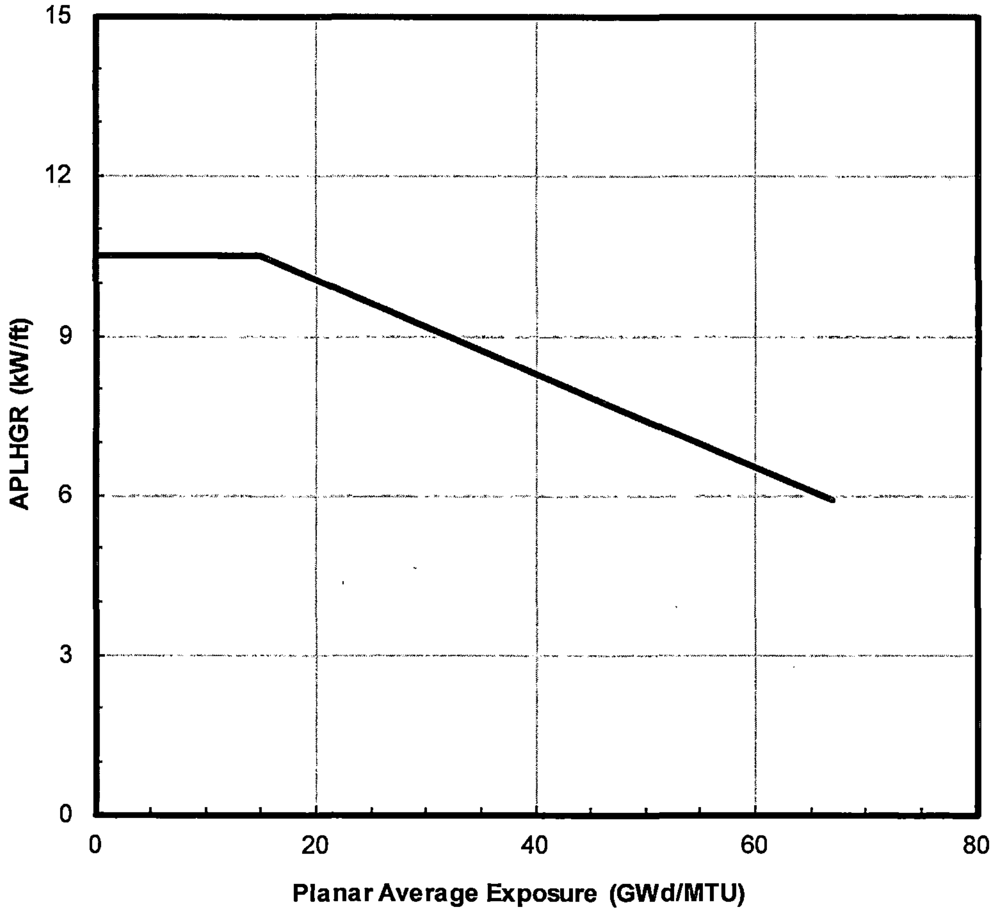
Planar Avg. Exposure (GWd/MTU)	APLHGR Limit (kW/ft)
0.0	12.5
15.0	12.5
67.0	7.3

Figure 2.1 APLHGR$_{RATED}$ for ATRIUM-10 Fuel



Planar Avg. Exposure (GWd/MTU)	APLHGR Limit (kW/ft)
0.0	13.0
15.0	13.0
67.0	7.6

Figure 2.2 APLHGR$_{RATED}$ for ATRIUM-10XM Fuel



Planar Avg. Exposure (GWd/MTU)	APLHGR Limit (kW/ft)
0.0	10.5
15.0	10.5
67.0	5.9

Figure 2.3 APLHGR$_{RATED}$ for ATRIUM-11 Fuel

2.5 Equipment Out-Of-Service Corrections

The limits shown in Figure 2.1, Figure 2.2, and Figure 2.3 are applicable for operation with all equipment In-Service as well as the following Equipment Out-Of-Service (EOOS) options; including combinations of the options.

In-Service	All equipment In-Service *
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
PLUOOS	Power Load Unbalance Out-Of-Service
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service or Final Feedwater Temperature Reduction
RCPOOS	One Recirculation Pump Out-Of-Service

* All equipment service conditions assume 1 SRVOOS.

3 LHGR Limits

(Technical Specification 3.2.3, 3.3.4.1, & 3.7.5)

The LHGR limit is determined by adjusting the rated power LHGR limit for off-rated power and off-rated flow conditions. The most limiting of these is then used as follows:

$$\text{LHGR limit} = \text{MIN} \left(LHGR_P, LHGR_F \right)$$

where:

$LHGR_P$	off-rated power LHGR limit	$[LHGR_{RATED} * LHGRFAC_P]$
$LHGR_F$	off-rated flow LHGR limit	$[LHGR_{RATED} * LHGRFAC_F]$

3.1 Rated Power and Flow Limit: $LHGR_{RATED}$

The rated conditions LHGR for all fuel are identified per Reference 1. The rated conditions LHGR for ATRIUM-10 are shown in Figure 3.1. The rated conditions LHGR for ATRIUM-10XM fuel is shown in Figure 3.2. The rated conditions LHGR for ATRIUM-11 fuel is shown in Figure 3.3. The LHGR limit is consistent with References 2 and 3.

3.2 Off-Rated Power Dependent Limit: $LHGR_P$

LHGR limits are adjusted for off-rated power conditions using the $LHGRFAC_P$ multiplier provided in Reference 1. The multiplier is split into two sub cases: turbine bypass valves in and out-of-service. The base case multipliers are shown in Figure 3.4 through Figure 3.6.

3.2.1 Startup without Feedwater Heaters

There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. Additional limits are shown in Figure 3.10 through Figure 3.15, based on temperature conditions identified in Table 3.1.

Table 3.1 Startup Feedwater Temperature Basis

Power	Temperature	
	Range 1	Range 2
(% Rated)	(°F)	(°F)
25	160.0	155.0
30	165.0	160.0
40	175.0	170.0
50	185.0	180.0

3.3 Off-Rated Flow Dependent Limit: $LHGR_F$

LHGR limits are adjusted for off-rated flow conditions using the $LHGRFAC_F$ multiplier provided in Reference 1. Multipliers are shown in Figure 3.7 through Figure 3.9.

3.4 Equipment Out-Of-Service Corrections

The limits shown in Figure 3.1 through Figure 3.3 are applicable for operation with all equipment In-Service as well as the following Equipment Out-Of-Service (EOOS) options; including combinations of the options. *

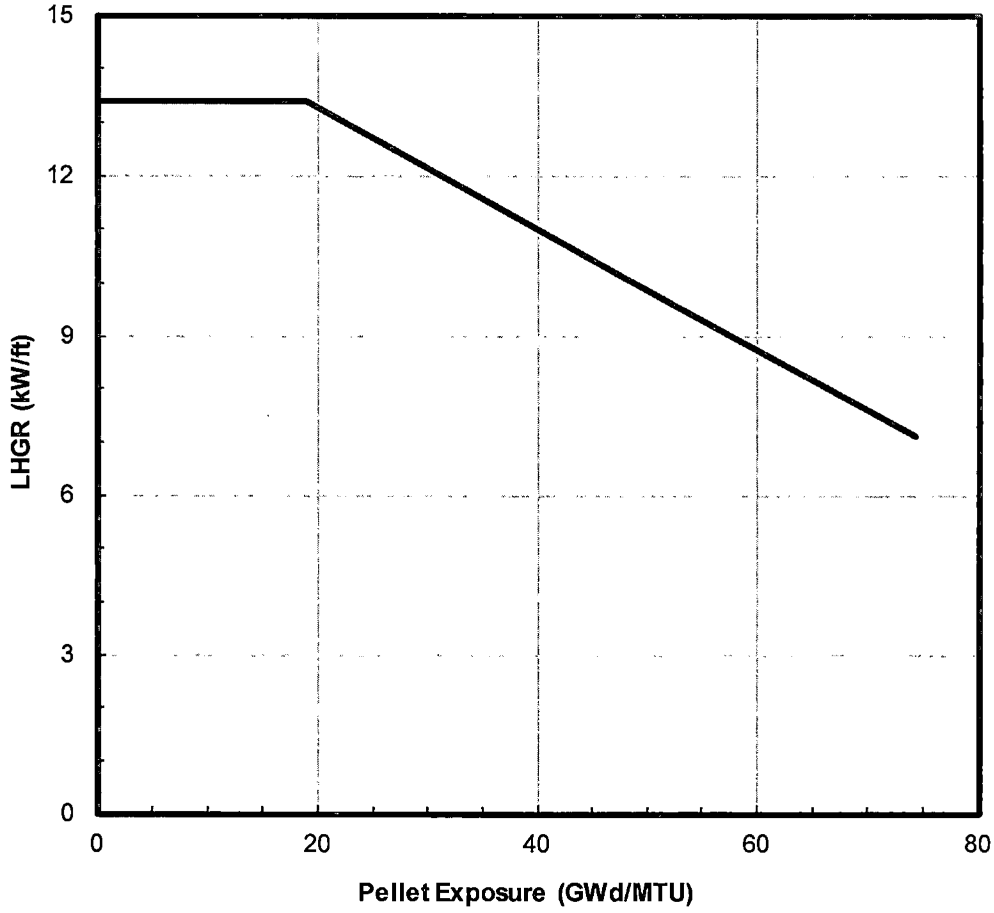
In-Service	All equipment In-Service
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
PLUOOS	Power Load Unbalance Out-Of-Service
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service or Final Feedwater Temperature Reduction
RCPOOS	One Recirculation Pump Out-Of-Service

Off-rated power corrections shown in Figure 3.4 through Figure 3.6 are dependent on operation of the Turbine Bypass Valve system. For this reason, separate limits are to be applied for TBVIS or TBVOOS operation. The limits have no dependency on RPTOOS, PLUOOS, FHOOS/FFWTR, or SLO.

Off-rated flow corrections shown in Figure 3.7 through Figure 3.9 are bounding for all EOOS conditions.

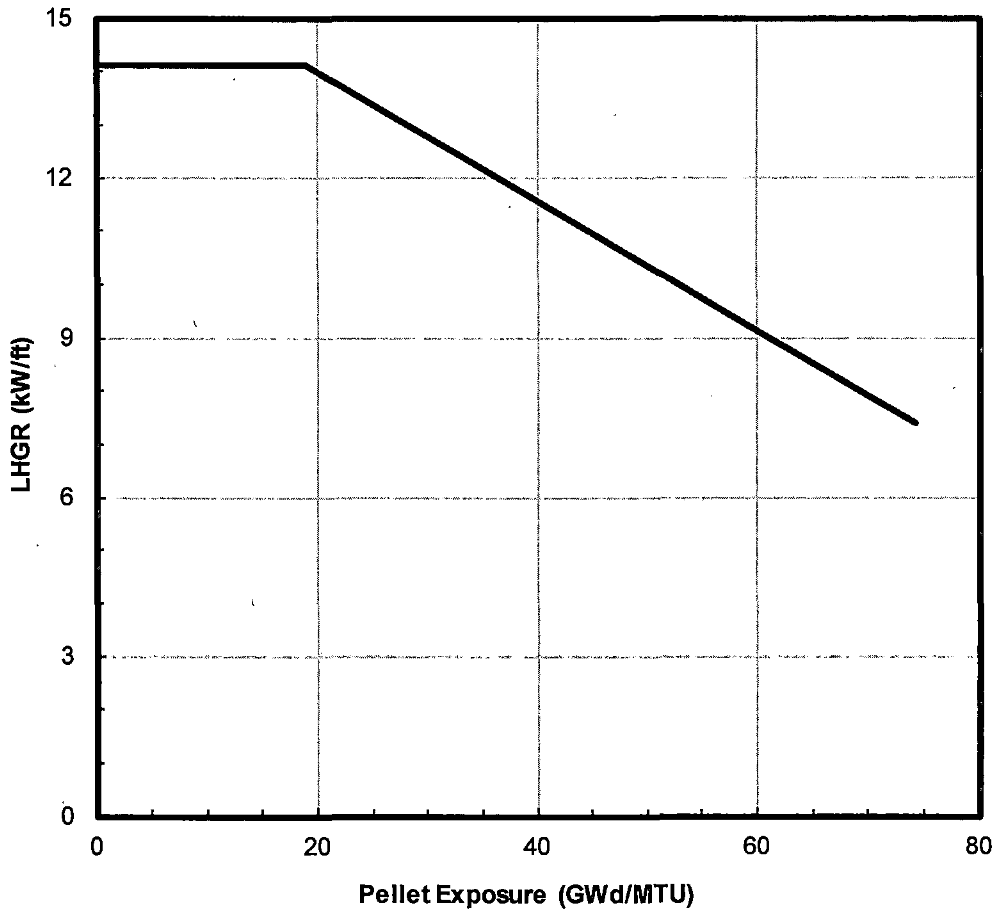
Off-rated power corrections shown in Figure 3.10 through Figure 3.15 are also dependent on operation of the Turbine Bypass Valve system. In this case, limits support FHOOS operation during startup. These limits have no dependency on RPTOOS, PLUOOS, or SLO.

* All equipment service conditions assume 1 SRVOOS.



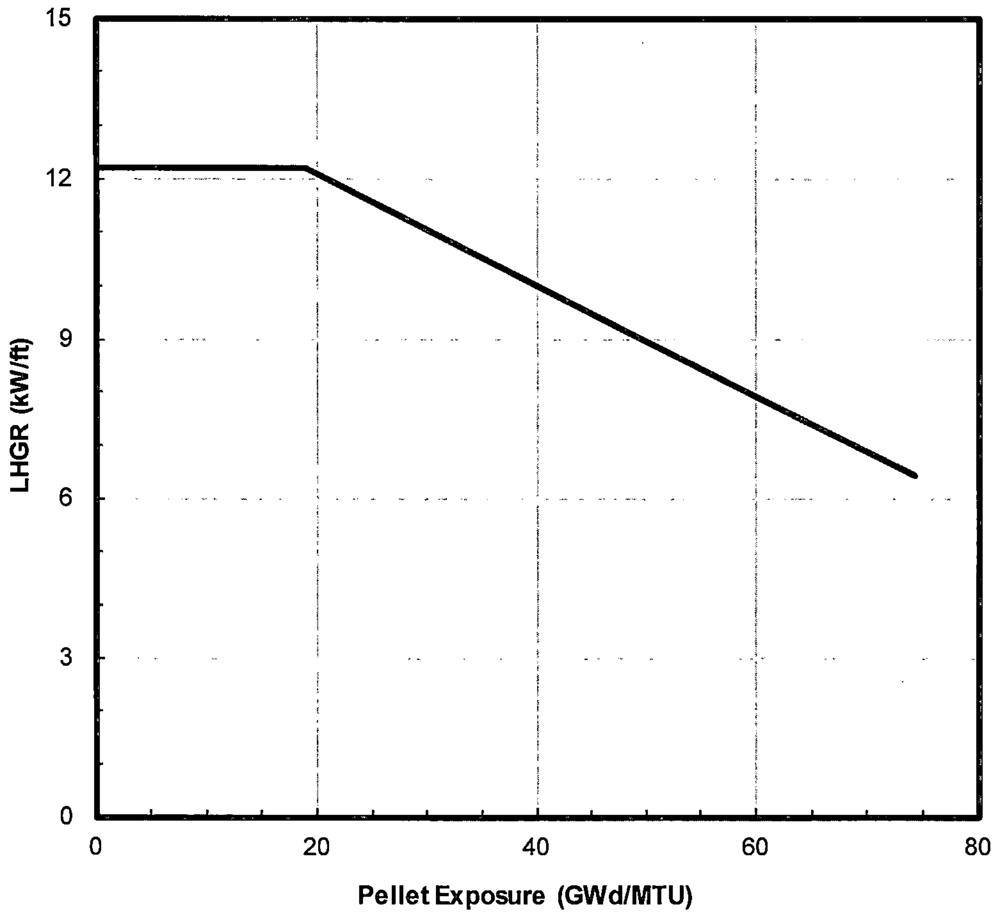
Pellet Exposure (GWd/MTU)	LHGR Limit (kW/ft)
0.0	13.4
18.9	13.4
74.4	7.1

Figure 3.1 LHGR$_{RATED}$ for ATRIUM-10 Fuel



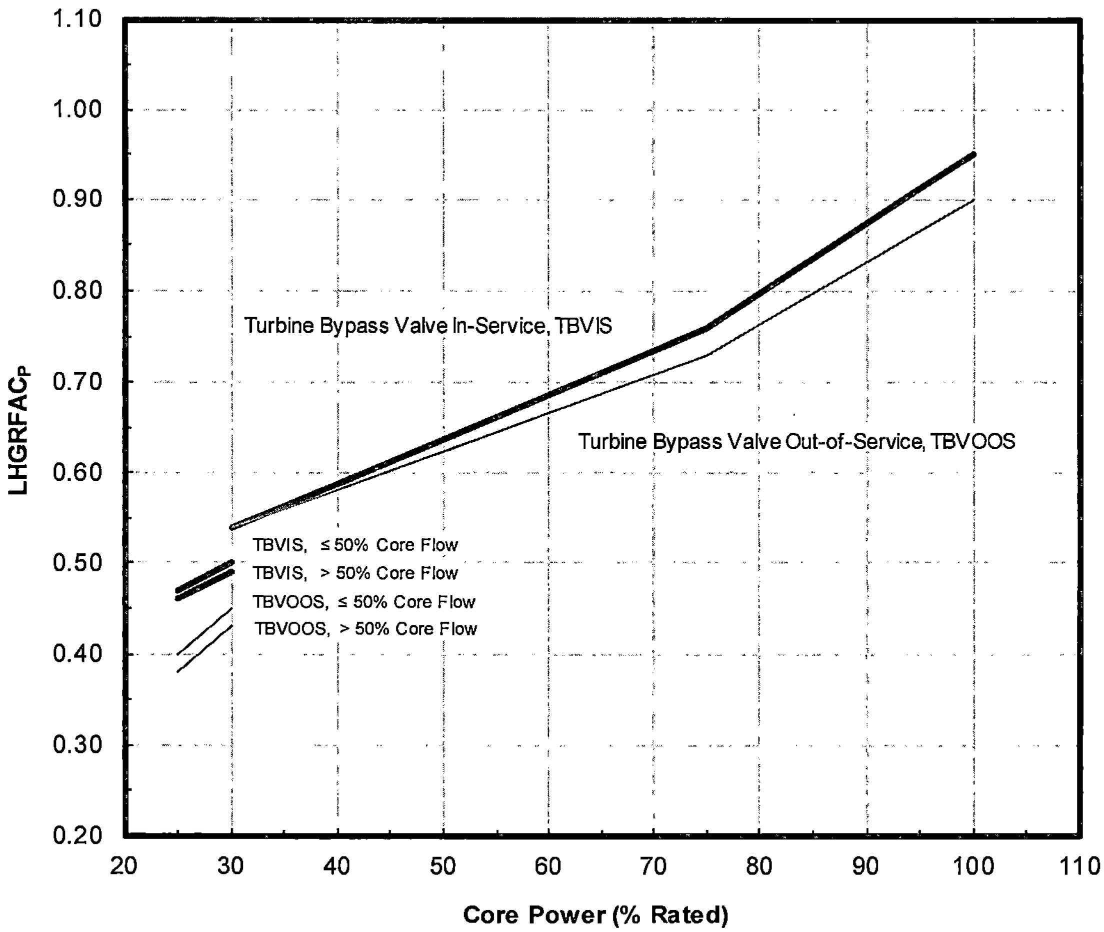
Pellet Exposure (GWd/MTU)	LHGR Limit (kW/ft)
0.0	14.1
18.9	14.1
74.4	7.4

Figure 3.2 LHGR$_{RATED}$ for ATRIUM-10XM Fuel



Pellet Exposure (GWd/MTU)	LHGR Limit (kW/ft)
0.0	12.2
18.9	12.2
74.4	6.4

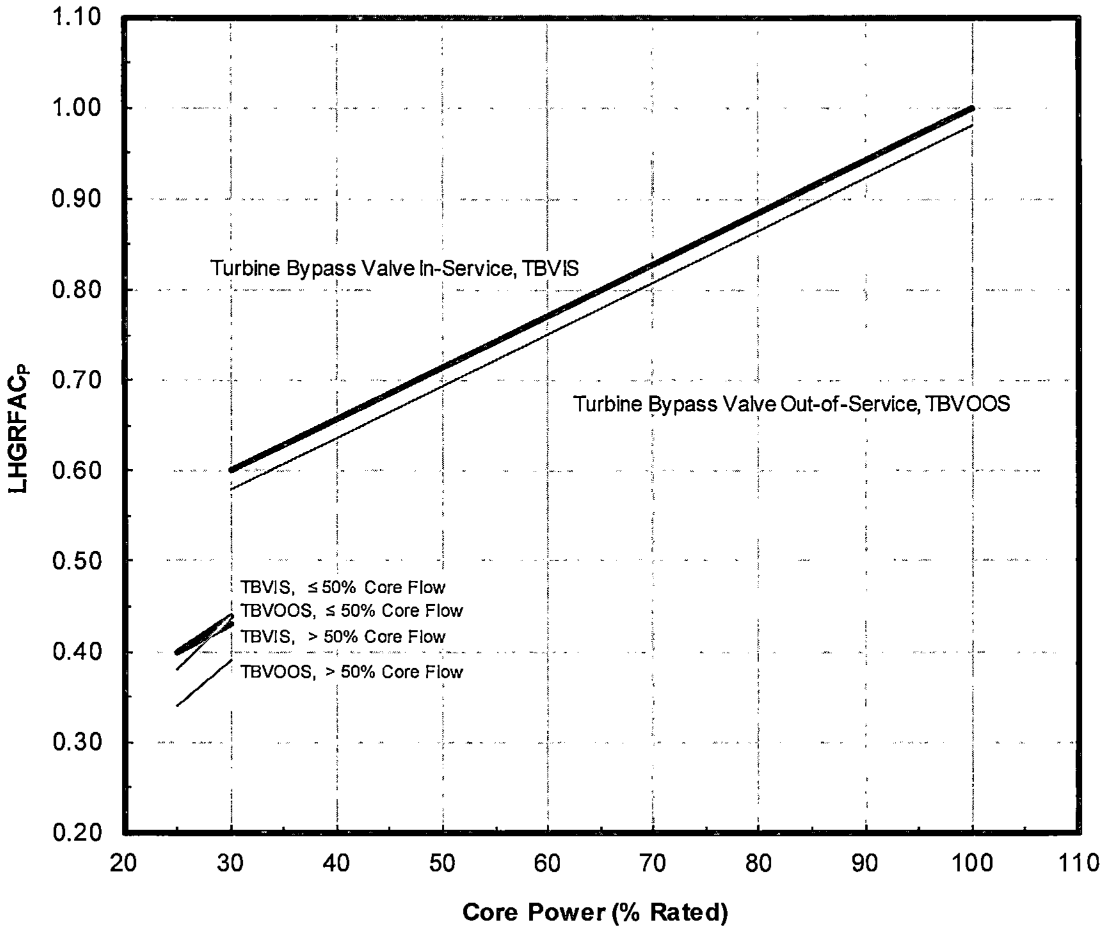
Figure 3.3 LHGR$_{RATED}$ for ATRIUM-11 Fuel



Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$	*Turbine Bypass Out-of-Service* Core Power (% Rated)	LHGRFAC$_P$
100.0	0.95	100.0	0.90
75.0	0.76	75.0	0.73
30.0	0.54	30.0	0.54
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.49	30.0	0.43
25.0	0.46	25.0	0.38
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.50	30.0	0.45
25.0	0.47	25.0	0.40

Figure 3.4 Base Operation LHGRFAC$_P$ for ATRIUM-10 Fuel

(*Independent of other EOOS conditions*)

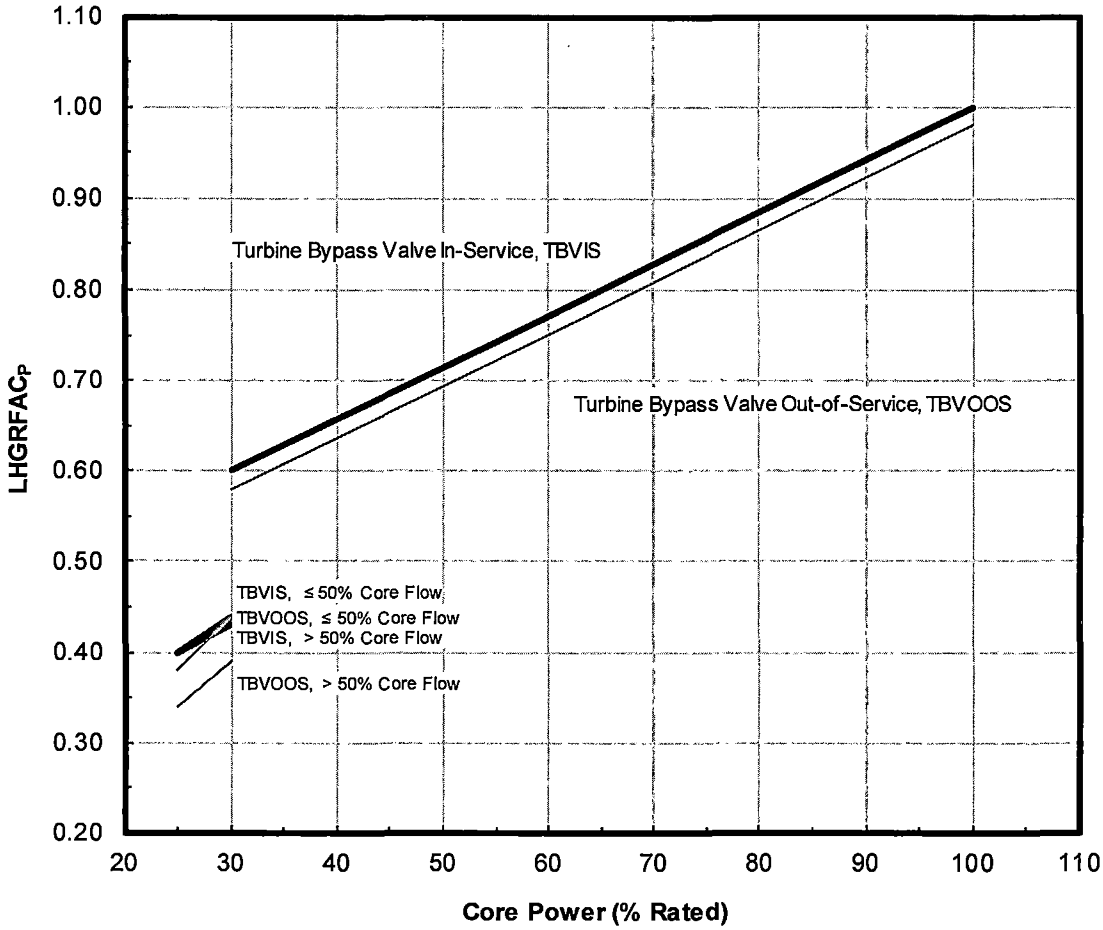


Turbine Bypass In-Service Core Power (% Rated)	$LHGRFAC_P$
100.0	1.00
30.0	0.60
Core Flow > 50% Rated	
30.0	0.43
25.0	0.40
Core Flow ≤ 50% Rated	
30.0	0.44
25.0	0.40

Turbine Bypass Out-of-Service Core Power (% Rated)	$LHGRFAC_P$
100.0	0.98
30.0	0.58
Core Flow > 50% Rated	
30.0	0.39
25.0	0.34
Core Flow ≤ 50% Rated	
30.0	0.44
25.0	0.38

Figure 3.5 Base Operation $LHGRFAC_P$ for ATRIUM-10XM Fuel

(*Independent of other EOOS conditions*)

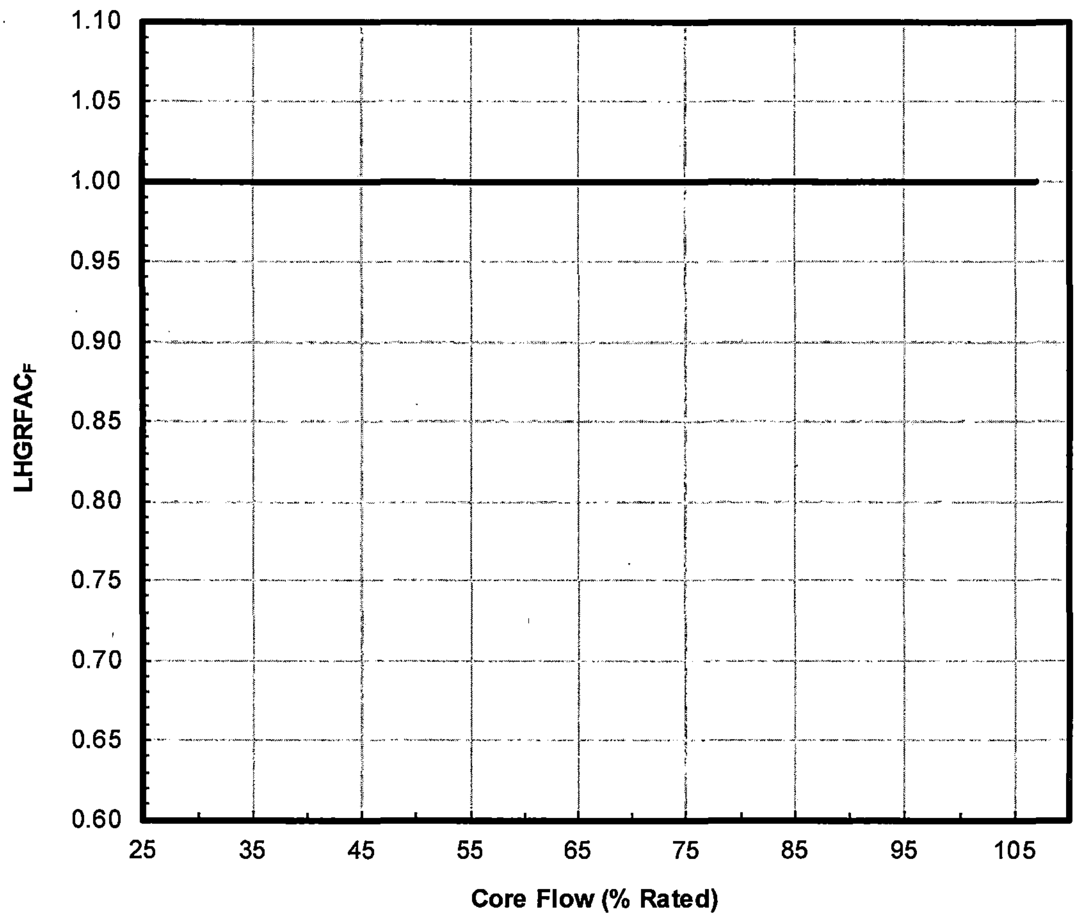


Turbine Bypass In-Service Core Power (% Rated)	$LHGRFAC_P$
100.0	1.00
30.0	0.60
Core Flow > 50% Rated	
30.0	0.43
25.0	0.40
Core Flow ≤ 50% Rated	
30.0	0.44
25.0	0.40

Turbine Bypass Out-of-Service Core Power (% Rated)	$LHGRFAC_P$
100.0	0.98
30.0	0.58
Core Flow > 50% Rated	
30.0	0.39
25.0	0.34
Core Flow ≤ 50% Rated	
30.0	0.44
25.0	0.38

Figure 3.6 Base Operation $LHGRFAC_P$ for ATRIUM-11 Fuel

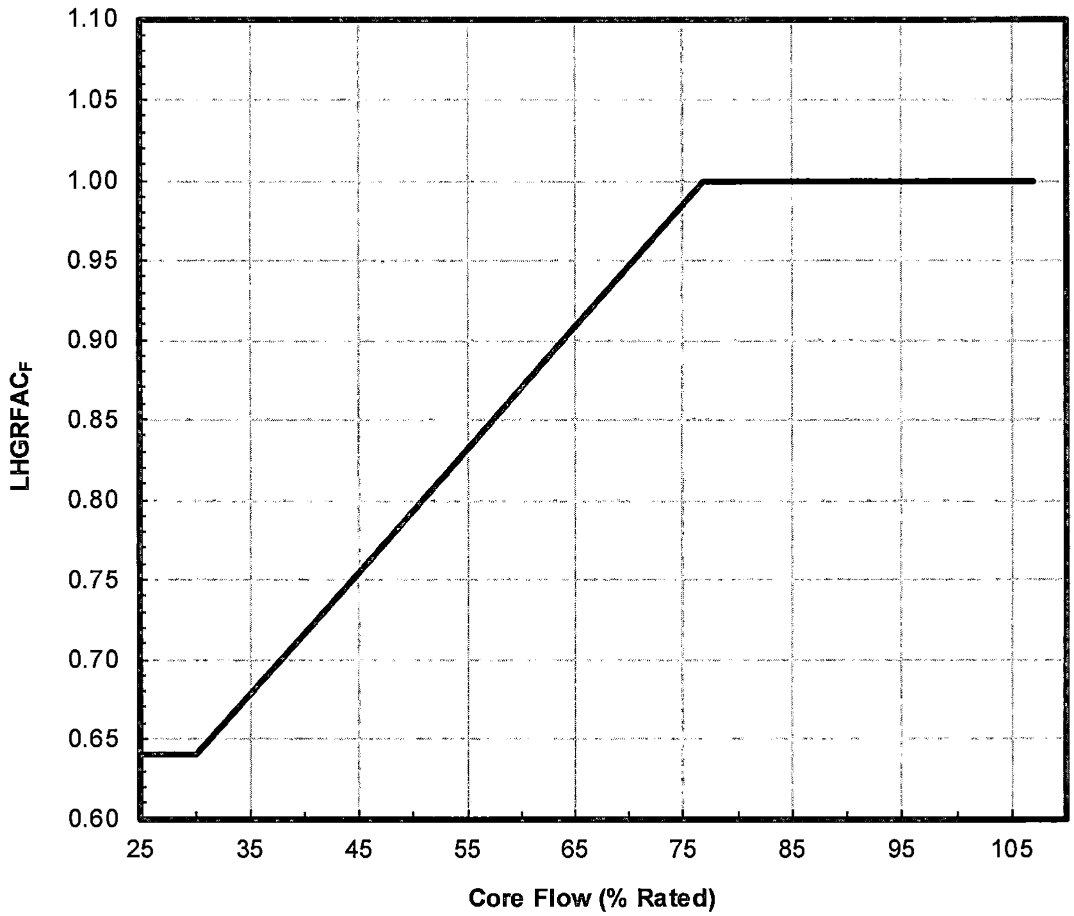
(*Independent of other EOOS conditions*)



Core Flow (% Rated)	$LHGRFAC_F$
0.0	1.00
30.0	1.00
107.0	1.00

Figure 3.7 $LHGRFAC_F$ for ATRIUM-10 Fuel
(Values bound all EOOS conditions)

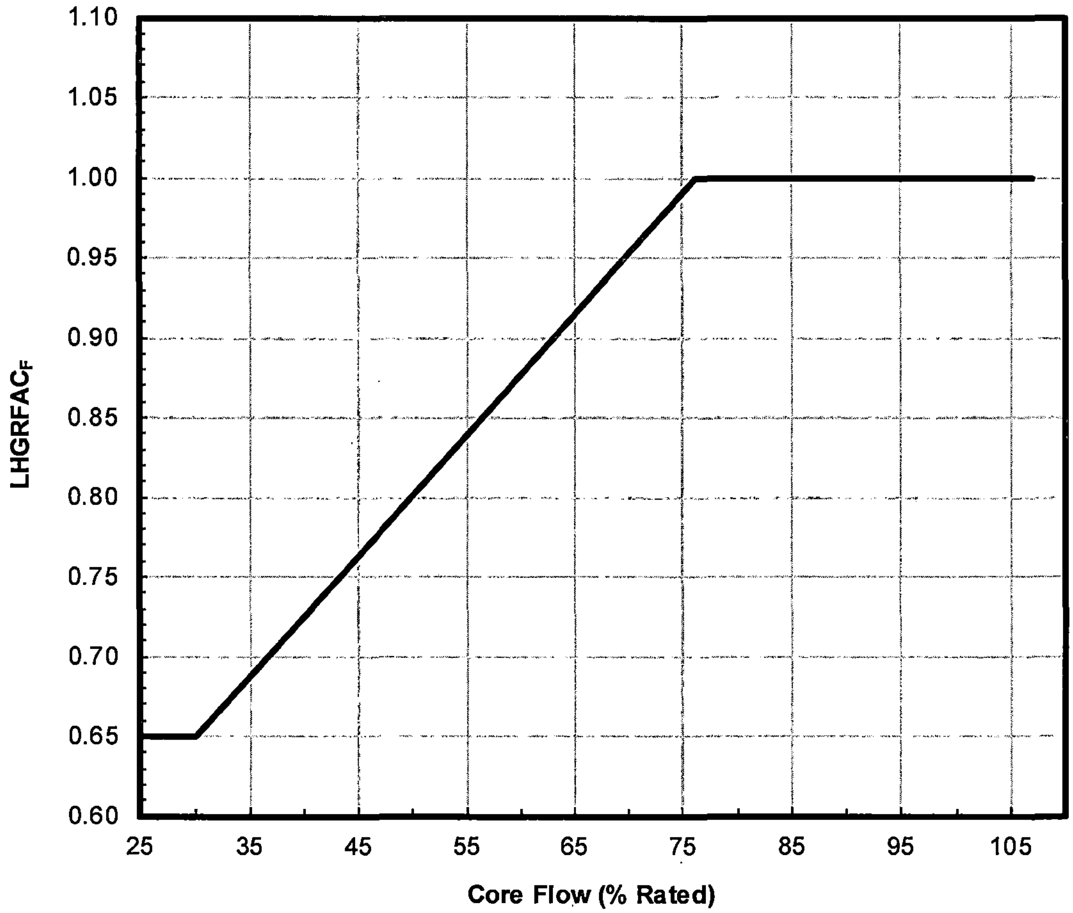
(107.0% maximum core flowline is used to support 105% rated flow operation, ICF)



Core Flow (% Rated)	LHGRFAC$_F$
0.0	0.64
30.0	0.64
76.8	1.00
107.0	1.00

Figure 3.8 LHGRFAC$_F$ for ATRIUM-10XM Fuel
(*Values bound all EOOS conditions*)

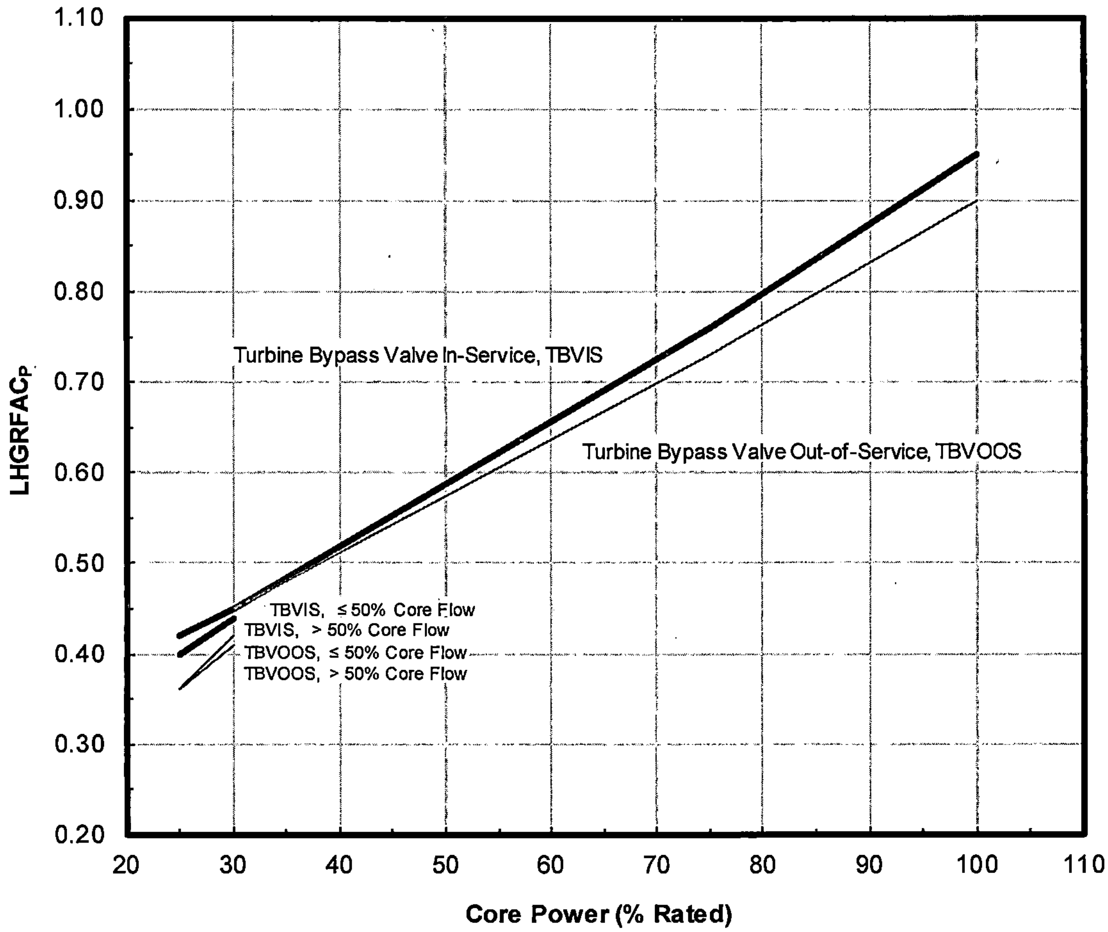
(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)



Core Flow (% Rated)	LHGRFAC$_F$
0.0	0.65
30.0	0.65
76.1	1.00
107.0	1.00

Figure 3.9 LHGRFAC$_F$ for ATRIUM-11 Fuel
(*Values bound all EOOS conditions*)

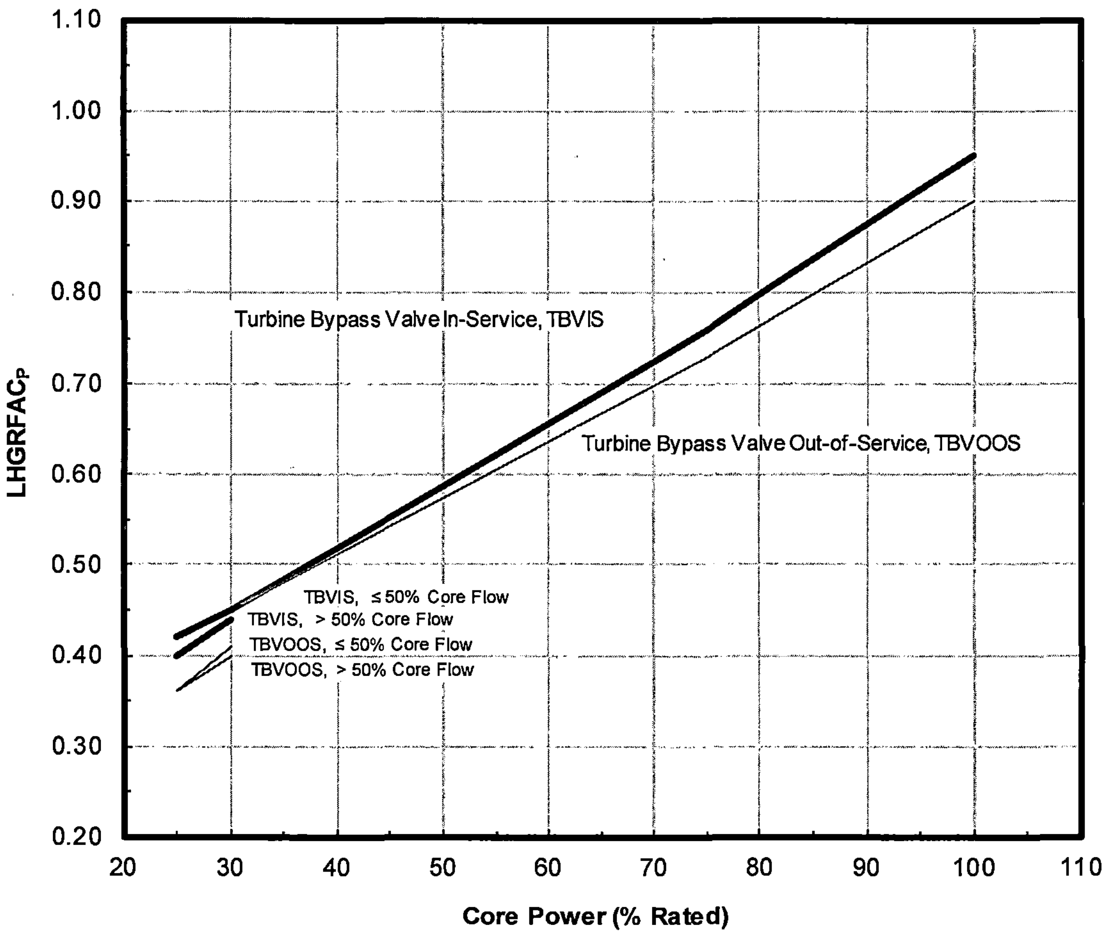
(*107.0% maximum core flowline is used to support 105% rated flow operation, ICF*)



Turbine Bypass In-Service	
Core Power	**LHGRFAC$_P$**
(% Rated)	
100.0	0.95
75.0	0.76
30.0	0.45
Core Flow > 50% Rated	
30.0	0.44
25.0	0.40
Core Flow ≤ 50% Rated	
30.0	0.45
25.0	0.42

Turbine Bypass Out-of-Service	
Core Power	**LHGRFAC$_P$**
(% Rated)	
100.0	0.90
75.0	0.73
30.0	0.45
Core Flow > 50% Rated	
30.0	0.41
25.0	0.36
Core Flow ≤ 50% Rated	
30.0	0.42
25.0	0.36

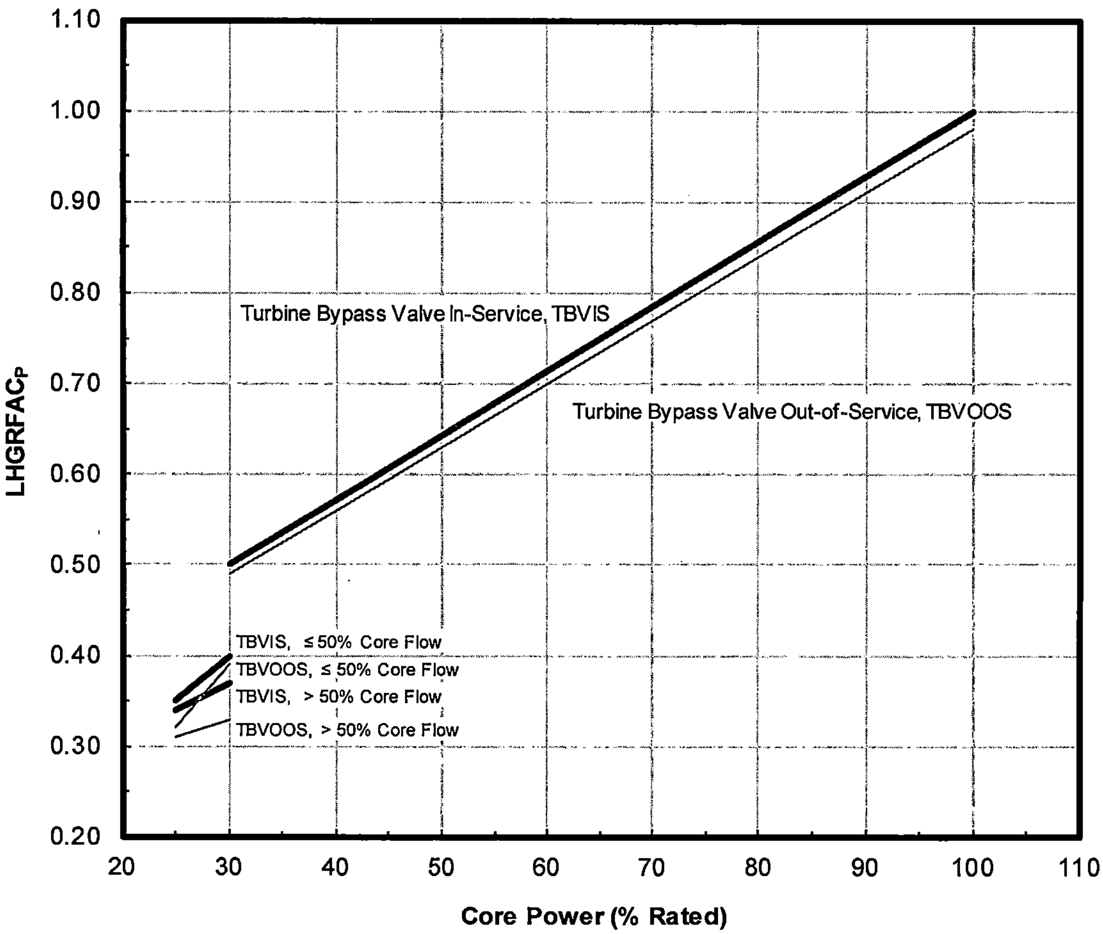
Figure 3.10 Startup Operation LHGRFAC$_P$ for ATRIUM-10 Fuel:
Table 3.1 Temperature Range 1
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$
100.0	0.95
75.0	0.76
30.0	0.45
Core Flow > 50% Rated	
30.0	0.44
25.0	0.40
Core Flow ≤ 50% Rated	
30.0	0.45
25.0	0.42

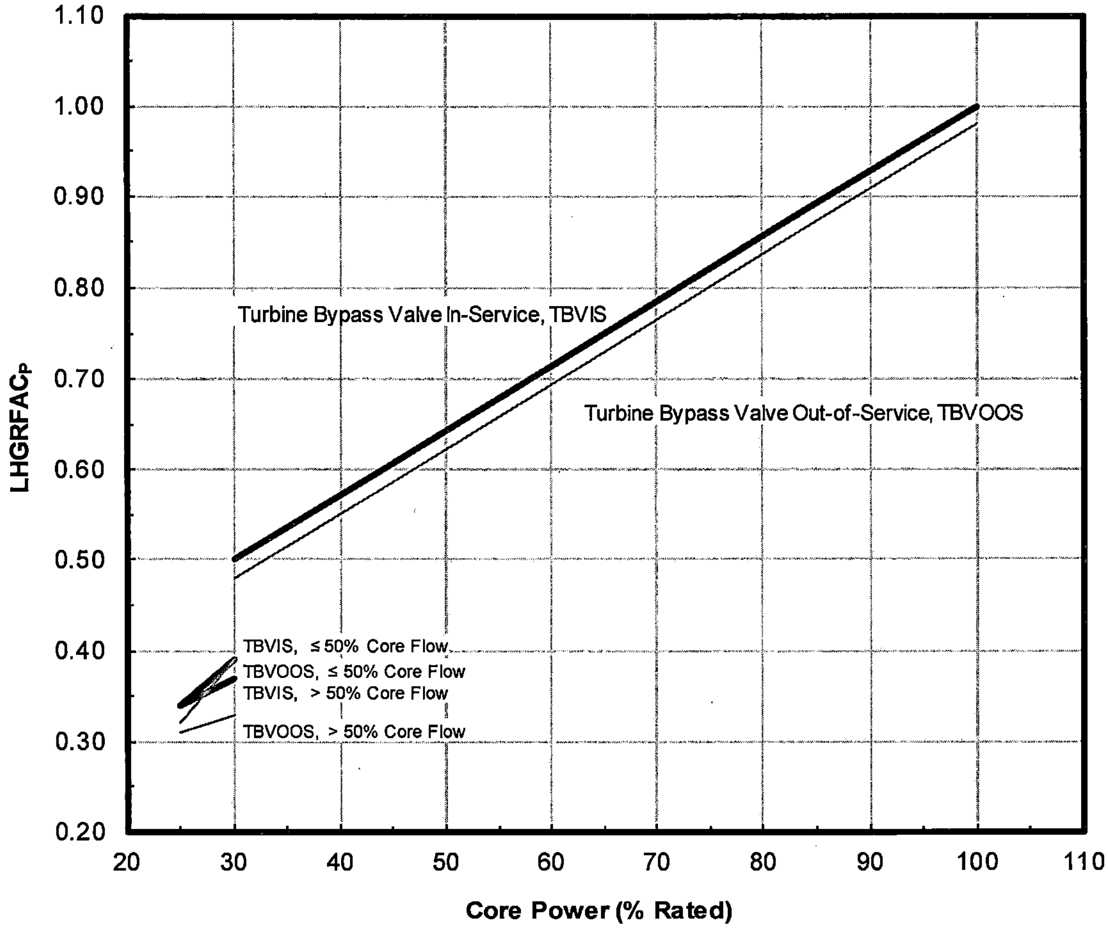
Turbine Bypass Out-of-Service Core Power (% Rated)	LHGRFAC$_P$
100.0	0.90
75.0	0.73
30.0	0.45
Core Flow > 50% Rated	
30.0	0.40
25.0	0.36
Core Flow ≤ 50% Rated	
30.0	0.41
25.0	0.36

Figure 3.11 Startup Operation LHGRFAC$_P$ for ATRIUM-10 Fuel:
Table 3.1 Temperature Range 2
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$	*Turbine Bypass Out-of-Service* Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00	100.0	0.98
30.0	0.50	30.0	0.49
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.37	30.0	0.33
25.0	0.34	25.0	0.31
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.40	30.0	0.39
25.0	0.35	25.0	0.32

Figure 3.12 Startup Operation LHGRFAC$_P$ for ATRIUM-10XM
Fuel: Table 3.1 Temperature Range 1
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



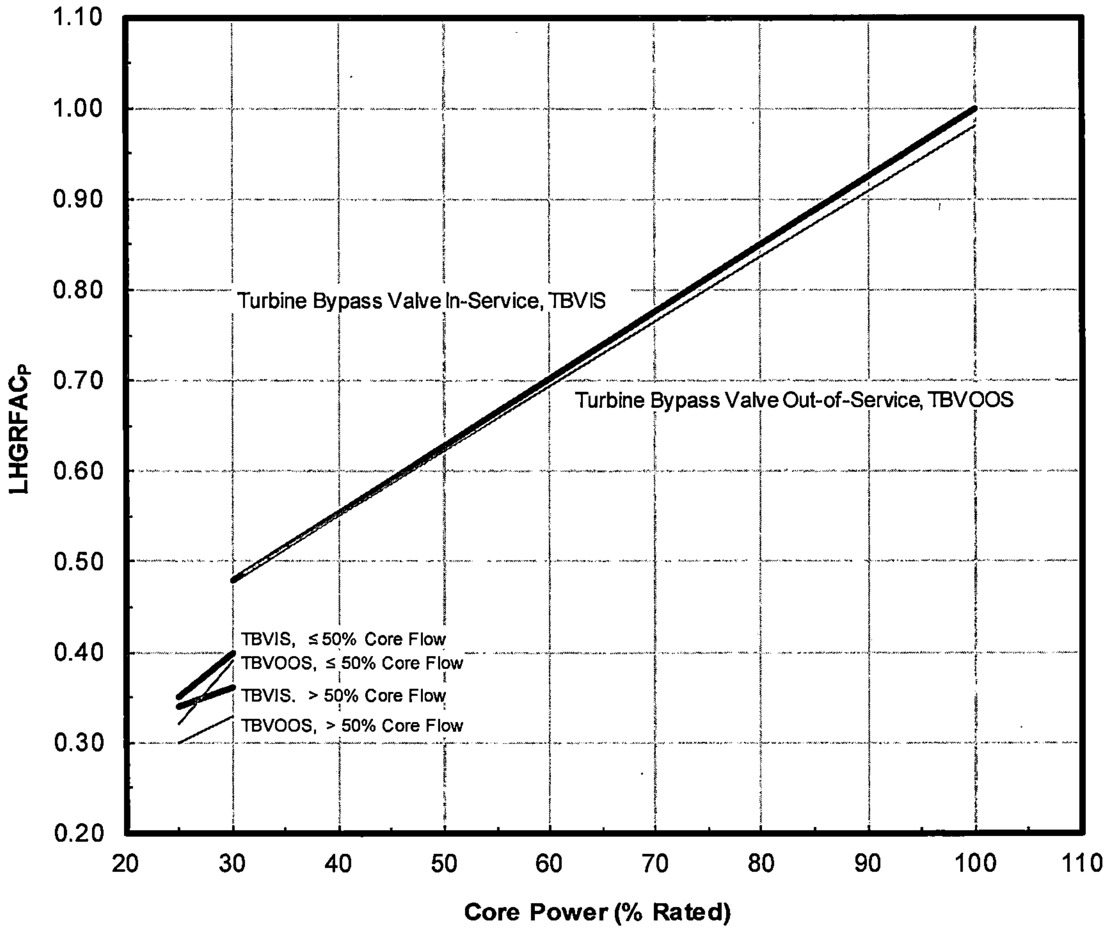
Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00
30.0	0.50
Core Flow > 50% Rated	
30.0	0.37
25.0	0.34
Core Flow ≤ 50% Rated	
30.0	0.39
25.0	0.34

Turbine Bypass Out-of-Service Core Power (% Rated)	LHGRFAC$_P$
100.0	0.98
30.0	0.48
Core Flow > 50% Rated	
30.0	0.33
25.0	0.31
Core Flow ≤ 50% Rated	
30.0	0.39
25.0	0.32

Figure 3.13 Startup Operation LHGRFAC$_P$ for ATRIUM-10XM Fuel: Table 3.1 Temperature Range 2

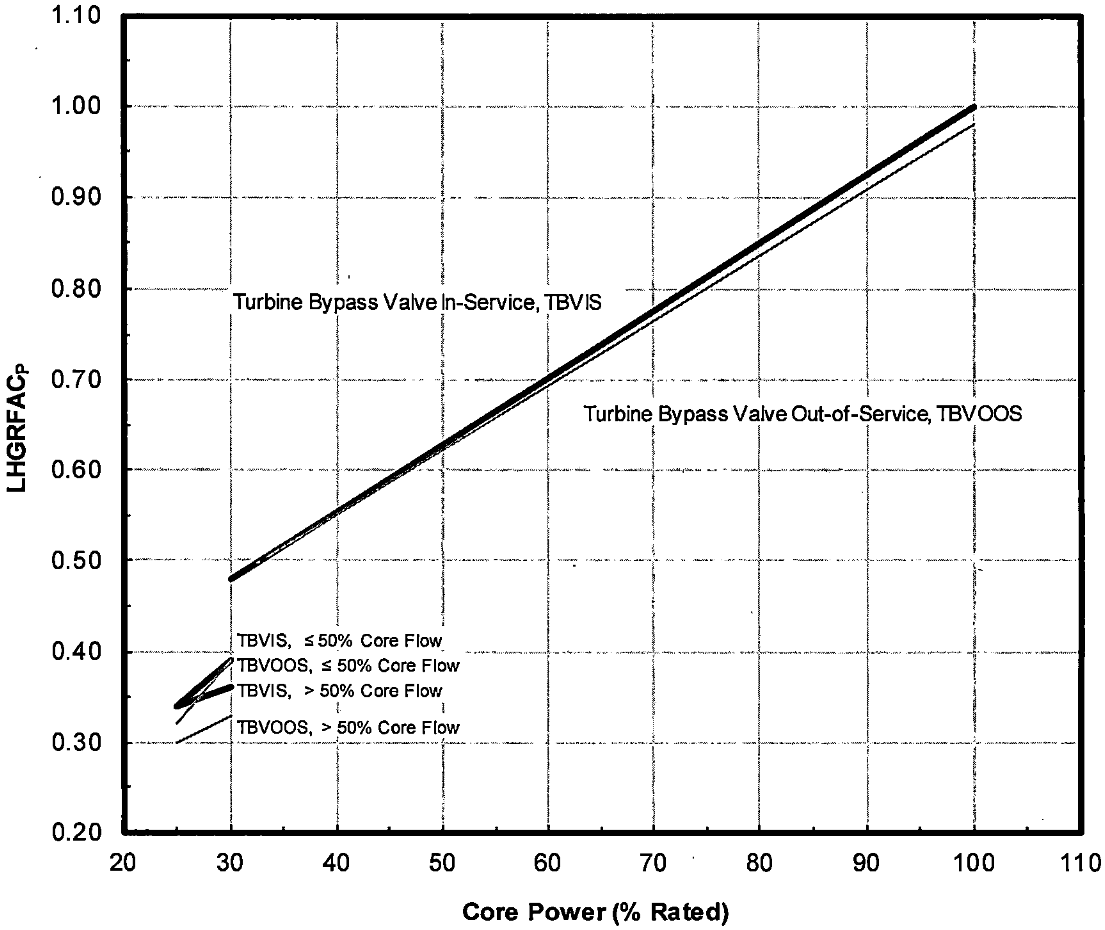
(no Feedwater heating during startup)

(Limits valid at and below 50% power)



Turbine Bypass In-Service Core Power (% Rated)	$LHGRFAC_P$	*Turbine Bypass Out-of-Service* Core Power (% Rated)	$LHGRFAC_P$
100.0	1.00	100.0	0.98
30.0	0.48	30.0	0.48
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.36	30.0	0.33
25.0	0.34	25.0	0.30
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.40	30.0	0.39
25.0	0.35	25.0	0.32

Figure 3.14 Startup Operation $LHGRFAC_P$ for ATRIUM-11 Fuel:
Table 3.1 Temperature Range 1
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



Turbine Bypass In-Service		*Turbine Bypass Out-of-Service*	
Core Power	**LHGRFAC$_P$**	**Core Power**	**LHGRFAC$_P$**
(% Rated)		**(% Rated)**	
100.0	1.00	100.0	0.98
30.0	0.48	30.0	0.48
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.36	30.0	0.33
25.0	0.34	25.0	0.30
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.39	30.0	0.39
25.0	0.34	25.0	0.32

Figure 3.15 Startup Operation LHGRFAC$_P$ for ATRIUM-11 Fuel:
Table 3.1 Temperature Range 2
(no Feedwater heating during startup)
(Limits valid at and below 50% power)

4 OLMCPR Limits

(Technical Specification 3.2.2, 3.3.4.1, & 3.7.5)

OLMCPR is calculated to be the most limiting of the flow or power dependent values

$$\text{OLMCPR limit} = \text{MAX}(\text{MCPR}_F, \text{MCPR}_P)$$

where:

$MCPR_F$	core flow-dependent MCPR limit
$MCPR_P$	power-dependent MCPR limit

4.1 Flow Dependent MCPR Limit: $MCPR_F$

$MCPR_F$ limits are dependent upon core flow (% of Rated), and the max core flow limit, (Rated or Increased Core Flow, ICF). $MCPR_F$ limits are shown in Figure 4.1, per Reference 1. Limits are valid for all EOOS combinations. No adjustment is required for SLO conditions.

4.2 Power Dependent MCPR Limit: $MCPR_P$

$MCPR_P$ limits are dependent upon:

- Core Power Level (% of Rated)
- Technical Specification Scram Speed (TSSS), Nominal Scram Speed (NSS), or Optimum Scram Speed (OSS)
- Cycle Operating Exposure (NEOC, EOC, and CD - as defined in this section)
- Equipment Out-Of-Service Options
- Two or Single recirculation Loop Operation (TLO vs. SLO)

The $MCPR_P$ limits are provided in Table 4.2 through Table 4.9, where each table contains the limits for all fuel types and EOOS options (for a specified scram speed and exposure range). The CMSS determines $MCPR_P$ limits, from these tables, based on linear interpolation between the specified powers.

4.2.1 <u>Startup without Feedwater Heaters</u>

There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. Additional power dependent limits are shown in Table 4.5 through Table 4.8 based on temperature conditions identified in Table 3.1.

4.2.2 Scram Speed Dependent Limits (TSSS vs. NSS vs. OSS)

MCPRP limits are provided for three different sets of assumed scram speeds. The Technical Specification Scram Speed (TSSS) MCPRP limits are applicable at all times, as long as the scram time surveillance demonstrates the times in Technical Specification Table 3.1.4-1 are met. Both Nominal Scram Speeds (NSS) and/or Optimum Scram Speeds (OSS) may be used, as long as the scram time surveillance demonstrates Table 4.1 times are applicable. * †

Table 4.1 Nominal Scram Time Basis

Notch Position	Nominal Scram Timing	Optimum Scram Timing
(index)	(seconds)	(seconds)
46	0.420	0.380
36	0.980	0.875
26	1.600	1.465
6	2.900	2.900

In demonstrating compliance with the NSS and/or OSS scram time basis, surveillance requirements from Technical Specification 3.1.4 apply; accepting the definition of SLOW rods should conform to scram speeds shown in Table 4.1. If conformance is not demonstrated, TSSS based MCPRP limits are applied.

On initial cycle startup, TSSS limits are used until the successful completion of scram timing confirms NSS and/or OSS based limits are applicable.

4.2.3 Exposure Dependent Limits

Exposures are tracked on a Core Average Exposure basis (CAVEX, not Cycle Exposure). Higher exposure MCPRP limits are always more limiting and may be used for any Core Average Exposure up to the ending exposure. Per Reference 1, MCPRP limits are provided for the following exposure ranges:

BOC to NEOC	NEOC corresponds to	**31,004.5 MWd / MTU**
BOC to EOCLB	EOCLB corresponds to	**34,274.0 MWd / MTU**
BOC to End of Coast	End of Coast	**35,793.3 MWd / MTU**

NEOC refers to a Near EOC exposure point.

* Reference 1 analysis results are based on information identified in Reference 4.

† Drop out times consistent with method used to perform actual timing measurements (i.e., including pickup/dropout effects).

The EOCLB exposure point is not the true End-Of-Cycle exposure. Instead it corresponds to a licensing exposure window exceeding expected end-of-full-power-life.

The End of Coast exposure point represents a licensing exposure point exceeding the expected end-of-cycle exposure including cycle extension options.

4.2.4 Equipment Out-Of-Service (EOOS) Options

EOOS options * covered by $MCPR_P$ limits are given by the following:

In-Service	All equipment In-Service
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
RPTOOS+TBVOOS	Combined RPTOOS and TBVOOS
PLUOOS	Power Load Unbalance Out-Of-Service
PLUOOS+RPTOOS	Combined PLUOOS and RPTOOS
PLUOOS+TBVOOS	Combined PLUOOS and TBVOOS
PLUOOS+TBVOOS+RPTOOS	Combined PLUOOS, RPTOOS, and TBVOOS
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service (or Final Feedwater Temperature Reduction)
RCPOOS	One Recirculation Pump Out-Of-Service

For exposure ranges up to NEOC and EOCLB, additional combinations of $MCPR_P$ limits are also provided including FHOOS. The coast down exposure range assumes application of FFWTR. FHOOS based $MCPR_P$ limits for the coast down exposure are redundant because the temperature setdown assumption is identical with FFWTR.

4.2.5 Single-Loop-Operation (SLO) Limits

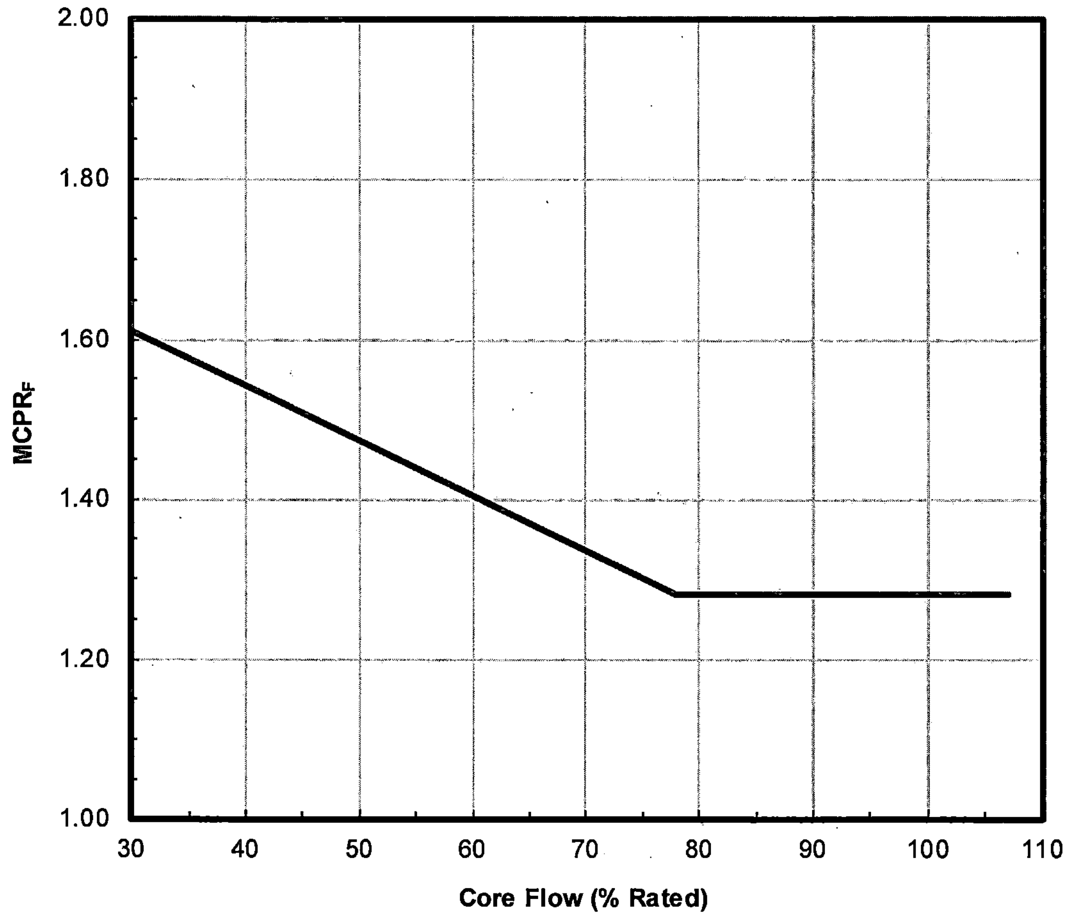
When operating in RCPOOS conditions, MCPRp limits are constructed differently from the normal operating RCP conditions. The limiting event for RCPOOS is a pump seizure scenario, which sets the upper bound for allowed core power and flow †. This event is not impacted by scram time assumptions. Specific $MCPR_P$ limits are shown in Table 4.9.

4.2.6 Below Pbypass Limits

Below Pbypass (30% rated power), $MCPR_P$ limits depend upon core flow. One set of $MCPR_P$ limits applies for core flow above 50% of rated; a second set applies if the core flow is less than or equal to 50% rated.

* All equipment service conditions assume 1 SRVOOS.

† RCPOOS limits are only valid up to 50% rated core power, 50% rated core flow, and an active recirculation drive flow of 17.73 Mlb_m/hr.



Core Flow (% Rated)	$MCPR_F$
30.0	1.61
78.0	1.28
107.0	1.28

Figure 4.1 $MCPR_F$ for All Fuel Types
(Values bound all EOOS conditions)

(107.0% maximum core flowline is used to support 105% rated flow operation, ICF)

Table 4.2 MCPRp Limits for All Fuel Types: Optimum Scram Time Basis *

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
Base Case	100	1.49	1.50	1.52	1.40	1.43	1.45	1.44	1.47	1.49
	75	1.62	1.62	1.68	1.53	1.53	1.57	1.58	1.58	1.63
	65	1.70	1.70	1.76	1.60	1.61	1.65	1.66	1.66	1.71
	50	1.85	1.85	1.93	1.75	1.75	1.81	1.87	1.87	---
	50	1.98	1.98	1.99	1.81	1.81	1.82	1.94	1.94	1.96
	40	2.09	2.09	2.20	1.94	1.94	2.03	2.13	2.13	2.25
	30	2.42	2.42	2.54	2.26	2.26	2.38	2.52	2.52	2.66
	30 at > 50%F	2.52	2.52	2.62	2.49	2.49	2.58	2.59	2.59	2.71
	25 at > 50%F	2.75	2.75	2.88	2.74	2.74	2.86	2.85	2.85	2.99
	30 at ≤ 50%F	2.43	2.43	2.54	2.27	2.27	2.38	2.56	2.56	2.67
	25 at ≤ 50%F	2.64	2.64	2.74	2.48	2.48	2.60	2.75	2.75	2.93
FHOOS	100	1.51	1.52	---	1.43	1.45	---	1.47	1.49	---
	75	1.68	1.68	---	1.57	1.57	---	1.63	1.63	---
	65	1.76	1.76	---	1.65	1.65	---	1.71	1.71	---
	50	1.93	1.93	---	1.81	1.81	---	---	---	---
	50	1.99	1.99	---	1.82	1.82	---	1.96	1.96	---
	40	2.20	2.20	---	2.03	2.03	---	2.25	2.25	---
	30	2.54	2.54	---	2.38	2.38	---	2.66	2.66	---
	30 at > 50%F	2.62	2.62	---	2.58	2.58	---	2.71	2.71	---
	25 at > 50%F	2.88	2.88	---	2.86	2.86	---	2.99	2.99	---
	30 at ≤ 50%F	2.54	2.54	---	2.38	2.38	---	2.67	2.67	---
	25 at ≤ 50%F	2.74	2.74	---	2.60	2.60	---	2.93	2.93	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR/FHOOS is supported for the BOC to End of Coast limits.

Table 4.3 MCPRP Limits for All Fuel Types: Nominal Scram Time Basis *

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
Base Case	100	1.50	1.51	1.53	1.42	1.44	1.46	1.47	1.48	1.50
	75	1.64	1.64	1.69	1.55	1.55	1.59	1.60	1.60	1.64
	65	1.72	1.72	1.78	1.61	1.62	1.66	1.67	1.67	1.74
	50	1.88	1.88	1.98	1.76	1.76	---	1.91	1.91	---
	50	1.98	1.98	1.99	1.81	1.81	1.85	1.95	1.95	2.01
	40	2.13	2.13	2.25	1.97	1.97	2.07	2.18	2.18	2.29
	30	2.46	2.46	2.60	2.30	2.30	2.42	2.57	2.57	2.71
	30 at > 50%F	2.52	2.52	2.62	2.49	2.49	2.58	2.59	2.59	2.71
	25 at > 50%F	2.75	2.75	2.88	2.74	2.74	2.86	2.85	2.85	2.99
	30 at ≤ 50%F	2.46	2.46	2.60	2.30	2.30	2.42	2.57	2.57	2.71
	25 at ≤ 50%F	2.64	2.64	2.74	2.48	2.48	2.60	2.75	2.75	2.93
TBVOOS	100	1.55	1.57	1.58	1.45	1.48	1.49	1.52	1.53	1.55
	75	1.69	1.69	1.75	1.58	1.58	1.62	1.64	1.65	1.69
	65	1.78	1.78	1.83	1.65	1.66	1.70	1.72	1.72	1.77
	50	1.92	1.92	---	1.80	1.80	---	1.91	1.91	---
	50	1.98	1.98	2.00	1.81	1.81	1.86	1.95	1.95	2.01
	40	2.14	2.14	2.25	1.97	1.97	2.07	2.18	2.18	2.29
	30	2.46	2.46	2.60	2.30	2.30	2.42	2.57	2.57	2.71
	30 at > 50%F	3.15	3.15	3.28	2.96	2.96	3.07	3.15	3.15	3.27
	25 at > 50%F	3.51	3.51	3.61	3.34	3.34	3.45	3.53	3.53	3.63
	30 at ≤ 50%F	2.85	2.85	2.96	2.63	2.63	2.75	2.97	2.97	3.09
	25 at ≤ 50%F	3.23	3.23	3.37	3.02	3.02	3.18	3.38	3.38	3.54
FHOOS	100	1.53	1.53	---	1.45	1.46	---	1.49	1.50	---
	75	1.69	1.69	---	1.59	1.59	---	1.64	1.64	---
	65	1.78	1.78	---	1.66	1.66	---	1.74	1.74	---
	50	1.98	1.98	---	---	---	---	---	---	---
	50	1.99	1.99	---	1.85	1.85	---	2.01	2.01	---
	40	2.25	2.25	---	2.07	2.07	---	2.29	2.29	---
	30	2.60	2.60	---	2.42	2.42	---	2.71	2.71	---
	30 at > 50%F	2.62	2.62	---	2.58	2.58	---	2.71	2.71	---
	25 at > 50%F	2.88	2.88	---	2.86	2.86	---	2.99	2.99	---
	30 at ≤ 50%F	2.60	2.60	---	2.42	2.42	---	2.71	2.71	---
	25 at ≤ 50%F	2.74	2.74	---	2.60	2.60	---	2.93	2.93	---
PLUOOS	100	1.50	1.51	1.53	1.42	1.44	1.46	1.47	1.48	1.50
	75	1.64	1.64	1.69	1.55	1.55	1.59	1.60	1.60	1.64
	65	1.89	1.91	1.91	1.72	1.72	1.72	1.84	1.85	1.85
	50	---	---	---	---	---	---	---	---	---
	50	1.98	1.98	1.99	1.81	1.81	1.85	1.95	1.95	2.01
	40	2.13	2.13	2.25	1.97	1.97	2.07	2.18	2.18	2.29
	30	2.46	2.46	2.60	2.30	2.30	2.42	2.57	2.57	2.71
	30 at > 50%F	2.52	2.52	2.62	2.49	2.49	2.58	2.59	2.59	2.71
	25 at > 50%F	2.75	2.75	2.88	2.74	2.74	2.86	2.85	2.85	2.99
	30 at ≤ 50%F	2.46	2.46	2.60	2.30	2.30	2.42	2.57	2.57	2.71
	25 at ≤ 50%F	2.64	2.64	2.74	2.48	2.48	2.60	2.75	2.75	2.93

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.3 MCPR$_P$ Limits for All Fuel Types: Nominal Scram Time Basis (*continued*)*

Operating Condition	Power (% of rated)	ATRIUM-10			ATRIUM-10XM			ATRIUM-11		
		BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVOOS FHOOS	100	1.58	1.58	---	1.47	1.49	---	1.53	1.55	---
	75	1.75	1.75	---	1.62	1.62	---	1.69	1.69	---
	65	1.83	1.83	---	1.69	1.70	---	1.77	1.77	---
	50	---	---	---	---	---	---	---	---	---
	50	2.00	2.00	---	1.86	1.86	---	2.01	2.01	---
	40	2.25	2.25	---	2.07	2.07	---	2.29	2.29	---
	30	2.60	2.60	---	2.42	2.42	---	2.71	2.71	---
	30 at > 50%F	3.28	3.28	---	3.07	3.07	---	3.27	3.27	---
	25 at > 50%F	3.61	3.61	---	3.45	3.45	---	3.63	3.63	---
	30 at ≤ 50%F	2.96	2.96	---	2.75	2.75	---	3.09	3.09	---
	25 at ≤ 50%F	3.37	3.37	---	3.18	3.18	---	3.54	3.54	---
TBVOOS PLUOOS	100	1.55	1.57	1.58	1.45	1.48	1.49	1.52	1.53	1.55
	75	1.69	1.69	1.75	1.58	1.58	1.62	1.64	1.65	1.69
	65	1.89	1.91	1.91	1.72	1.72	1.72	1.84	1.85	1.85
	50	---	---	---	---	---	---	---	---	---
	50	1.98	1.98	2.00	1.81	1.81	1.86	1.95	1.95	2.01
	40	2.14	2.14	2.25	1.97	1.97	2.07	2.18	2.18	2.29
	30	2.46	2.46	2.60	2.30	2.30	2.42	2.57	2.57	2.71
	30 at > 50%F	3.15	3.15	3.28	2.96	2.96	3.07	3.15	3.15	3.27
	25 at > 50%F	3.51	3.51	3.61	3.34	3.34	3.45	3.53	3.53	3.63
	30 at ≤ 50%F	2.85	2.85	2.96	2.63	2.63	2.75	2.97	2.97	3.09
	25 at ≤ 50%F	3.23	3.23	3.37	3.02	3.02	3.18	3.38	3.38	3.54
FHOOS PLUOOS	100	1.53	1.53	---	1.45	1.46	---	1.49	1.50	---
	75	1.69	1.69	---	1.59	1.59	---	1.64	1.64	---
	65	1.89	1.91	---	1.72	1.72	---	1.84	1.85	---
	50	---	---	---	---	---	---	---	---	---
	50	1.99	1.99	---	1.85	1.85	---	2.01	2.01	---
	40	2.25	2.25	---	2.07	2.07	---	2.29	2.29	---
	30	2.60	2.60	---	2.42	2.42	---	2.71	2.71	---
	30 at > 50%F	2.62	2.62	---	2.58	2.58	---	2.71	2.71	---
	25 at > 50%F	2.88	2.88	---	2.86	2.86	---	2.99	2.99	---
	30 at ≤ 50%F	2.60	2.60	---	2.42	2.42	---	2.71	2.71	---
	25 at ≤ 50%F	2.74	2.74	---	2.60	2.60	---	2.93	2.93	---
TBVOOS FHOOS PLUOOS	100	1.58	1.58	---	1.47	1.49	---	1.53	1.55	---
	75	1.75	1.75	---	1.62	1.62	---	1.69	1.69	---
	65	1.89	1.91	---	1.72	1.72	---	1.84	1.85	---
	50	---	---	---	---	---	---	---	---	---
	50	2.00	2.00	---	1.86	1.86	---	2.01	2.01	---
	40	2.25	2.25	---	2.07	2.07	---	2.29	2.29	---
	30	2.60	2.60	---	2.42	2.42	---	2.71	2.71	---
	30 at > 50%F	3.28	3.28	---	3.07	3.07	---	3.27	3.27	---
	25 at > 50%F	3.61	3.61	---	3.45	3.45	---	3.63	3.63	---
	30 at ≤ 50%F	2.96	2.96	---	2.75	2.75	---	3.09	3.09	---
	25 at ≤ 50%F	3.37	3.37	---	3.18	3.18	---	3.54	3.54	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.4 MCPRp Limits for All Fuel Types: Technical Specification Scram Time Basis *

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
Base Case	100	1.52	1.52	1.55	1.43	1.45	1.47	1.48	1.49	1.51
	75	1.65	1.65	1.70	1.56	1.56	1.60	1.61	1.61	1.67
	65	1.73	1.73	1.79	1.63	1.63	1.67	1.70	1.70	1.78
	50	1.93	1.93	---	1.80	1.80	---	---	---	---
	50	1.99	2.00	2.04	1.81	1.81	1.88	1.95	1.95	2.05
	40	2.18	2.18	2.31	2.01	2.01	2.11	2.22	2.22	2.34
	30	2.51	2.51	2.65	2.34	2.34	2.47	2.61	2.61	2.76
	30 at > 50%F	2.52	2.52	2.65	2.49	2.49	2.58	2.61	2.61	2.76
	25 at > 50%F	2.75	2.75	2.88	2.74	2.74	2.86	2.85	2.85	2.99
	30 at ≤ 50%F	2.51	2.51	2.65	2.34	2.34	2.47	2.61	2.61	2.76
	25 at ≤ 50%F	2.64	2.64	2.74	2.48	2.48	2.60	2.75	2.75	2.93
TBVOOS	100	1.57	1.58	1.60	1.47	1.49	1.50	1.54	1.55	1.56
	75	1.71	1.71	1.76	1.59	1.59	1.63	1.66	1.66	1.70
	65	1.79	1.79	1.85	1.66	1.67	1.71	1.74	1.74	1.79
	50	1.95	1.95	---	---	---	---	---	---	---
	50	1.99	2.00	2.05	1.81	1.81	1.88	1.95	1.95	2.05
	40	2.20	2.20	2.31	2.01	2.01	2.11	2.22	2.22	2.34
	30	2.51	2.51	2.65	2.34	2.34	2.47	2.61	2.61	2.76
	30 at > 50%F	3.15	3.15	3.28	2.96	2.96	3.07	3.15	3.15	3.27
	25 at > 50%F	3.51	3.51	3.61	3.34	3.34	3.45	3.53	3.53	3.63
	30 at ≤ 50%F	2.85	2.85	2.96	2.63	2.63	2.75	2.97	2.97	3.09
	25 at ≤ 50%F	3.23	3.23	3.37	3.02	3.02	3.18	3.38	3.38	3.54
FHOOS	100	1.55	1.55	---	1.46	1.47	---	1.51	1.51	---
	75	1.70	1.70	---	1.60	1.60	---	1.67	1.67	---
	65	1.79	1.79	---	1.67	1.67	---	1.78	1.78	---
	50	---	---	---	---	---	---	---	---	---
	50	2.04	2.04	---	1.88	1.88	---	2.05	2.05	---
	40	2.31	2.31	---	2.11	2.11	---	2.34	2.34	---
	30	2.65	2.65	---	2.47	2.47	---	2.76	2.76	---
	30 at > 50%F	2.65	2.65	---	2.58	2.58	---	2.76	2.76	---
	25 at > 50%F	2.88	2.88	---	2.86	2.86	---	2.99	2.99	---
	30 at ≤ 50%F	2.65	2.65	---	2.47	2.47	---	2.76	2.76	---
	25 at ≤ 50%F	2.74	2.74	---	2.60	2.60	---	2.93	2.93	---
PLUOOS	100	1.52	1.52	1.55	1.43	1.45	1.47	1.48	1.49	1.51
	75	1.65	1.65	1.70	1.56	1.56	1.60	1.61	1.61	1.67
	65	1.90	1.93	1.93	1.73	1.74	1.74	1.85	1.87	1.87
	50	---	---	---	---	---	---	---	---	---
	50	1.99	2.00	2.04	1.82	1.82	1.88	1.95	1.95	2.05
	40	2.18	2.18	2.31	2.01	2.01	2.11	2.22	2.22	2.34
	30	2.51	2.51	2.65	2.34	2.34	2.47	2.61	2.61	2.76
	30 at > 50%F	2.52	2.52	2.65	2.49	2.49	2.58	2.61	2.61	2.76
	25 at > 50%F	2.75	2.75	2.88	2.74	2.74	2.86	2.85	2.85	2.99
	30 at ≤ 50%F	2.51	2.51	2.65	2.34	2.34	2.47	2.61	2.61	2.76
	25 at ≤ 50%F	2.64	2.64	2.74	2.48	2.48	2.60	2.75	2.75	2.93

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.4 MCPR$_P$ Limits for All Fuel Types: Technical Specification Scram Time Basis (*continued*) *

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
TBVOOS FHOOS	100	1.60	1.60	---	1.49	1.50	---	1.55	1.56	---
	75	1.76	1.76	---	1.63	1.63	---	1.70	1.70	---
	65	1.85	1.85	---	1.71	1.71	---	1.79	1.79	---
	50	---	---	---	---	---	---	---	---	---
	50	2.05	2.05	---	1.88	1.88	---	2.05	2.05	---
	40	2.31	2.31	---	2.11	2.11	---	2.34	2.34	---
	30	2.65	2.65	---	2.47	2.47	---	2.76	2.76	---
	30 at > 50%F	3.28	3.28	---	3.07	3.07	---	3.27	3.27	---
	25 at > 50%F	3.61	3.61	---	3.45	3.45	---	3.63	3.63	---
	30 at ≤ 50%F	2.96	2.96	---	2.75	2.75	---	3.09	3.09	---
	25 at ≤ 50%F	3.37	3.37	---	3.18	3.18	---	3.54	3.54	---
TBVOOS PLUOOS	100	1.57	1.58	1.60	1.47	1.49	1.50	1.54	1.55	1.56
	75	1.71	1.71	1.76	1.59	1.59	1.63	1.66	1.66	1.70
	65	1.90	1.93	1.93	1.73	1.74	1.74	1.85	1.87	1.87
	50	---	---	---	---	---	---	---	---	---
	50	1.99	2.00	2.05	1.82	1.82	1.88	1.95	1.95	2.05
	40	2.20	2.20	2.31	2.01	2.01	2.11	2.22	2.22	2.34
	30	2.51	2.51	2.65	2.34	2.34	2.47	2.61	2.61	2.76
	30 at > 50%F	3.15	3.15	3.28	2.96	2.96	3.07	3.15	3.15	3.27
	25 at > 50%F	3.51	3.51	3.61	3.34	3.34	3.45	3.53	3.53	3.63
	30 at ≤ 50%F	2.85	2.85	2.96	2.63	2.63	2.75	2.97	2.97	3.09
	25 at ≤ 50%F	3.23	3.23	3.37	3.02	3.02	3.18	3.38	3.38	3.54
FHOOS PLUOOS	100	1.55	1.55	---	1.46	1.47	---	1.51	1.51	---
	75	1.70	1.70	---	1.60	1.60	---	1.67	1.67	---
	65	1.90	1.93	---	1.73	1.74	---	1.85	1.87	---
	50	---	---	---	---	---	---	---	---	---
	50	2.04	2.04	---	1.88	1.88	---	2.05	2.05	---
	40	2.31	2.31	---	2.11	2.11	---	2.34	2.34	---
	30	2.65	2.65	---	2.47	2.47	---	2.76	2.76	---
	30 at > 50%F	2.65	2.65	---	2.58	2.58	---	2.76	2.76	---
	25 at > 50%F	2.88	2.88	---	2.86	2.86	---	2.99	2.99	---
	30 at ≤ 50%F	2.65	2.65	---	2.47	2.47	---	2.76	2.76	---
	25 at ≤ 50%F	2.74	2.74	---	2.60	2.60	---	2.93	2.93	---
TBVOOS FHOOS PLUOOS	100	1.60	1.60	---	1.49	1.50	---	1.55	1.56	---
	75	1.76	1.76	---	1.63	1.63	---	1.70	1.70	---
	65	1.90	1.93	---	1.73	1.74	---	1.85	1.87	---
	50	---	---	---	---	---	---	---	---	---
	50	2.05	2.05	---	1.88	1.88	---	2.05	2.05	---
	40	2.31	2.31	---	2.11	2.11	---	2.34	2.34	---
	30	2.65	2.65	---	2.47	2.47	---	2.76	2.76	---
	30 at > 50%F	3.28	3.28	---	3.07	3.07	---	3.27	3.27	---
	25 at > 50%F	3.61	3.61	---	3.45	3.45	---	3.63	3.63	---
	30 at ≤ 50%F	2.96	2.96	---	2.75	2.75	---	3.09	3.09	---
	25 at ≤ 50%F	3.37	3.37	---	3.18	3.18	---	3.54	3.54	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.5 Startup Operation MCPRp Limits for Table 3.1 Temperature Range 1 for All Fuel Types: Nominal Scram Time Basis *

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
TBVIS	100	1.53	1.53	1.53	1.45	1.46	1.46	1.49	1.50	1.50
	75	1.69	1.69	1.69	1.59	1.59	1.59	1.64	1.64	1.64
	65	1.89	1.91	1.91	1.72	1.72	1.72	1.84	1.85	1.85
	50	---	---	---	---	---	---	---	---	---
	50	2.20	2.20	2.20	2.03	2.03	2.03	2.25	2.25	2.25
	40	2.52	2.52	2.52	2.29	2.29	2.29	2.57	2.57	2.57
	30	2.92	2.92	2.92	2.72	2.72	2.72	3.05	3.05	3.05
	30 at > 50%F	2.92	2.92	2.92	2.82	2.82	2.82	3.05	3.05	3.05
	25 at > 50%F	3.21	3.21	3.21	3.16	3.16	3.16	3.34	3.34	3.34
	30 at ≤ 50%F	2.92	2.92	2.92	2.72	2.72	2.72	3.05	3.05	3.05
	25 at ≤ 50%F	3.15	3.15	3.15	2.88	2.88	2.88	3.25	3.25	3.25
TBVOOS	100	1.58	1.58	1.58	1.47	1.49	1.49	1.53	1.55	1.55
	75	1.75	1.75	1.75	1.62	1.62	1.62	1.69	1.69	1.69
	65	1.89	1.91	1.91	1.72	1.72	1.72	1.84	1.85	1.85
	50	---	---	---	---	---	---	---	---	---
	50	2.20	2.20	2.20	2.03	2.03	2.03	2.25	2.25	2.25
	40	2.52	2.52	2.52	2.29	2.29	2.29	2.57	2.57	2.57
	30	2.92	2.92	2.92	2.72	2.72	2.72	3.05	3.05	3.05
	30 at > 50%F	3.47	3.47	3.47	3.25	3.25	3.25	3.46	3.46	3.46
	25 at > 50%F	3.84	3.84	3.84	3.66	3.66	3.66	3.84	3.84	3.84
	30 at ≤ 50%F	3.15	3.15	3.15	2.97	2.97	2.97	3.39	3.39	3.39
	25 at ≤ 50%F	3.59	3.59	3.59	3.43	3.43	3.43	3.81	3.81	3.81

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.6 Startup Operation MCPRp Limits for Table 3.1 Temperature Range 2 for All Fuel Types: Nominal Scram Time Basis *

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
TBVIS	100	1.53	1.53	1.53	1.45	1.46	1.46	1.49	1.50	1.50
	75	1.69	1.69	1.69	1.59	1.59	1.59	1.64	1.64	1.64
	65	1.89	1.91	1.91	1.72	1.72	1.72	1.84	1.85	1.85
	50	---	---	---	---	---	---	---	---	---
	50	2.22	2.22	2.22	2.04	2.04	2.04	2.26	2.26	2.26
	40	2.54	2.54	2.54	2.30	2.30	2.30	2.59	2.59	2.59
	30	2.95	2.95	2.95	2.74	2.74	2.74	3.07	3.07	3.07
	30 at > 50%F	2.96	2.96	2.96	2.83	2.83	2.83	3.07	3.07	3.07
	25 at > 50%F	3.23	3.23	3.23	3.18	3.18	3.18	3.37	3.37	3.37
	30 at ≤ 50%F	2.95	2.95	2.95	2.78	2.78	2.78	3.07	3.07	3.07
	25 at ≤ 50%F	3.19	3.19	3.19	3.12	3.12	3.12	3.27	3.27	3.27
TBVOOS	100	1.58	1.58	1.58	1.47	1.49	1.49	1.53	1.55	1.55
	75	1.75	1.75	1.75	1.62	1.62	1.62	1.69	1.69	1.69
	65	1.89	1.91	1.91	1.72	1.72	1.72	1.84	1.85	1.85
	50	---	---	---	---	---	---	---	---	---
	50	2.22	2.22	2.22	2.04	2.04	2.04	2.26	2.26	2.26
	40	2.54	2.54	2.54	2.30	2.30	2.30	2.59	2.59	2.59
	30	2.95	2.95	2.95	2.74	2.74	2.74	3.07	3.07	3.07
	30 at > 50%F	3.48	3.48	3.48	3.27	3.27	3.27	3.47	3.47	3.47
	25 at > 50%F	3.85	3.85	3.85	3.68	3.68	3.68	3.86	3.86	3.86
	30 at ≤ 50%F	3.16	3.16	3.16	3.15	3.15	3.15	3.41	3.41	3.41
	25 at ≤ 50%F	3.60	3.60	3.60	3.68	3.68	3.68	3.86	3.86	3.86

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.7 Startup Operation MCPRp Limits for Table 3.1 Temperature Range 1 for All Fuel Types: Technical Specification Scram Time Basis *

Operating Condition	Power (% of rated)	ATRIUM-10			ATRIUM-10XM			ATRIUM-11		
		BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVIS	100	1.55	1.55	1.55	1.46	1.47	1.47	1.51	1.51	1.51
	75	1.70	1.70	1.70	1.60	1.60	1.60	1.67	1.67	1.67
	65	1.90	1.93	1.93	1.73	1.74	1.74	1.85	1.87	1.87
	50	---	---	---	---	---	---	---	---	---
	50	2.26	2.26	2.26	2.07	2.07	2.07	2.30	2.30	2.30
	40	2.58	2.58	2.58	2.33	2.33	2.33	2.62	2.62	2.62
	30	2.98	2.98	2.98	2.77	2.77	2.77	3.10	3.10	3.10
	30 at > 50%F	2.98	2.98	2.98	2.82	2.82	2.82	3.10	3.10	3.10
	25 at > 50%F	3.21	3.21	3.21	3.16	3.16	3.16	3.34	3.34	3.34
	30 at ≤ 50%F	2.98	2.98	2.98	2.77	2.77	2.77	3.10	3.10	3.10
	25 at ≤ 50%F	3.15	3.15	3.15	2.88	2.88	2.88	3.25	3.25	3.25
TBVOOS	100	1.60	1.60	1.60	1.49	1.50	1.50	1.55	1.56	1.56
	75	1.76	1.76	1.76	1.63	1.63	1.63	1.70	1.70	1.70
	65	1.90	1.93	1.93	1.73	1.74	1.74	1.85	1.87	1.87
	50	---	---	---	---	---	---	---	---	---
	50	2.26	2.26	2.26	2.07	2.07	2.07	2.30	2.30	2.30
	40	2.58	2.58	2.58	2.33	2.33	2.33	2.62	2.62	2.62
	30	2.98	2.98	2.98	2.77	2.77	2.77	3.10	3.10	3.10
	30 at > 50%F	3.47	3.47	3.47	3.25	3.25	3.25	3.46	3.46	3.46
	25 at > 50%F	3.84	3.84	3.84	3.66	3.66	3.66	3.84	3.84	3.84
	30 at ≤ 50%F	3.15	3.15	3.15	2.97	2.97	2.97	3.39	3.39	3.39
	25 at ≤ 50%F	3.59	3.59	3.59	3.43	3.43	3.43	3.81	3.81	3.81

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.8 Startup Operation MCPRp Limits for Table 3.1 Temperature Range 2 for All Fuel Types: Technical Specification Scram Time Basis *

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast	ATRIUM-11 BOC to NEOC	ATRIUM-11 BOC to EOCLB	ATRIUM-11 BOC to End of Coast
TBVIS	100	1.55	1.55	1.55	1.46	1.47	1.47	1.51	1.51	1.51
	75	1.70	1.70	1.70	1.60	1.60	1.60	1.67	1.67	1.67
	65	1.90	1.93	1.93	1.73	1.74	1.74	1.85	1.87	1.87
	50	---	---	---	---	---	---	---	---	---
	50	2.28	2.28	2.28	2.08	2.08	2.08	2.31	2.31	2.31
	40	2.60	2.60	2.60	2.35	2.35	2.35	2.63	2.63	2.63
	30	3.01	3.01	3.01	2.79	2.79	2.79	3.13	3.13	3.13
	30 at > 50%F	3.01	3.01	3.01	2.83	2.83	2.83	3.13	3.13	3.13
	25 at > 50%F	3.23	3.23	3.23	3.18	3.18	3.18	3.37	3.37	3.37
	30 at ≤ 50%F	3.01	3.01	3.01	2.79	2.79	2.79	3.13	3.13	3.13
	25 at ≤ 50%F	3.19	3.19	3.19	3.12	3.12	3.12	3.27	3.27	3.27
TBVOOS	100	1.60	1.60	1.60	1.49	1.50	1.50	1.55	1.56	1.56
	75	1.76	1.76	1.76	1.63	1.63	1.63	1.70	1.70	1.70
	65	1.90	1.93	1.93	1.73	1.74	1.74	1.85	1.87	1.87
	50	---	---	---	---	---	---	---	---	---
	50	2.28	2.28	2.28	2.08	2.08	2.08	2.31	2.31	2.31
	40	2.60	2.60	2.60	2.35	2.35	2.35	2.63	2.63	2.63
	30	3.01	3.01	3.01	2.79	2.79	2.79	3.13	3.13	3.13
	30 at > 50%F	3.48	3.48	3.48	3.27	3.27	3.27	3.47	3.47	3.47
	25 at > 50%F	3.85	3.85	3.85	3.68	3.68	3.68	3.86	3.86	3.86
	30 at ≤ 50%F	3.16	3.16	3.16	3.15	3.15	3.15	3.41	3.41	3.41
	25 at ≤ 50%F	3.60	3.60	3.60	3.68	3.68	3.68	3.86	3.86	3.86

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.9 MCPRp Limits for All Fuel Types: Single Loop Operation for All Scram Times *

Operating Condition	Power (% of rated)	BOC to End of COAST		
		ATRIUM-10	ATRIUM-10XM	ATRIUM-11
RCPOOS FHOOS	100	2.06	2.04	2.07
	50	2.06	2.04	2.07
	40	2.33	2.13	2.36
	30	2.67	2.49	2.78
	30 at > 50%F	2.67	2.60	2.78
	25 at > 50%F	2.90	2.88	3.01
	30 at ≤ 50%F	2.67	2.49	2.78
	25 at ≤ 50%F	2.76	2.62	2.95
RCPOOS TBVOOS PLUOOS FHOOS	100	2.07	2.04	2.07
	50	2.07	2.04	2.07
	40	2.33	2.13	2.36
	30	2.67	2.49	2.78
	30 at > 50%F	3.30	3.09	3.29
	25 at > 50%F	3.63	3.47	3.65
	30 at ≤ 50%F	2.98	2.77	3.11
	25 at ≤ 50%F	3.39	3.20	3.56
RCPOOS TBVOOS FHOOS1	100	2.28	2.09	2.32
	50	2.28	2.09	2.32
	40	2.60	2.35	2.64
	30	3.00	2.79	3.12
	30 at > 50%F	3.49	3.27	3.48
	25 at > 50%F	3.86	3.68	3.86
	30 at ≤ 50%F	3.17	2.99	3.41
	25 at ≤ 50%F	3.61	3.45	3.83
RCPOOS TBVOOS FHOOS2	100	2.30	2.10	2.33
	50	2.30	2.10	2.33
	40	2.62	2.37	2.65
	30	3.03	2.81	3.15
	30 at > 50%F	3.50	3.29	3.49
	25 at > 50%F	3.87	3.70	3.88
	30 at ≤ 50%F	3.18	3.17	3.43
	25 at ≤ 50%F	3.62	3.70	3.88

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop.

RCPOOS limits are only valid up to 50% rated core power, 50% rated core flow, and an active recirculation drive flow of 17.73 Mlbm/hr.

5 Oscillation Power Range Monitor (OPRM) Setpoint

(Technical Specification 3.3.1.1)

Technical Specification Table 3.3.1.1-1, Function 2f, identifies the OPRM upscale function.

Instrument setpoints are established, such that the reactor will be tripped before an oscillation can grow to the point where the SLMCPR is exceeded. An Option III stability analysis is performed for each reload core to determine allowable OLMCPR's as a function of OPRM setpoint. Analyses consider both steady state startup operation, and the case of a two recirculation pump trip from rated power.

The resulting stability based OLMCPR's are reported in Reference 1. The OPRM setpoint (*sometimes referred to as the Amplitude Trip, S_p*) is selected, such that required margin to the SLMCPR is provided without stability being a limiting event. Analyses are based on cycle specific DIVOM analyses performed per Reference 22. The calculated OLMCPR's are shown in Table 5.1. Review of results shown in COLR Table 4.2 indicates an OPRM setpoint of **1.14** may be used. The successive confirmation count (*sometimes referred to as N_p*) is provided in Table 5.2, per Reference 30.

Table 5.1 OPRM Setpoint Range *

OPRM Setpoint	OLMCPR (SS)	OLMCPR (2PT)
1.05	1.15	1.18
1.06	1.17	1.20
1.07	1.19	1.22
1.08	1.20	1.24
1.09	1.22	1.26
1.10	1.24	1.28
1.11	1.26	1.30
1.12	1.28	1.32
1.13	1.30	1.34
1.14	1.33	1.36
1.15	1.35	1.38

Table 5.2 OPRM Successive Confirmation Count Setpoint

Count	OPRM Setpoint
6	≥ 1.04
8	≥ 1.05
10	≥ 1.07
12	≥ 1.09
14	≥ 1.11
16	≥ 1.14
18	≥ 1.18
20	≥ 1.24

* Extrapolation beyond a setpoint of 1.15 is not allowed

6 APRM Flow Biased Rod Block Trip Settings

(Technical Requirements Manual Section 5.3.1 and Table 3.3.4-1)

The APRM rod block trip setting is based upon References 26 & 27, and is defined by the following:

SRB $\leq$ $(0.66(W-\Delta W) + 61\%)$	Allowable Value
SRB $\leq$ $(0.66(W-\Delta W) + 59\%)$	Nominal Trip Setpoint (NTSP)

where:

SRB	=	Rod Block setting in percent of rated thermal power (3458 MW_t)
W	=	Loop recirculation flow rate in percent of rated
ΔW	=	Difference between two-loop and single-loop effective recirculation flow at the same core flow (ΔW=0.0 for two-loop operation)

The APRM rod block trip setting is clamped at a maximum allowable value of 115% (corresponding to a NTSP of 113%).

7 Rod Block Monitor (RBM) Trip Setpoints and Operability (Technical Specification Table 3.3.2.1-1)

The RBM trip setpoints and applicable power ranges, based on References 26 & 27, are shown in Table 7.1. Setpoints are based on an HTSP, unfiltered analytical limit of 114%. Unfiltered setpoints are consistent with a nominal RBM filter setting of 0.0 seconds; filtered setpoints are consistent with a nominal RBM filter setting less than 0.5 seconds. Cycle specific CRWE analyses of OLMCPR are documented in Reference 1, superseding values reported in References 26, 27, and 29.

Table 7.1 Analytical RBM Trip Setpoints *

RBM Trip Setpoint	Allowable Value (AV)	Nominal Trip Setpoint (NTSP)
LPSP	27%	25%
IPSP	62%	60%
HPSP	82%	80%
LTSP - unfiltered - filtered	121.7% 120.7%	120.0% 119.0%
ITSP - unfiltered - filtered	116.7% 115.7%	115.0% 114.0%
HTSP - unfiltered - filtered	111.7% 110.9%	110.0% 109.2%
DTSP	90%	92%

As a result of cycle specific CRWE analyses, RBM setpoints in Technical Specification Table 3.3.2.1-1 are applicable as shown in Table 7.2. Cycle specific setpoint analysis results are shown in Table 7.3, per Reference 1.

Table 7.2 RBM Setpoint Applicability

Thermal Power (% Rated)	Applicable MCPR †	Notes from Table 3.3.2.1-1	Comment
> 27% and < 90%	< 1.67	(a), (b), (f), (h)	two loop operation
	< 1.71	(a), (b), (f), (h)	single loop operation
≥ 90%	< 1.37	(g)	two loop operation ‡

* Values are considered maximums. Using lower values, due to RBM system hardware/software limitations, is conservative, and acceptable.

† MCPR values shown correspond with, (support), SLMPCR values identified in Reference 1.

‡ Greater than 90% rated power is not attainable in single loop operation.

Table 7.3 Control Rod Withdrawal Error Results

RBM HTSP Analytical Limit	CRWE OLMCPR
Unfiltered	
107	1.29
111	1.32
114	1.35
117	1.35

Results, compared against the base case OLMCPR results of Table 4.2, indicate SLMCPR remains protected for RBM inoperable conditions (i.e., 114% unblocked).

8 Shutdown Margin Limit

(Technical Specification 3.1.1)

Assuming the strongest OPERABLE control blade is fully withdrawn, and all other OPERABLE control blades are fully inserted, the core shall be sub-critical and meet the following minimum shutdown margin:

SDM > 0.38% dk/k

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B 3.9-18	Revision 52	06-24-2005

ACTIONS (continued)

C.1 and C.2

When required control room indication channels are inoperable but the redundant channels for the parameters are still OPERABLE, the required control room indication channels must be returned to OPERABLE status in 30 days (Required Action C.2). However, if both redundant channels for one or more of the associated parameters are not indicating in the control room, the 30 day allowed out of service time is not acceptable and one of the required control room indication channels for each associated parameter must be returned to OPERABLE status in 7 days (Required Action C.1).

D.1, D.2, and D.3

When the Tailpipe Thermocouple Temperature and Acoustic Monitor is inoperable for one or more Main Steam Relief Valves (MSRVs), the torus temperature must be observed once per 12 hours to observe any unexplained temperature increase which might be indicative of an open relief valve (Required Action D.1) and control room indication by either the Tailpipe Thermocouple Temperature or Acoustic Monitor must be returned to OPERABLE status for each relief valve in 30 days (Required Action D.2). The condition must be entered into the Corrective Action Program within 24 hours if control room indication is not restored in 30 days (Required Action D.3).

E.1.1 and E.1.2

When the Wide Range Gaseous Effluent Radiation Monitor and Recorder instrument channel is inoperable, either the inoperable channel must be returned to OPERABLE status in 72 hours (Required Action E.1.1), or the preplanned alternate method of monitoring the parameter must be initiated (Required Action E.1.2). A note is provided to indicate that Required Actions E.1.1 and E.1.2 are not applicable when in MODES 4 and 5.

E.2

The condition must be entered into the Corrective Action Program within 24 hours after the Wide Range Gaseous Effluent Radiation Monitor and Recorder instrument channel has been inoperable for 7 days.

BASES

LCO 3.3.7

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation dose to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public.

APPLICABILITY

At all times.

ACTIONS

A.1

Seven days to obtain replacement parts and repair is reasonable for these instruments given the requirements to be available for radiological emergencies.

B.1

If the instruments cannot be repaired in the allowed time frame, the condition must be entered into the Corrective Action Program within 24 hours.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.3.7.1

Daily checks of these parameters assures prompt replacement/repair of inoperable or questionable instruments.

TSR 3.3.7.2

Surveillance requirement times are based on equipment reliability and engineering judgment and conservatively set to provide adequate assurance of instrument performance.

ACTIONS (continued)

B.1 (continued)

the plant conditions at the locations of these instruments to determine the impact of the vibratory ground motion on structures and equipment in these locations following any required unit shutdown after a seismic event.

C.1

If any Required Action and association Completion Time of Condition A or B is not met, the failure to restore the inoperable seismic monitoring instrumentation within the required Completion Time must be entered into the Corrective Action Program within 24 hours.

TECHNICAL SURVEILLANCE REQUIREMENTS

The TSRs for each seismic monitoring Function are identified by the Technical Surveillance Requirements column of Table 3.3.8-1.

TSR 3.3.8.1

Performance of a CHANNEL CHECK on the seismic monitoring instrumentation once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a check of external system status indications that the seismic monitoring equipment is in a state of readiness to properly function should an earthquake occur. A CHANNEL CHECK will detect gross system failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL FUNCTIONAL TEST. The Surveillance Frequency of 31 days is based on operating experience related to instrumentation systems, which demonstrates that gross instrumentation system failure in any 31-day interval is a rare event.

TSR 3.3.8.2

A CHANNEL FUNCTIONAL TEST is to be performed on each required channel to ensure entire channel will perform the intended function. A CHANNEL FUNCTIONAL TEST is the comparison of the response of the instrumentation, including all components of the instrument, to a known signal. Although the seismic

BASES

LCO 3.3.11

The primary containment hydrogen monitoring instrumentation allows the operators to detect trends in hydrogen concentration to diagnose the course of beyond design basis accidents. High hydrogen concentration is measured, continuously recorded, and displayed in the control room by a single instrument channel. The analyzer has the capability for sampling both the drywell and the suppression chamber. LCO 3.3.11 requires the primary containment hydrogen analyzer to be OPERABLE.

APPLICABILITY

The primary containment hydrogen analyzer is required to be OPERABLE when primary containment is inerted, except as allowed by the relaxations during startup and shutdown addressed below. The primary containment must be inert in MODE 1, since this is the condition with the highest probability of an event that could produce hydrogen.

Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and de-inerted as soon as possible in the plant shutdown. As long as reactor power is < 15% RTP, the potential for an event that generates significant hydrogen is low and the primary containment need not be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a startup, or within the last 24 hours before a shutdown, is low enough that these "windows," when the primary containment is not inerted, are also justified. The 24 hour time period is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.

ACTIONS

A.1

Seven days to restore the instrument is reasonable given the requirements to be available for use in diagnosing beyond design basis events.

TR 3.7 PLANT SYSTEMS

TR 3.7.4 Snubbers

BASES

BACKGROUND

Snubbers are designed to prevent unrestrained component or system motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to the component or system as a result of a seismic or other event initiating dynamic loads. An inoperable snubber (ex: failed by locked in place) may cause damage to the supported component or system from normal operating modes such as thermal operation. It is, therefore, required that all snubbers required to protect the primary coolant system or any other safety related component or system be OPERABLE during MODES 1, 2, 3, 4, and 5. The Technical Requirements Manual (TRM) action statements establish allowable outage times for components or systems addressed by the Limiting Conditions of Operation (LCO) for snubbers. These time limits are applicable when a snubber must be removed from service to perform required surveillance tests. For snubbers, the allowable outage time is 72 hours. Table 3.7.4-1, "Visual Examination Table" is published in ASME OM Code Subsection ISTD, Table ISTD 4252-1, and is based on previous table issued to all nuclear plant license holders by the Nuclear Regulatory Commission (NRC) under Generic Letter (GL) 90-09, which was added to the old Technical Specification and approved by the NRC under Technical Specification Amendment 225.

APPLICABLE SAFETY ANALYSIS

During MODES 1, 2, 3, 4, and 5 snubbers may be removed from service for functional surveillance testing to satisfy the required testing interval. When a snubber is removed from a component or system, the snubber is declared inoperable since it cannot perform its intended function while removed. This type of inoperability is not a failure. Examples of snubber failures include locked in place, high drag force, does not activate, no lockup, high lockup, low lockup, high bleed, no bleed, and damage to the snubber hardware. If a snubber is determined to be inoperable based on failure to meet the functional test acceptance criteria, an engineering evaluation is performed to establish whether, during

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

A note is provided to define the term "code snubber," as it is used in this Technical Requirements, and it shall mean snubbers that are identified by BFN ASME Code IST program as ASME Class 1, 2, or 3 equivalent snubbers. It shall also mean these safety-related snubbers that are not identified as ASME Class 1, 2, or 3 equivalent, but are treated as such.

A note is provided to indicate that each code snubber, identified by those snubbers listed in plant procedures, shall be demonstrated OPERABLE by performance of the following inservice examination and test program requirements, which are derived from ASME OM Code Subsection ISTD.

An additional note is provided to indicate that in this Technical Requirement, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

The augmented inservice inspection program includes the following.

All code snubbers are visually inspected, pin to pin, inclusive, for overall integrity and OPERABILITY. The visual inspection will include verification that no visible indications of damage, leakage, corrosion, degradation, binding, misalignment, deformation or other external characteristics that may indicate impaired OPERABILITY are present, verification that proper attachment of the snubber to the component or system and structures exist, and that no loose or missing fasteners exist. In addition, hydraulic fluid level is verified. The removal of insulation or the verification of torque values for threaded fasteners is not required for visual inspections.

The visual inspection frequency is based upon maintaining a constant level of snubber protection. In accordance with Table 3.7.4-1, the number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25 percent) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)	When the cause of the rejection of a snubber in a visual inspection is clearly established and remedied for that snubber and for any other snubber(s) that may be generically susceptible and OPERABILITY verified by inservice functional testing, if applicable, that snubber(s) may be reclassified as OPERABLE. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to the rejected snubber, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

vibration. The inspection population or category may be established based on design features, or installed conditions which may be expected to be generic. Each of these inspection populations or categories may be inspected and tested separately unless an engineering analysis indicates the inspection population or category is improperly constituted. All suspect snubbers are subject to inspection and testing regardless of inspection population or category.

To verify snubber OPERABILITY, a functional test shall be performed once per fuel cycle.

These tests will include stroking of the snubbers to verify proper movement, activation, and bleed or release. Ten percent represents an adequate sample for such tests. Observed failures on these samples will require a failure analysis and testing of additional units. If the failure analysis results in the determination that the failure of a snubber to activate or to stroke (i.e., seized components) is the result of a manufacture or design deficiency, all snubbers subject to the same defect shall be functionally tested. Also, an engineering evaluation shall be performed to determine the effects on the supported component or system during the previous unit operating cycle with the snubber inoperable, and to ensure it remains capable of meeting its designed service. A thorough visual inspection of the snubber threaded attachments to the component or system and the anchorage will be made in conjunction with all required functional tests. The stroke setting of the snubbers selected for functional testing also will be verified.

Exemption from Visual Inspection or Functional Tests:

Permanent or other exemptions from visual inspections and/or functional testing for individual snubbers may be granted by the Nuclear Regulatory Commission if a justifiable basis for exemption is presented and if applicable snubber life destructive testing was performed to qualify the snubber OPERABILITY for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall continue to be listed in the plant instructions with footnotes indicating the extent of the exemptions.

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

Snubber Service Life Program:

The service life of snubbers may be extended based on an evaluation of the records of functional tests, maintenance history, and environmental conditions to which the snubbers have been exposed.

The following will be implemented by the augmented inservice inspection program:

TSR 3.7.4.1

Visual Inspections:

At BFN, code snubbers are visually examined as one population or category, regardless of accessibility, in accordance with the schedule and interval determined by Table 3.7.4-1, which is Table ISTD-4252-1. The first inspection interval determined using Table 3.7.4-1 criteria shall be based on the previous inspection interval as established by the requirements in effect before Revision 005 of these Technical Requirements were issued.

Visual Inspection Acceptance Criteria:

Visual inspections shall verify that:

a) The snubber has no visible indications of damage, leakage, corrosion, degradation, or other external characteristics that may indicate impaired OPERABILITY, pin to pin, inclusive.

b) Fasteners for the attachment of the snubber are functional.

c) No indications of binding, misalignment, or deformation of the snubber.

d) Hydraulic snubber fluid is at the recommended level and vented reservoir is oriented such that fluid can gravitate to snubber body.

e) The absence of weld arc strikes, paint, weld slag, adhesive, or other deposits on piston rod or support cylinder that could result in unacceptable snubber performance.

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.7.4.1 (continued)

f) Snubber spherical bearing is fully engaged in attachment lug.

g) Snubber position setting is adequate.

Snubbers which appear inoperable as a result of visual inspection shall be classified unacceptable. Snubbers confirmed as unacceptable snubbers are adjusted, repaired, modified, or replaced, and counted in the determination of the subsequent examination interval in accordance with Table 3.7.4-1, regardless if the affected snubber is functionally tested in the as-found condition and determined OPERABLE per the functional test acceptance criteria of TSR 3.7.4.2.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.4.1 (continued)

A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable.

Additionally, snubbers attached to sections of safety-related systems that have experienced unexpected potentially damaging transients since the last inspection period shall be evaluated for the possibility of concealed damage and functionally tested, if applicable, to confirm OPERABILITY. Snubbers which have been made inoperable as the result of unexpected transients, isolated damage, or other random events, when the provisions of TSR 3.7.4.5 and 3.7.4.6 have been met and any other appropriate corrective action implemented, shall not be counted in determining the next visual inspection interval.

TSR 3.7.4.2

Functional Test Schedule, Lot Size, and Composition:

Once per fuel cycle, a representative sample of 10% of the total of each type of code snubbers in use in the plant shall be functionally tested either in place or in a bench test. The sample population is rounded up to the next whole integer. Safety-related snubbers that are not ASME Class 1, 2, or 3 equivalent snubbers shall not be included in the snubber population when selecting the initial or additional samples.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.4.2 (continued)

As practical, the representative sample selected for functional testing shall include representation from the Defined Test Plan Group (DTPG) based on the significant features (i.e., the various designs, configurations, operating environments, and the range of size and capacity of snubbers within the types) and based on the ratio of the number of snubbers of each significant feature, to the total number of snubbers in the DTPG. The sample shall be generally representative as specified in ISTD-5311(a), but may also be selected from snubbers concurrently scheduled for seal replacement or other similar activity related to service life monitoring. The snubbers shall be tested on a generally rotational basis to coincide with the service life monitoring. The representative sample should be weighed to include more snubbers from severe service areas such as near heavy equipment. After testing, at the time of reinstallation, the snubber shall meet visual examination attributes described in ISTD-4110(a), -4110(c), -4110(d), and -4110(e). The stroke setting shall be verified.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.4.2 (continued)

Functional Test Acceptance Criteria:

The snubber functional test shall verify that:

a. Activation (restraining action) is achieved in both tension and compression within the specified range of velocity or acceleration.

b. Snubber bleed, or release where required, is present in both compression and tension within the specified range.

c. For mechanical snubbers, the drag force is within the specified limits, in tension and compression.

d. For hydraulic snubbers, if required to verify proper assembly, drag force is within specific limits in tension and compression.

e. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

f. Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

TSR 3.7.4.3

Functional Test Failure Analysis and Additional Test Lots:

A failure analysis shall be performed for each failure to meet the functional test acceptance criteria to determine the cause of the failure. The result of this analysis shall be used, if applicable, in selecting snubbers to be tested in the subsequent lot in an effort to determine the OPERABILITY of other snubbers which may be subject to the same failure mode. Selection of snubbers for future testing may also be based on the failure analysis.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.4.3 (continued)

For each snubber that does not meet the functional test acceptance criteria, an additional lot equal to 5 percent of the subject snubber DTPG shall be functionally tested. When establishing additional sample testing, failure analysis results of unacceptable snubbers are to be used to determine if establishing an FMG is appropriate. Additional lot (i.e., sample) population is rounded up to next whole integer. Rounding up satisfies ISTD-5312 requirement that additional samples shall be at least one-half the size of the initial sample from the DTPG. Additional samples selected from a DTPG follow the same composition as the original sample. However, additional samples from FMGs are random samples. Testing shall continue until no additional inoperable snubbers are found within subsequent lots or all snubbers of the original functional test type have been tested or all suspect snubbers identified by the failure analysis have been tested, as applicable.

The discovery of loose or missing attachment fasteners will be evaluated to determine whether the cause may be localized or generic. The result of the evaluation will be used to select other suspect snubbers for verifying the attachment fasteners, as applicable.

TSR 3.7.4.4

(Deleted by TRM Revision 125.)

TSR 3.7.4.5

Functional Test Failure - Supported Component or System Analysis:

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the component or system which is restrained by the snubber(s) due to not meeting their functional test acceptance criteria. The purpose of this engineering evaluation shall be to determine if the component or system restrained by the snubber(s) was adversely affected by the inoperability of the snubber(s), and in order to ensure that the restrained component or system remains capable of meeting the designed service.

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.7.4.6

Functional Testing of Repaired and Spare Snubbers:

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be adjusted, repaired, modified, scrapped, or replaced. Replacement snubbers and snubbers having repairs, adjustments, or modifications which might affect the functional test results shall meet the functional test acceptance criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the functional test must have been performed within 12 months prior to being installed in the unit.

REFERENCES

1. BFN Technical Specifications (version prior to standardized version)

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3.3-1	Revision 12	08-17-1999
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3.3-26	Revision 112	10-29-2014
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3.3-56	0	Initial
3.3-57	Revision 44	03-22-2004
3.3-58	Revision 40	10-03-2003
3.3-59	Revision 93	05-16-2012
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3.3-61	Revision 78	01-07-2011
3.4-1	Revision 21	03-13-2001
3.4-2	Revisions 19 and 32	08-15-2002
3.4-3	0	Initial
3.4-4	Revision 21	03-13-2001
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3.4-6	0	Initial
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3.4-8	0	Initial
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3.6-3	0	Initial
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5.0-4	Revision 34	10-03-2002
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ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. As required by Required Action A.1 and referenced in Table 3.3.5-1.	D.1	Monitor torus temperature to observe any unexplained temperature increase which might be indicative of an open relief valve.	Once per 12 hours
	<u>AND</u>		
	D.2	Restore control room indication by either the Tailpipe Thermocouple Temperature or Acoustic Monitor to OPERABLE status for each relief valve.	30 days
	<u>AND</u>		
	D.3	When inoperable for more than 30 days, enter the condition into the Corrective Action Program.	24 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action A.1 and referenced in Table 3.3.5-1.	------------------NOTE---------------- Required Actions E.1.1 and E.1.2 are not applicable when in MODES 4 and 5. --	
	E.1.1 Restore required control room indication channel to OPERABLE status.	72 hours
	OR	
	E.1.2 Initiate the preplanned alternate method of monitoring the parameter.	72 hours
	AND	
	E.2 When inoperable for more than seven days, enter the condition into the Corrective Action Program.	24 hours

(continued)

TR 3.3 INSTRUMENTATION

TR 3.3.7 Meteorological Monitoring Instrumentation

LCO 3.3.7 The meteorological monitoring instrumentation listed in Table 3.3.7-1 shall be OPERABLE.

APPLICABILITY: At all times

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. The number of OPERABLE meteorological monitoring channels less than required by Table 3.3.7-1.	A.1	Restore inoperable channel(s) to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1	Enter the condition into the Corrective Action Program.	24 hours

TR 3.3 INSTRUMENTATION

TR 3.3.8 Seismic Monitoring Instrumentation

LCO 3.3.8 The seismic monitoring instruments listed in Table 3.3.8-1 shall be OPERABLE.

APPLICABILITY: At all times

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more seismic monitoring instruments in Panel 1-PNLA-9-44 or foundation instrument BFN-0-ACGR-052-0001 on the Unit 1 Reactor Building Base Slab inoperable.	A.1 Restore inoperable instrument(s) to an OPERABLE status.	30 days
B. One or more seismic monitoring instruments BFN-0-ACGR-052-0002 or BFN-0-ACGR-052-0003 inoperable.	B.1 Restore inoperable instrument(s) to an OPERABLE status.	60 days
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Enter the condition into the Corrective Action Program.	24 hours

NOTES

1. As used in this Technical Requirement "code snubber," shall mean snubbers that are identified by BFN ASME Code IST program as ASME Class 1, 2, or 3 equivalent snubbers. It shall also mean those BFN safety-related snubbers that are not identified as ASME Class 1, 2, or 3 equivalent, but are treated as such.
2. Each code snubber, identified by those snubbers listed in plant procedures, shall be demonstrated OPERABLE by performance of the following inservice examination and test program requirements, which is derived from ASME OM Code Subsection ISTD.
3. As used in this Technical Requirement, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.
4. As used in this Technical Requirement, "population or category" shall mean the total number of snubbers being visually inspected as a lot.

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.7.4.1	NOTE At BFN, code snubbers are visually examined as one population or category, regardless of accessibility, in accordance with the schedule and interval determined by Table 3.7.4-1, which is Table ISTD-4252-1. The first inspection interval determined using Table 3.7.4-1 criteria shall be based on the previous inspection interval established by the requirements in effect before Revision 007 of these Technical Requirements were issued.	
	Perform visual examination of required snubber(s) based on the acceptance criteria of plant visual examination procedures and the frequency based on Table 3.7.4-1, which are both based on ASME OM Code, Subsection ISTD. Visual Examinations shall confirm:	In accordance with Table 3.7.4-1
a.	No visible indications of damage, leakage, corrosion, degradation, or other external	

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.7.4.1 (continued)	characteristics that may indicate impaired OPERABILITY, pin to pin inclusive. b. Fasteners for the attachment of the snubber are functional. c. No indication of binding, misalignment, or deformation of the snubber. d. Hydraulic snubber fluid is at the recommended level and that vented reservoir is oriented such that fluid can gravitate to snubber body. e. The absence of weld arc strikes, paint, weld slag, adhesive, or other deposits on piston rod or support cylinder that could result in unacceptable snubber performance. f. Snubber spherical bearing is fully engaged in attachment lug. g. Snubber position setting is adequate. Snubbers which appear inoperable as a result of visual inspection shall be classified unacceptable. Snubbers confirmed as unacceptable snubbers are adjusted, repaired, modified, or replaced, and counted in determination of the subsequent examination interval in accordance with Table 3.7.4.1, regardless if the affected snubber is functionally tested in the as-found condition and determined OPERABLE per the criteria of TSR 3.7.4.2. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the system or train shall be declared inoperable.	

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
TSR 3.7.4.1 (continued)	Additionally, snubbers attached to sections of safety related systems that have experienced unexpected potentially damaging transients since the last inspection period shall be evaluated for the possibility of concealed damage and functionally tested, if applicable, to confirm OPERABILITY. Snubbers which have been made inoperable as the result of unexpected transients, isolated damage, or other random events, when the provisions of TSR 3.7.4.5. and TSR 3.7.4.6 have been met and any other appropriate corrective action implemented, shall not be counted in determining the next visual inspection interval.	
TSR 3.7.4.2	Perform an in-place or bench functional test of a representative sample of 10% of the total of each type of code snubber(s). Sample population is rounded up to next whole integer. Safety-related snubbers that are not ASME Class 1, 2, or 3 equivalent snubbers shall not be included in the snubber population when selecting the initial or additional samples. a. As practice, the representative sample selected for functional testing shall include representation from the Defined Test Plan Group (DTPG) based on the significant features (i.e., the various designs, configurations, operating environments, and the range of size and capacity of snubbers within the types) and based on the ratio of the number of snubbers of each significant feature, to the total number of snubbers in the DTPG. b. The sample shall be generally representative as specified in ISTD-5311 (a), but may also be selected from	In accordance with Inservice Testing Program

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.7.4.2 (continued)	snubbers concurrently scheduled for seal replacement or other similar activity related to service life monitoring. The snubbers shall be tested on a generally rotational basis to coincide with the service life monitoring. c. The representative sample should be weighed to include more snubbers from severe service areas such as near heavy equipment. d. After the testing, at the time of reinstallation, the snubber shall meet visual examination attributes as described in ISTD-4110(a), -4110(c), -4110(d), -4110(e). The stroke setting shall be verified. Functional Test Acceptance Criteria: The snubber functional test shall verify that: a. Activation (restraining action) is achieved in both tension and compression within the specified range of velocity or acceleration. b. Snubber bleed or release, where required, is present in both compression and tension within the specified range. c. For mechanical snubbers, the drag force is within the specified limits in tension and compression. d. For hydraulic snubbers, if required to verify proper assembly, drag force is within specific limits in tension and compression. e. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified. f. Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.	

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TSR 3.7.4.3	A failure analysis shall be made of each failure to meet the functional test acceptance criteria of TSR 3.7.4.2 to determine the cause of the failure. The result of this analysis shall be used, if applicable, in selecting snubbers to be tested in the subsequent lot in an effort to determine the OPERABILITY of other snubbers which may be subject to the same failure mode. Selection of snubbers for future testing may also be based on the failure analysis. For each failed snubber, perform in-place or bench functional test on an additional lot equal to 5% of the subject snubber DTPG. When establishing additional sample testing, failure analysis results of unacceptable snubbers are to be used to determine if establishing a FMG is appropriate. Additional lot (i.e., sample) population is rounded up to next whole integer. Rounding up satisfies ISTD-5312 requirement that additional samples shall be at least one-half the size of the initial sample from the DTPG. Additional samples from DTPGs follow the composition of the original sample. However, additional samples selected from FMGs are random samples. Testing shall continue until no additional inoperable snubbers are found within subsequent lots or all snubbers of the original test type are tested or all suspect snubbers identified by the failure analysis have been tested, as applicable. The functional test criteria shall be as specified in TSR 3.7.4.2. Prior to functional testing the discovery of loose or missing attachment fasteners will be evaluated to determine whether the cause may be localized or generic. The result of the evaluation will be used to select other suspect snubbers for verifying the attachment fasteners, as applicable.	Once for each discovery of snubber failure to meet functional test acceptance criteria

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TSR 3.7.4.4	(Deleted by TRM Revision 125.)	
TSR 3.7.4.5	Perform an engineering evaluation on the component or system which is restrained by the snubber(s) found inoperable due to not meeting their functional test acceptance criteria as specified in TSR 3.7.4.2.	Once for each discovery of an inoperable snubber
TSR 3.7.4.6	Verify replacement snubbers and snubbers having repairs, adjustments, or modifications which might affect the functional test results meet the test criteria of TSR 3.7.4.2. a. These snubbers shall have met the acceptance criteria subsequent to their most recent service; and b. The functional test must have been performed within the 12 months prior to being installed in the unit.	Once prior to installation in the unit for each replaced, repaired, adjusted, or modified snubber where functional test results might be affected.

Table 3.7.4-1
Visual Examination Table
(Ref Table ISTD-4252-1)

	NUMBER OF UNACCEPTABLE SNUBBERS		
Population or Category (Note 1)	Column A Extend Interval (Notes 2 and 3)	Column B Repeat Interval (Notes 2, 4 and 5)	Column C Reduce Interval 2/3 (Notes 2, 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or more	29	56	109

Note 1: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. The next lower integer shall be used when interpolation results in a fraction.

Note 2: The basic interval shall be the normal fuel cycle up to 24 months. The examination interval may be as great as twice, the same, or as small as fractions of the previous interval as required by the following Notes. The examination interval may vary + / - 25% of the current interval.

Table 3.7.4-1 (Continued)
Visual Examination Table

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months. In that case, the next visual examination according to the previous interval may be skipped.

Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is less than the number in Column C, but greater than the number is Column B, the next interval shall be reduced to two-thirds of the previous examination interval or, in accordance with the interpolation between Columns B and C, in proportion to the exact number of unacceptable snubbers.

Note 6: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval.

The provisions of TSR 3.0.1, 3.0.2, and 3.0.3 are applicable for all inspection intervals up to and including 48 months.

EDMS L32 160107 800
QA Document
BFE-3603, Revision 2

TVA NPG

Reactor Engineering and Fuels - BWRFE
1101 Market Street, Chattanooga, TN 37402

Browns Ferry Unit 3 Cycle 17

Core Operating Limits Report, (105% OLTP)

TVA-COLR-BF3C17 Revision 2 (Final)
(Revision Log, Page v)

January 2016

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Chairman, PORC

Approved: ____________________ Date: 1.21.16
Plant Manager

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Revision Log

Number	Page	Description
1-R2	ix	Added Reference 28 disposition of cycle exposure extension.
2-R2	23	Updated maximum End of Coast CAVEX basis for MCPRp limits per new Reference 28.
1-R1	vii	Added RCP and RCPOOS to the list.
2-R1	viii	Updated Reference 1. This revised reference provides explicit $MCPR_p$ results for single loop operation (RCPOOS). No other results are impacted.
3-R1	24	Revised Section 4.2.5 wording to be consistent with the Reference 1 reload safety analysis report revision. Includes a new footnote.
4-R1	26-34	Removed footnote information regarding the addition of a 0.02 adder to the safety limit for constructing single loop operating limits. The foot note is still true, but not relevant to the document as the Reference 1 reload safety analysis report provides explicit limits.
5-R1	35	Provided new Table 4.9, and footnotes, showing the explicit $MCPR_p$ limits for single loop operation (RCPOOS)
0-R0	All	New document.

Nomenclature

APLHGR	Average Planar LHGR
APRM	Average Power Range Monitor
AREVA NP	Vendor (Framatome, Siemens)
ARTS	APRM/RBM Technical Specification Improvement
BOC	Beginning of Cycle
BWR	Boiling Water Reactor
CAVEX	Core Average Exposure
CD	Coast Down
CMSS	Core Monitoring System Software
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRWE	Control Rod Withdrawal Error
CSDM	Cold SDM
DIVOM	Delta CPR over Initial CPR vs. Oscillation Magnitude
ECCS	Emergency Core Cooling System
EOC	End of Cycle
EOCLB	End-of-Cycle Licensing Basis
EOOS	Equipment OOS
FFTR	Final Feedwater Temperature Reduction
FFWTR	Final Feedwater Temperature Reduction
FHOOS	Feedwater Heaters OOS
ft	Foot: English unit of measure for length
GNF	Vendor (General Electric, Global Nuclear Fuels)
GWd	Giga Watt Day
HTSP	High TSP
ICA	Interim Corrective Action
ICF	Increased Core Flow (beyond rated)
IS	In-Service
kW	kilo watt: SI unit of measure for power.
LCO	License Condition of Operation
LFWH	Loss of Feedwater Heating
LHGR	Linear Heat Generation Rate
LHGRFAC	LHGR Multiplier (Power or Flow dependent)
LPRM	Low Power Range Monitor
LRNB	Generator Load Reject, No Bypass

MAPFAC	MAPLHGR multiplier (Power or Flow dependent)
MCPR	Minimum CPR
MELL	Maximum Extended Load Line
MSRV	Moisture Separator Reheater Valve
MSRVOOS	MSRV OOS
MTU	Metric Ton Uranium
MWd/MTU	Mega Watt Day per Metric Ton Uranium
NEOC	Near EOC
NRC	United States Nuclear Regulatory Commission
NSS	Nominal Scram Speed
NTSP	Nominal TSP
OLMCPR	MCPR Operating Limit
OOS	Out-Of-Service
OPRM	Oscillation Power Range Monitor
OSS	Optimum Scram Speed
PBDA	Period Based Detection Algorithm
Pbypass	Power, below which TSV Position and TCV Fast Closure Scrams are Bypassed
PLU	Power Load Unbalance
PLUOOS	PLU OOS
PRNM	Power Range Neutron Monitor
RBM	Rod Block Monitor
RCP	Recirculation Pump
RCPOOS	RCP OOS
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPTOOS	RPT OOS
SDM	Shutdown Margin
SLMCPR	MCPR Safety Limit
SLO	Single Loop Operation
SRV	Safety Relief Valve
SRVOOS	SRV OOS
TBV	Turbine Bypass Valve
TBVIS	TBV IS
TBVOOS	TBV OOS
TIP	Transversing In-core Probe
TIPOOS	TIP OOS
TLO	Two Loop Operation
TSP	Trip Setpoint
TSSS	Technical Specification Scram Speed
TVA	Tennessee Valley Authority

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1 Introduction

In anticipation of cycle startup, it is necessary to describe the expected limits of operation.

1.1 Purpose

The primary purpose of this document is to satisfy requirements identified by unit technical specification section 5.6.5. This document may be provided, upon final approval, to the NRC.

1.2 Scope

This document will discuss the following areas:

- Average Planar Linear Heat Generation Rate (APLHGR) Limit
  (Technical Specifications 3.2.1 and 3.7.5)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Linear Heat Generation Rate (LHGR) Limit
  (Technical Specification 3.2.3, 3.3.4.1, and 3.7.5)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Minimum Critical Power Ratio Operating Limit (OLMCPR)
  (Technical Specifications 3.2.2, 3.3.4.1, and 3.7.5)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Oscillation Power Range Monitor (OPRM) Setpoint
  (Technical Specification Table 3.3.1.1)
  Applicability: Mode 1, ≥ (as specified in Technical Specifications Table 3.3.1.1-1)
- Average Power Range Monitor (APRM) Flow Biased Rod Block Trip Setting
  (Technical Requirements Manual Section 5.3.1 and Table 3.3.4-1)
  Applicability: Mode 1, ≥ (as specified in Technical Requirements Manuals Table 3.3.4-1)
- Rod Block Monitor (RBM) Trip Setpoints and Operability
  (Technical Specification Table 3.3.2.1-1)
  Applicability: Mode 1, ≥ % RTP as specified in Table 3.3.2.1-1 (TS definition of RTP)
- Shutdown Margin (SDM) Limit
  (Technical Specification 3.1.1)
  Applicability: All Modes

1.3 Fuel Loading

The core will contain previously exposed and fresh AREVA NP, Inc., ATRIUM-10 fuel. Nuclear fuel types used in the core loading are shown in Table 1.1. The core shuffle and final loading were explicitly evaluated for BOC cold shutdown margin performance as documented in Reference 5.

Table 1.1 Nuclear Fuel Types*

Fuel Description	Original Cycle	Number of Assemblies	Nuclear Fuel Type (NFT)	Fuel Names (Range)
ATRIUM-10 A10-3831B-15GV80-FCD	15	120	6	FCD001-FCD200
ATRIUM-10 A10-3403B-9GV80-FCD	15	20	7	FCD257-FCD276
ATRIUM-10 A10-3392B-10GV80-FCD	15	7	8	FCD221-FCD256
ATRIUM-10 A10-4218B-15GV80-FCC	15	2	9	FCC217-FCC218
ATRIUM-10 A10-4218B-13GV80-FCC	15	4	10	FCC307-FCC310
ATRIUM-10 A10-3757B-10GV80-FCC	15	40	11	FCC335-FCC374
ATRIUM-10 A10-3440B-11GV80-FCE	16	144	12	FCE001-FCE144
ATRIUM-10 A10-3826B-13GV80-FCE	16	44	13	FCE145-FCE188
ATRIUM-10 A10-4075B-13GV80-FCE	16	47	14	FCE189-FCE236
ATRIUM-10 A10-4081B-12GV80-FCE	16	48	15	FCE237-FCE284
ATRIUM-10 A10-3849B-13GV80-FCF	17	176	16	FCF301-FCF476
ATRIUM-10 A10-3882B-10GV70-FCF	17	40	17	FCF477-FCF516
ATRIUM-10 A10-4116B-12GV70-FCF	17	72	18	FCF517-FCF588

1.4 Acceptability

Limits discussed in this document were generated based on NRC approved methodologies per References 6 through 22.

* The table identifies the expected fuel type breakdown in anticipation of final core loading. The final composition of the core depends upon uncertainties during the outage such as discovering a failed fuel bundle, or other bundle damage. Minor core loading changes, due to unforeseen events, will conform to the safety and monitoring requirements identified in this document.

2 APLHGR Limits

(Technical Specifications 3.2.1 & 3.7.5)

The APLHGR limit is determined by adjusting the rated power APLHGR limit for off-rated power, off-rated flow, and SLO conditions. The most limiting of these is then used as follows:

$$\text{APLHGR limit} = \text{MIN} \left(\text{APLHGR}_P , \text{APLHGR}_F, \text{APLHGR}_{SLO} \right)$$

where:

APLHGR_P	off-rated power APLHGR limit	[APLHGR_{RATED} * MAPFAC_P]
APLHGR_F	off-rated flow APLHGR limit	[APLHGR_{RATED} * MAPFAC_F]
APLHGR_{SLO}	SLO APLHGR limit	[APLHGR_{RATED} * SLO Multiplier]

2.1 Rated Power and Flow Limit: APLHGR_{RATED}

The rated conditions APLHGR for ATRIUM-10 fuel is identified in Reference 1 and shown in Figure 2.1.

2.2 Off-Rated Power Dependent Limit: APLHGR_P

Reference 1, for ATRIUM-10 fuel, does not specify a power dependent APLHGR. Therefore, MAPFAC_P is set to a value of **1.0**.

2.2.1 Startup without Feedwater Heaters

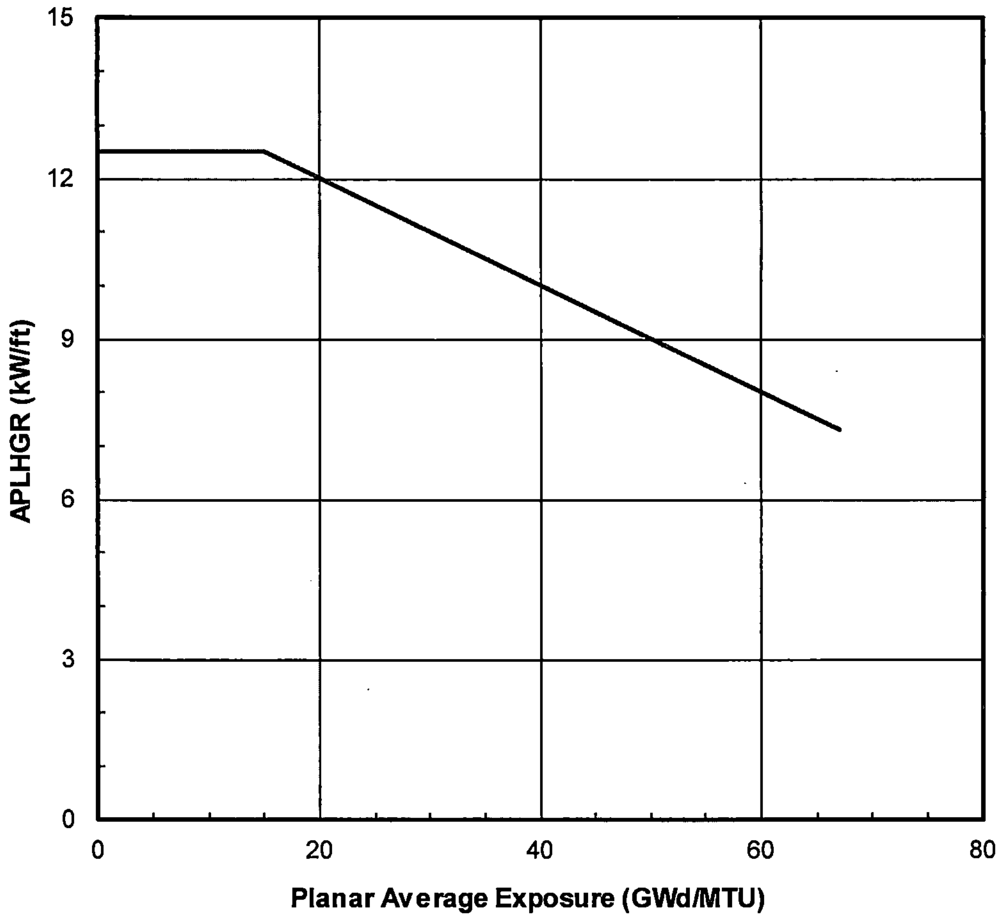
There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. No Additional power dependent limitation is required.

2.3 Off-Rated Flow Dependent Limit: APLHGR_F

Reference 1, for ATRIUM-10 fuel, does not specify a flow dependent APLHGR. Therefore, MAPFAC_F is set to a value of **1.0**.

2.4 Single Loop Operation Limit: APLHGR_{SLO}

The single loop operation multiplier for ATRIUM-10 fuel is **0.85**, per Reference 1.



Planar Avg. Exposure (GWd/MTU)	APLHGR Limit (kW/ft)
0.0	12.5
15.0	12.5
67.0	7.3

Figure 2.1 APLHGR$_{RATED}$ for ATRIUM-10 Fuel

2.5 Equipment Out-Of-Service Corrections

The limits shown in Figure 2.1 are applicable for operation with all equipment In-Service as well as the following Equipment Out-Of-Service (EOOS) options; including combinations of the options.

In-Service	All equipment In-Service*
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
PLUOOS	Power Load Unbalance Out-Of-Service
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service or Final Feedwater Temperature Reduction

Single Recirculation Loop Operation (SLO) requires the application of the SLO multipliers to the rated APLHGR limits as described previously.

* All equipment service conditions assume 1 SRVOOS.

3 LHGR Limits

(Technical Specification 3.2.3, 3.3.4.1, & 3.7.5)

The LHGR limit is determined by adjusting the rated power LHGR limit for off-rated power and off-rated flow conditions. The most limiting of these is then used as follows:

LHGR limit = MIN ($LHGR_P$, $LHGR_F$)

where:

$LHGR_P$	off-rated power LHGR limit	[$LHGR_{RATED}$ * $LHGRFAC_P$]
$LHGR_F$	off-rated flow LHGR limit	[$LHGR_{RATED}$ * $LHGRFAC_F$]

3.1 Rated Power and Flow Limit: $LHGR_{RATED}$

The rated conditions LHGR for all fuel types, is identified in Reference 1 and shown in Figure 3.1. The LHGR limit is consistent with References 2 and 3.

3.2 Off-Rated Power Dependent Limit: $LHGR_P$

LHGR limits are adjusted for off-rated power conditions using the $LHGRFAC_P$ multiplier provided in Reference 1. The multiplier is split into two sub cases: turbine bypass valves in and out-of-service. The multipliers are shown in Figure 3.2.

3.2.1 Startup without Feedwater Heaters

There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. Additional limits are shown in Figure 3.4 and Figure 3.5, based on temperature conditions identified in Table 3.1.

Table 3.1 Startup Feedwater Temperature Basis

Power	Temperature Range 1	Temperature Range 2
(%Rated)	(°F)	(°F)
25	160.0	155.0
30	165.0	160.0
40	175.0	170.0
50	185.0	180.0

3.3 Off-Rated Flow Dependent Limit: $LHGR_F$

The LHGR limit is adjusted for off-rated flow conditions using the $LHGRFAC_F$ multiplier provided in Reference 1. The multiplier are shown in Figure 3.3.

3.4 Equipment Out-Of-Service Corrections

The limit shown in Figure 3.1 is applicable for operation with all equipment In-Service as well as the following Equipment Out-Of-Service (EOOS) options; including combinations of the options.*

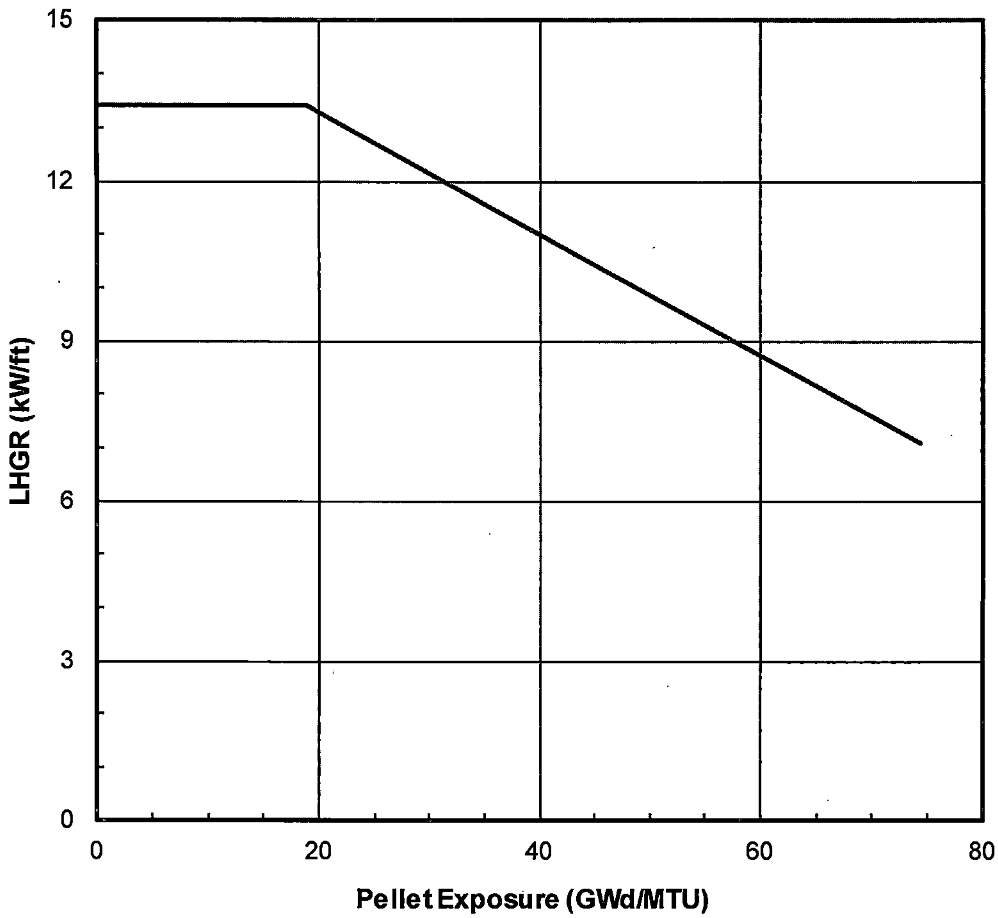
In-Service	All equipment In-Service
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
PLUOOS	Power Load Unbalance Out-Of-Service
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service or Final Feedwater Temperature Reduction
SLO	Single Loop Operation, One Recirculation Pump Out--Of-Service

Off-rated power corrections shown in Figure 3.2 are dependent on operation of the Turbine Bypass Valve system. For this reason, separate limits are to be applied for TBVIS or TBVOOS operation. The limits have no dependency on RPTOOS, PLUOOS, FHOOS/FFWTR, or SLO.

Off-rated flow corrections shown in Figure 3.3 are bounding for all EOOS conditions.

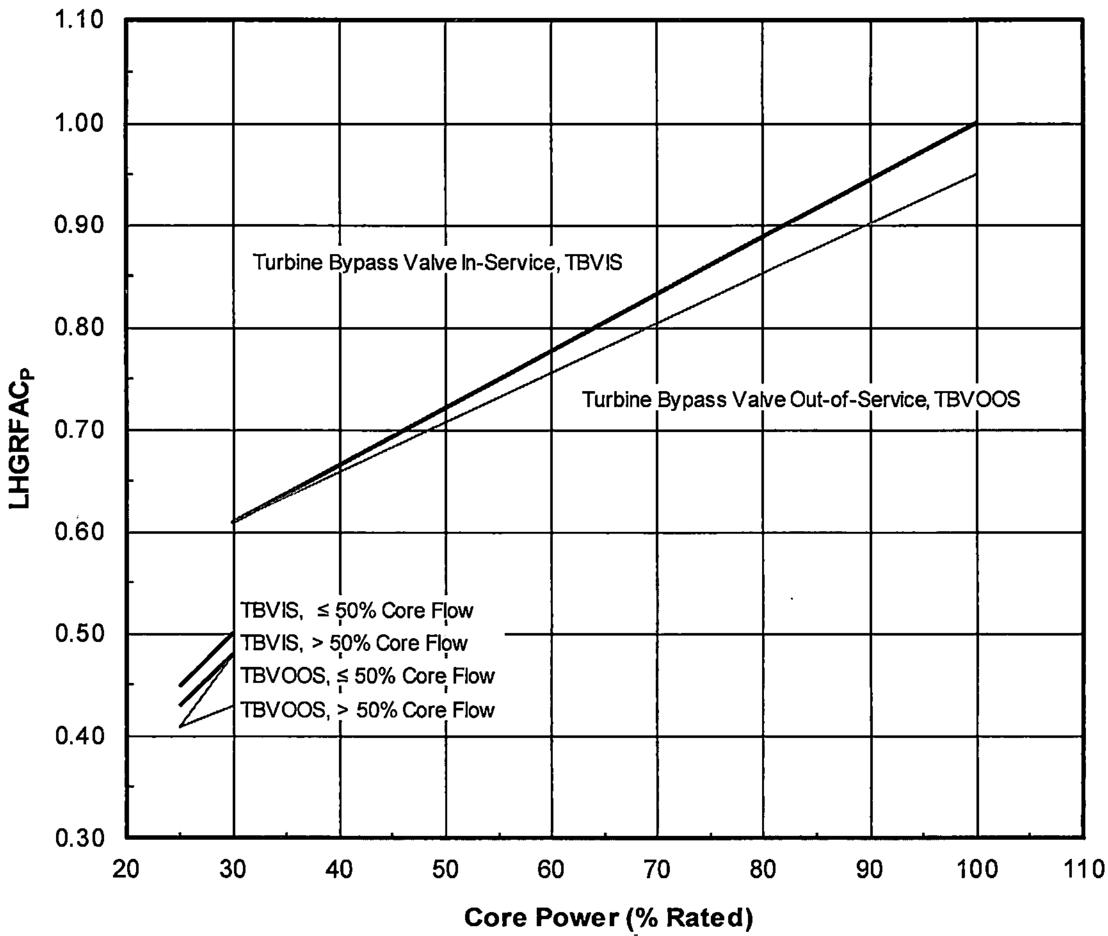
Off-rated power corrections shown in Figure 3.4 and Figure 3.5 are also dependent on operation of the Turbine Bypass Valve system. In this case, limits support FHOOS operation during startup. These limits have no dependency on RPTOOS, PLUOOS, or SLO.

* All equipment service conditions assume 1 SRVOOS.



Pellet Exposure (GWd/MTU)	LHGR Limit (kW/ft)
0.0	13.4
18.9	13.4
74.4	7.1

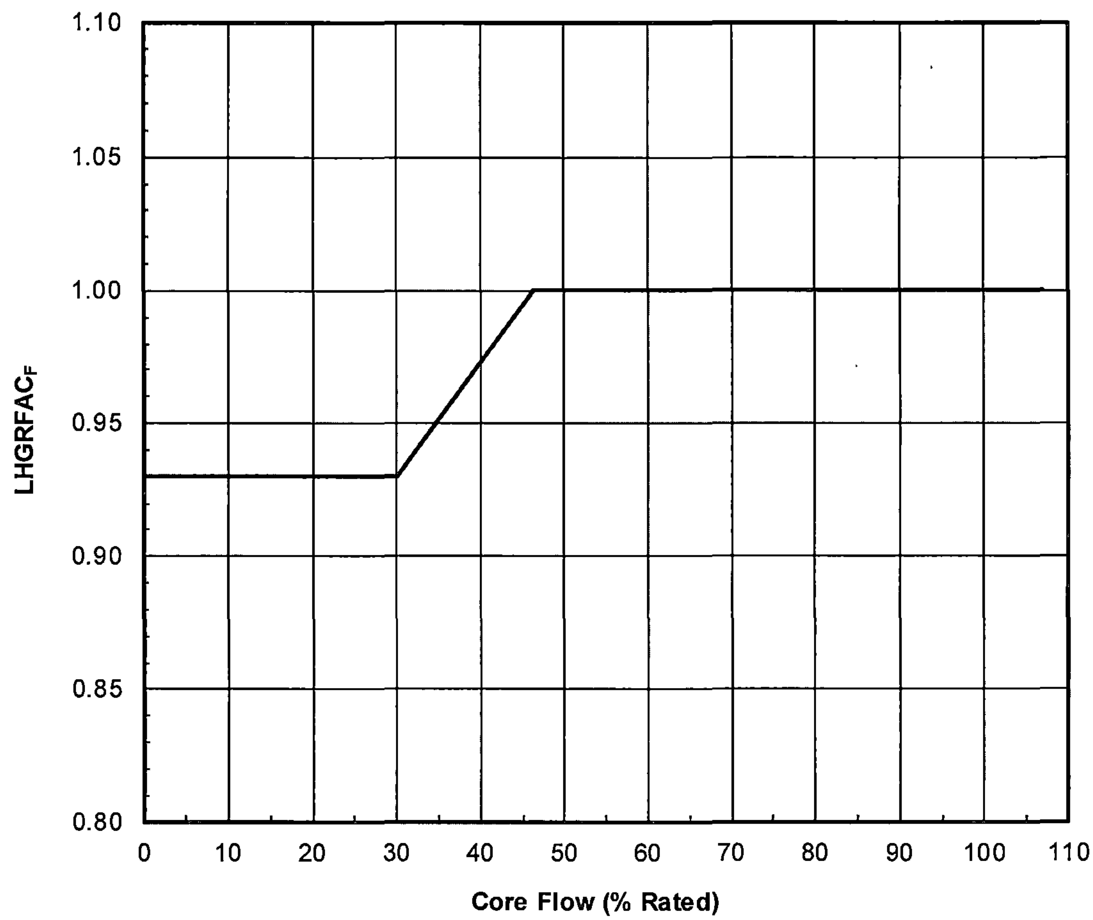
Figure 3.1 $LHGR_{RATED}$ for ATRIUM-10 Fuel



Turbine Bypass In-Service		*Turbine Bypass Out-of-Service*	
Core Power (% Rated)	$LHGRFAC_P$	Core Power (% Rated)	$LHGRFAC_P$
100.0	1.00	100.0	0.95
30.0	0.61	30.0	0.61
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.48	30.0	0.43
25.0	0.43	25.0	0.41
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.50	30.0	0.48
25.0	0.45	25.0	0.41

Figure 3.2 Base Operation $LHGRFAC_P$ for ATRIUM-10 Fuel

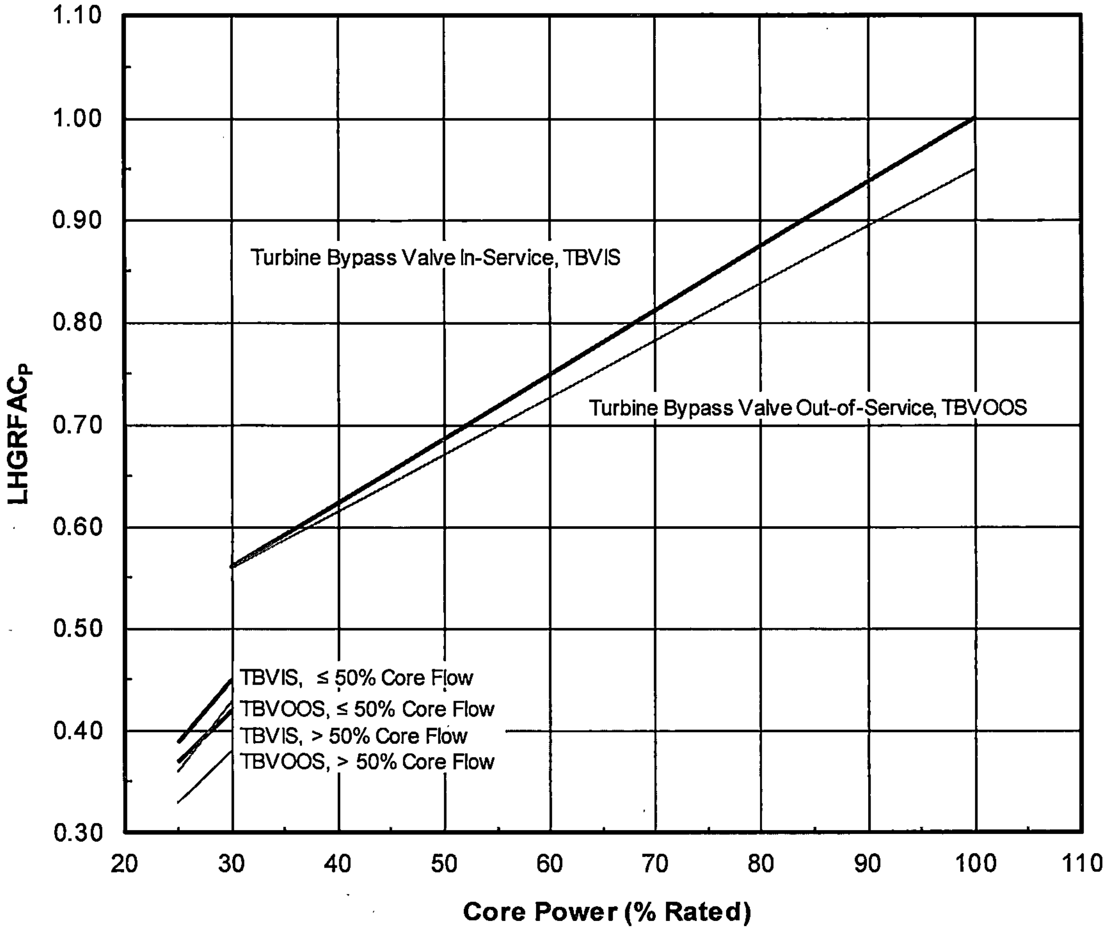
(*Independent of other EOOS conditions*)



Core Flow (% Rated)	$LHGRFAC_F$
0.0	0.93
30.0	0.93
46.4	1
107.0	1

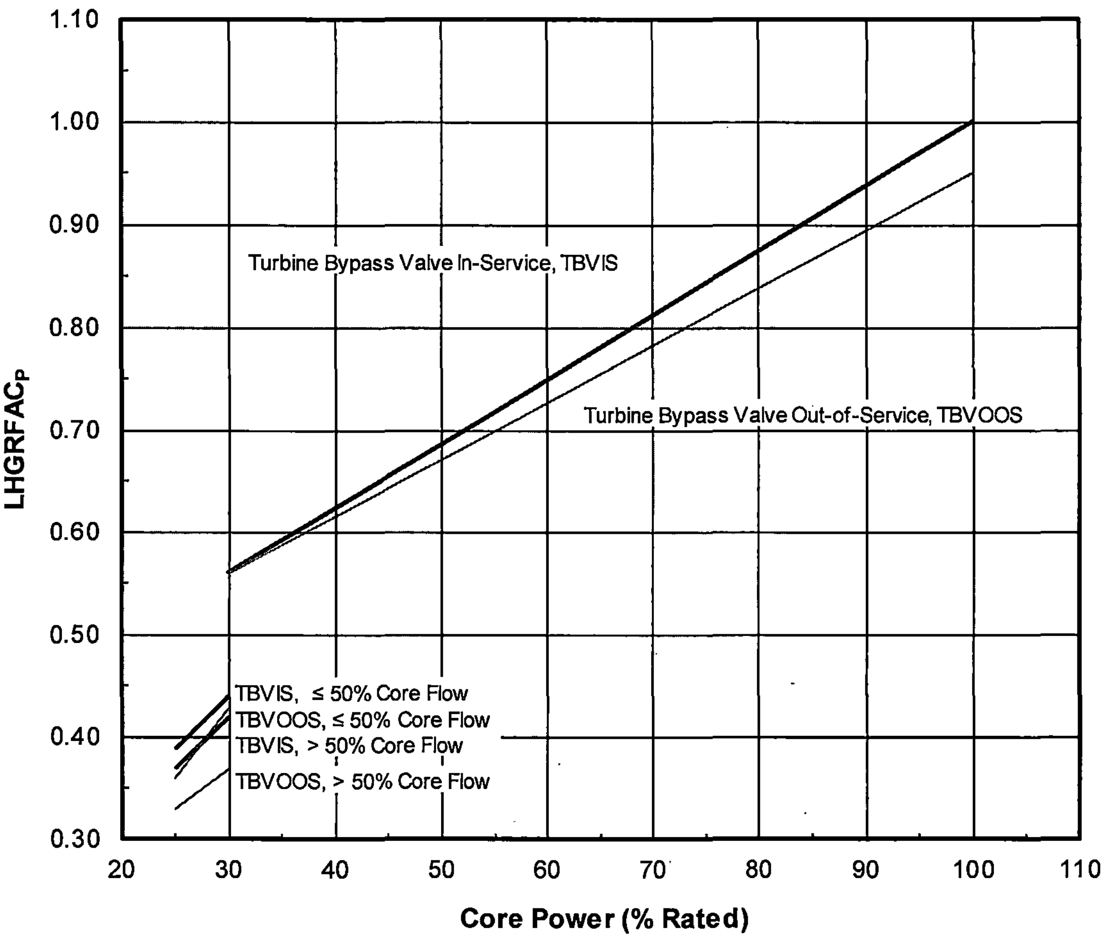
Figure 3.3 $LHGRFAC_F$ for ATRIUM-10 Fuel
(*Values bound all EOOS conditions*)

(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)



Turbine Bypass In-Service		*Turbine Bypass Out-of-Service*	
Core Power (% Rated)	LHGRFAC$_P$	Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00	100.0	0.95
30.0	0.56	30.0	0.56
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.42	30.0	0.38
25.0	0.37	25.0	0.33
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.45	30.0	0.43
25.0	0.39	25.0	0.36

Figure 3.4 Startup Operation LHGRFAC$_P$ for ATRIUM-10 Fuel:
Table 3.1 Temperature Range 1
(no Feedwater heating during startup)



Turbine Bypass In-Service		*Turbine Bypass Out-of-Service*	
Core Power (% Rated)	**LHGRFAC$_P$**	**Core Power (% Rated)**	**LHGRFAC$_P$**
100.0	1.00	100.0	0.95
30.0	0.56	30.0	0.56
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.42	30.0	0.37
25.0	0.37	25.0	0.33
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.44	30.0	0.43
25.0	0.39	25.0	0.36

Figure 3.5 Startup Operation LHGRFAC$_P$ for ATRIUM-10 Fuel: Table 3.1 Temperature Range 2
(no Feedwater heating during startup)

4 OLMCPR Limits

(Technical Specification 3.2.2, 3.3.4.1, & 3.7.5)

OLMCPR is calculated to be the most limiting of the flow or power dependent values

$$\text{OLMCPR limit} = \text{MAX}\,(\,MCPR_F\,,\,MCPR_P\,)$$

where:

$MCPR_F$	core flow-dependent MCPR limit
$MCPR_P$	power-dependent MCPR limit

4.1 Flow Dependent MCPR Limit: $MCPR_F$

$MCPR_F$ limits are dependent upon core flow (% of Rated), and the max core flow limit, (Rated or Increased Core Flow, ICF). $MCPR_F$ limits are shown in Figure 4.1, consistent with Reference 1. Limits are valid for all EOOS combinations. No adjustment is required for SLO conditions.

4.2 Power Dependent MCPR Limit: $MCPR_P$

$MCPR_P$ limits are dependent upon:

- Core Power Level (% of Rated)
- Technical Specification Scram Speed (TSSS), Nominal Scram Speed (NSS), or Optimum Scram Speed (OSS)
- Cycle Operating Exposure (NEOC, EOC, and CD - as defined in this section)
- Equipment Out-Of-Service Options
- Two or Single recirculation Loop Operation (TLO vs. SLO)

The $MCPR_P$ limits are provided in the following tables, where each table contains the limits for all fuel types and EOOS options (for a specified scram speed and exposure range). The CMSS determines $MCPR_P$ limits, from these tables, based on linear interpolation between the specified powers.

4.2.1 Startup without Feedwater Heaters

There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. Additional power dependent limits are shown in Table 4.5 through Table 4.8, based on temperature conditions identified in Table 3.1.

4.2.2 Scram Speed Dependent Limits (TSSS vs. NSS vs. OSS)

$MCPR_P$ limits are provided for three different sets of assumed scram speeds. The Technical Specification Scram Speed (TSSS) $MCPR_P$ limits are applicable at all times, as long as the scram time surveillance demonstrates the times in Technical Specification Table 3.1.4-1 are met. Both Nominal Scram Speeds (NSS) and/or Optimum Scram Speeds (OSS) may be used, as long as the scram time surveillance demonstrates Table 4.1 times are applicable.*†

Table 4.1 Nominal Scram Time Basis

Notch Position (index)	Nominal Scram Timing (seconds)	Optimum Scram Timing (seconds)
46	0.420	0.380
36	0.980	0.875
26	1.600	1.465
6	2.900	2.900

In demonstrating compliance with the NSS and/or OSS scram time basis, surveillance requirements from Technical Specification 3.1.4 apply; accepting the definition of SLOW rods should conform to scram speeds shown in Table 4.1. If conformance is not demonstrated, TSSS based $MCPR_P$ limits are applied.

On initial cycle startup, TSSS limits are used until the successful completion of scram timing confirms NSS and/or OSS based limits are applicable.

4.2.3 Exposure Dependent Limits

Exposures are tracked on a Core Average Exposure basis (CAVEX, not Cycle Exposure). Higher exposure $MCPR_P$ limits are always more limiting and may be used for any Core Average Exposure up to the ending exposure. Per Reference 1, $MCPR_P$ limits are provided for the following exposure ranges:

BOC to NEOC	NEOC corresponds to	**27,393.0 MWd / MTU**
BOC to EOCLB	EOCLB corresponds to	**31,304.9 MWd / MTU**
BOC to End of Coast	End of Coast	**33,000.0 MWd / MTU**

NEOC refers to a Near EOC exposure point.

* Reference 1 analysis results are based on information identified in Reference 4.

† Drop out times consistent with method used to perform actual timing measurements (i.e., including pickup/dropout effects).

The EOCLB exposure point is not the true End-Of-Cycle exposure. Instead it corresponds to a licensing exposure window exceeding expected end-of-full-power-life.

The End of Coast exposure point represents a licensing exposure point exceeding the expected end-of-cycle exposure including cycle extension options.

4.2.4 Equipment Out-Of-Service (EOOS) Options

EOOS options* covered by $MCPR_P$ limits are given by the following:

In-Service	All equipment In-Service
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
RPTOOS+TBVOOS	Combined RPTOOS and TBVOOS
PLUOOS	Power Load Unbalance Out-Of-Service
PLUOOS+RPTOOS	Combined PLUOOS and RPTOOS
PLUOOS+TBVOOS	Combined PLUOOS and TBVOOS
PLUOOS+TBVOOS+RPTOOS	Combined PLUOOS, RPTOOS, and TBVOOS
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service (or Final Feedwater Temperature Reduction)

For exposure ranges up to NEOC and EOCLB, additional combinations of $MCPR_P$ limits are also provided including FHOOS. The coast down exposure range assumes application of FFWTR. FHOOS based $MCPR_P$ limits for the coast down exposure are redundant because the temperature setdown assumption is identical with FFWTR.

4.2.5 Single-Loop-Operation (SLO) Limits

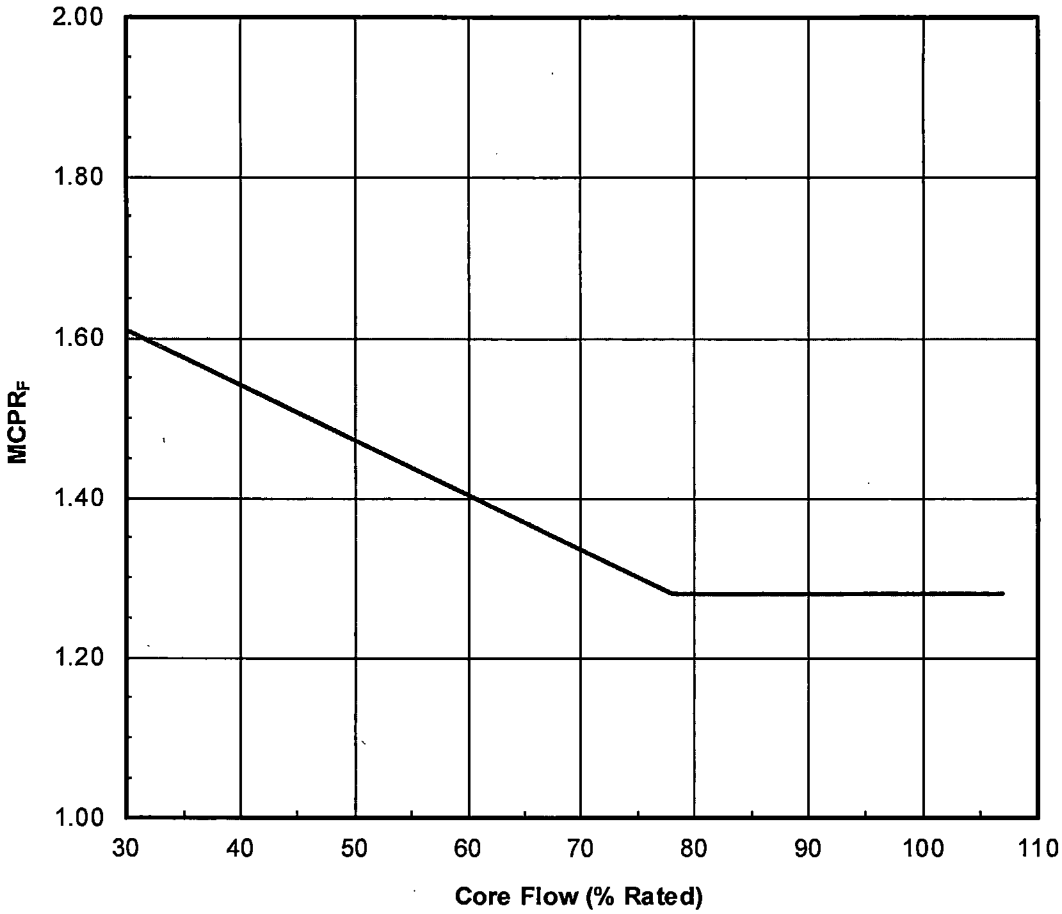
When operating in RCPOOS conditions, MCPRp limits are constructed differently from the normal operating RCP conditions. The limiting event for RCPOOS is a pump seizure scenario, which sets the upper bound for allowed core power and flow†. This event is not impacted by scram time assumptions. Specific $MCPR_P$ limits are shown in Table 4.9.

4.2.6 Below Pbypass Limits

Below Pbypass (30% rated power), $MCPR_P$ limits depend upon core flow. One set of $MCPR_P$ limits applies for core flow above 50% of rated; a second set applies if the core flow is less than or equal to 50% rated.

* All equipment service conditions assume 1 SRVOOS.

† RCPOOS limits are only valid up to 50% rated core power, 50% rated core flow, and an active recirculation drive flow of 17.73 Mlb_m/hr.



Core Flow (% Rated)	$MCPR_F$
30.0	1.61
78.0	1.28
107.0	1.28

Figure 4.1 $MCPR_F$ for ATRIUM-10 Fuel
(*Values bound all EOOS conditions*)

(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)

Table 4.2 $MCPR_P$ Limits for Optimum Scram Time Basis*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
Base Case	100	1.38	1.41	1.43
	75	1.51	1.51	1.55
	65	1.57	1.57	1.61
	50	1.70	1.70	1.76
	50	1.93	1.93	1.93
	40	2.03	2.03	2.03
	30	2.19	2.19	2.30
	30 at > 50%F	2.53	2.53	2.63
	25 at > 50%F	2.77	2.77	2.89
	30 at ≤ 50%F	2.45	2.45	2.52
	25 at ≤ 50%F	2.68	2.68	2.80
FHOOS	100	1.40	1.43	---
	75	1.55	1.55	---
	65	1.61	1.61	---
	50	1.76	1.76	---
	50	1.93	1.93	---
	40	2.03	2.03	---
	30	2.30	2.30	---
	30 at > 50%F	2.63	2.63	---
	25 at > 50%F	2.89	2.89	---
	30 at ≤ 50%F	2.52	2.52	---
	25 at ≤ 50%F	2.80	2.80	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR/FHOOS is supported for the BOC to End of Coast limits.

Table 4.3 MCPR$_P$ Limits for Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
Base Case	100	1.40	1.42	1.43
	75	1.53	1.53	1.56
	65	1.59	1.59	1.62
	50	1.72	1.72	1.79
	50	1.93	1.93	1.94
	40	2.04	2.04	2.04
	30	2.22	2.22	2.33
	30 at > 50%F	2.53	2.53	2.63
	25 at > 50%F	2.77	2.77	2.89
	30 at ≤ 50%F	2.45	2.45	2.52
	25 at ≤ 50%F	2.68	2.68	2.80
TBVOOS	100	1.44	1.46	1.47
	75	1.57	1.57	1.60
	65	1.62	1.63	1.66
	50	1.75	1.75	1.81
	50	1.93	1.93	1.94
	40	2.04	2.04	2.04
	30	2.23	2.23	2.34
	30 at > 50%F	3.14	3.14	3.26
	25 at > 50%F	3.53	3.53	3.64
	30 at ≤ 50%F	2.74	2.74	2.88
	25 at ≤ 50%F	3.17	3.17	3.32
FHOOS	100	1.43	1.43	---
	75	1.55	1.56	---
	65	1.62	1.62	---
	50	1.79	1.79	---
	50	1.94	1.94	---
	40	2.04	2.04	---
	30	2.33	2.33	---
	30 at > 50%F	2.63	2.63	---
	25 at > 50%F	2.89	2.89	---
	30 at ≤ 50%F	2.52	2.52	---
	25 at ≤ 50%F	2.80	2.80	---
PLUOOS	100	1.40	1.42	1.43
	75	1.53	1.53	1.56
	65	1.82	1.82	1.83
	50	---	---	---
	50	1.94	1.94	1.94
	40	2.04	2.04	2.04
	30	2.22	2.22	2.33
	30 at > 50%F	2.53	2.53	2.63
	25 at > 50%F	2.77	2.77	2.89
	30 at ≤ 50%F	2.45	2.45	2.52
	25 at ≤ 50%F	2.68	2.68	2.80

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

A 50% power step change for PLUOOS limits is not supported. When core power is ≤ 50%, the LRNB event is the same with, or without PLUOOS.

Table 4.3 MCPR$_P$ Limits for Nominal Scram Time Basis (*continued*)*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVOOS FHOOS	100	1.46	1.47	---
	75	1.59	1.60	---
	65	1.66	1.66	---
	50	1.81	1.81	---
	50	1.94	1.94	---
	40	2.04	2.04	---
	30	2.34	2.34	---
	30 at > 50%F	3.26	3.26	---
	25 at > 50%F	3.64	3.64	---
	30 at ≤ 50%F	2.88	2.88	---
	25 at ≤ 50%F	3.32	3.32	---
TBVOOS PLUOOS	100	1.44	1.46	1.47
	75	1.57	1.57	1.60
	65	1.82	1.82	1.83
	50	---	---	---
	50	1.94	1.94	1.94
	40	2.04	2.04	2.04
	30	2.23	2.23	2.34
	30 at > 50%F	3.14	3.14	3.26
	25 at > 50%F	3.53	3.53	3.64
	30 at ≤ 50%F	2.74	2.74	2.88
	25 at ≤ 50%F	3.17	3.17	3.32
FHOOS PLUOOS	100	1.43	1.43	---
	75	1.55	1.56	---
	65	1.83	1.83	---
	50	---	---	---
	50	1.94	1.94	---
	40	2.04	2.04	---
	30	2.33	2.33	---
	30 at > 50%F	2.63	2.63	---
	25 at > 50%F	2.89	2.89	---
	30 at ≤ 50%F	2.52	2.52	---
	25 at ≤ 50%F	2.80	2.80	---
TBVOOS FHOOS PLUOOS	100	1.46	1.47	---
	75	1.59	1.60	---
	65	1.83	1.83	---
	50	---	---	---
	50	1.94	1.94	---
	40	2.04	2.04	---
	30	2.34	2.34	---
	30 at > 50%F	3.26	3.26	---
	25 at > 50%F	3.64	3.64	---
	30 at ≤ 50%F	2.88	2.88	---
	25 at ≤ 50%F	3.32	3.32	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

A 50% power step change for PLUOOS limits is not supported. When core power is ≤ 50%, the LRNB event is the same with, or without PLUOOS.

Table 4.4 MCPR$_P$ Limits for Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
Base Case	100	1.42	1.43	1.44
	75	1.55	1.55	1.57
	65	1.60	1.60	1.64
	50	1.75	1.75	1.82
	50	1.94	1.94	1.95
	40	2.05	2.05	2.05
	30	2.24	2.24	2.36
	30 at > 50%F	2.53	2.53	2.63
	25 at > 50%F	2.77	2.77	2.89
	30 at ≤ 50%F	2.45	2.45	2.52
	25 at ≤ 50%F	2.68	2.68	2.80
TBVOOS	100	1.46	1.47	1.48
	75	1.59	1.59	1.61
	65	1.64	1.64	1.68
	50	1.77	1.77	1.83
	50	1.94	1.94	1.95
	40	2.05	2.05	2.07
	30	2.26	2.26	2.37
	30 at > 50%F	3.14	3.14	3.26
	25 at > 50%F	3.53	3.53	3.64
	30 at ≤ 50%F	2.74	2.74	2.88
	25 at ≤ 50%F	3.17	3.17	3.32
FHOOS	100	1.44	1.44	---
	75	1.57	1.57	---
	65	1.64	1.64	---
	50	1.82	1.82	---
	50	1.95	1.95	---
	40	2.05	2.05	---
	30	2.36	2.36	---
	30 at > 50%F	2.63	2.63	---
	25 at > 50%F	2.89	2.89	---
	30 at ≤ 50%F	2.52	2.52	---
	25 at ≤ 50%F	2.80	2.80	---
PLUOOS	100	1.42	1.43	1.44
	75	1.55	1.55	1.57
	65	1.83	1.83	1.84
	50	---	---	---
	50	1.95	1.95	1.95
	40	2.05	2.05	2.05
	30	2.24	2.24	2.36
	30 at > 50%F	2.53	2.53	2.63
	25 at > 50%F	2.77	2.77	2.89
	30 at ≤ 50%F	2.45	2.45	2.52
	25 at ≤ 50%F	2.68	2.68	2.80

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

A 50% power step change for PLUOOS limits is not supported. When core power is ≤ 50%, the LRNB event is the same with, or without PLUOOS.

Table 4.4 MCPR$_P$ Limits for Technical Specification Scram Time Basis (*continued*)*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVOOS FHOOS	100	1.48	1.48	---
	75	1.61	1.61	---
	65	1.68	1.68	---
	50	1.83	1.83	---
	50	1.95	1.95	---
	40	2.07	2.07	---
	30	2.37	2.37	---
	30 at > 50%F	3.26	3.26	---
	25 at > 50%F	3.64	3.64	---
	30 at ≤ 50%F	2.88	2.88	---
	25 at ≤ 50%F	3.32	3.32	---
TBVOOS PLUOOS	100	1.46	1.47	1.48
	75	1.59	1.59	1.61
	65	1.83	1.83	1.84
	50	---	---	---
	50	1.95	1.95	1.95
	40	2.05	2.05	2.07
	30	2.26	2.26	2.37
	30 at > 50%F	3.14	3.14	3.26
	25 at > 50%F	3.53	3.53	3.64
	30 at ≤ 50%F	2.74	2.74	2.88
	25 at ≤ 50%F	3.17	3.17	3.32
FHOOS PLUOOS	100	1.44	1.44	---
	75	1.57	1.57	---
	65	1.84	1.84	---
	50	---	---	---
	50	1.95	1.95	---
	40	2.05	2.05	---
	30	2.36	2.36	---
	30 at > 50%F	2.63	2.63	---
	25 at > 50%F	2.89	2.89	---
	30 at ≤ 50%F	2.52	2.52	---
	25 at ≤ 50%F	2.80	2.80	---
TBVOOS FHOOS PLUOOS	100	1.48	1.48	---
	75	1.61	1.61	---
	65	1.84	1.84	---
	50	---	---	---
	50	1.95	1.95	---
	40	2.07	2.07	---
	30	2.37	2.37	---
	30 at > 50%F	3.26	3.26	---
	25 at > 50%F	3.64	3.64	---
	30 at ≤ 50%F	2.88	2.88	---
	25 at ≤ 50%F	3.32	3.32	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

A 50% power step change for PLUOOS limits is not supported. When core power is ≤ 50%, the LRNB event is the same with, or without PLUOOS.

Table 4.5 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 1: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVIS	100	1.44	1.44	1.44
	75	1.57	1.57	1.57
	65	1.84	1.84	1.84
	50	1.95	1.95	1.95
	50	1.99	1.99	1.99
	40	2.24	2.24	2.24
	30	2.61	2.61	2.61
	30 at > 50%F	2.88	2.88	2.88
	25 at > 50%F	3.21	3.21	3.21
	30 at ≤ 50%F	2.79	2.79	2.79
	25 at ≤ 50%F	3.07	3.07	3.07
TBVOOS	100	1.48	1.48	1.48
	75	1.61	1.61	1.61
	65	1.84	1.84	1.84
	50	1.95	1.95	1.95
	50	1.99	1.99	1.99
	40	2.25	2.25	2.25
	30	2.61	2.61	2.61
	30 at > 50%F	3.44	3.44	3.44
	25 at > 50%F	3.85	3.85	3.85
	30 at ≤ 50%F	3.10	3.10	3.10
	25 at ≤ 50%F	3.54	3.54	3.54

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Table 4.6 Startup Operation MCPR$_P$ Limits for Table 3.1 Temperature Range 2: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVIS	100	1.44	1.44	1.44
	75	1.57	1.57	1.57
	65	1.84	1.84	1.84
	50	1.95	1.95	1.95
	50	2.00	2.00	2.00
	40	2.26	2.26	2.26
	30	2.63	2.63	2.63
	30 at > 50%F	2.90	2.90	2.90
	25 at > 50%F	3.23	3.23	3.23
	30 at ≤ 50%F	2.80	2.80	2.80
	25 at ≤ 50%F	3.09	3.09	3.09
TBVOOS	100	1.48	1.48	1.48
	75	1.61	1.61	1.61
	65	1.84	1.84	1.84
	50	1.95	1.95	1.95
	50	2.00	2.00	2.00
	40	2.26	2.26	2.26
	30	2.63	2.63	2.63
	30 at > 50%F	3.45	3.45	3.45
	25 at > 50%F	3.86	3.86	3.86
	30 at ≤ 50%F	3.12	3.12	3.12
	25 at ≤ 50%F	3.56	3.56	3.56

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Table 4.7 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 1: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVIS	100	1.43	1.43	1.43
	75	1.55	1.56	1.56
	65	1.83	1.83	1.83
	50	1.94	1.94	1.94
	50	1.96	1.96	1.96
	40	2.22	2.22	2.22
	30	2.58	2.58	2.58
	30 at > 50%F	2.88	2.88	2.88
	25 at > 50%F	3.21	3.21	3.21
	30 at ≤ 50%F	2.79	2.79	2.79
	25 at ≤ 50%F	3.07	3.07	3.07
TBVOOS	100	1.46	1.47	1.47
	75	1.59	1.60	1.60
	65	1.83	1.83	1.83
	50	1.94	1.94	1.94
	50	1.96	1.96	1.96
	40	2.22	2.22	2.22
	30	2.58	2.58	2.58
	30 at > 50%F	3.44	3.44	3.44
	25 at > 50%F	3.85	3.85	3.85
	30 at ≤ 50%F	3.10	3.10	3.10
	25 at ≤ 50%F	3.54	3.54	3.54

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Table 4.8 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 2: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	BOC to NEOC	BOC to EOCLB	BOC to End of Coast
TBVIS	100	1.43	1.43	1.43
	75	1.55	1.56	1.56
	65	1.83	1.83	1.83
	50	1.94	1.94	1.94
	50	1.97	1.97	1.97
	40	2.23	2.23	2.23
	30	2.60	2.60	2.60
	30 at > 50%F	2.90	2.90	2.90
	25 at > 50%F	3.23	3.23	3.23
	30 at ≤ 50%F	2.80	2.80	2.80
	25 at ≤ 50%F	3.09	3.09	3.09
TBVOOS	100	1.46	1.47	1.47
	75	1.59	1.60	1.60
	65	1.83	1.83	1.83
	50	1.94	1.94	1.94
	50	1.97	1.97	1.97
	40	2.23	2.23	2.23
	30	2.60	2.60	2.60
	30 at > 50%F	3.45	3.45	3.45
	25 at > 50%F	3.86	3.86	3.86
	30 at ≤ 50%F	3.12	3.12	3.12
	25 at ≤ 50%F	3.56	3.56	3.56

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Table 4.9 $MCPR_P$ Limits for Single Loop Operation for All Scram Times*

Operating Condition	Power (% of rated)	BOC to End of Coast
RCPOOS FHOOS	100	2.00
	50	2.00
	40	2.07
	30	2.38
	30 at > 50%F	2.65
	25 at > 50%F	2.91
	30 at ≤ 50%F	2.54
	25 at ≤ 50%F	2.82
RCPOOS TBVOOS PLUOOS FHOOS	100	2.00
	50	2.00
	40	2.09
	30	2.39
	30 at > 50%F	3.28
	25 at > 50%F	3.66
	30 at ≤ 50%F	2.90
	25 at ≤ 50%F	3.34
RCPOOS TBVOOS FHOOS1	100	2.01
	50	2.01
	40	2.27
	30	2.63
	30 at > 50%F	3.46
	25 at > 50%F	3.87
	30 at ≤ 50%F	3.12
	25 at ≤ 50%F	3.56
RCPOOS TBVOOS FHOOS2	100	2.02
	50	2.02
	40	2.28
	30	2.65
	30 at > 50%F	3.47
	25 at > 50%F	3.88
	30 at ≤ 50%F	3.14
	25 at ≤ 50%F	3.58

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop.

RCPOOS limits are only valid up to 50% rated core power, 50% rated core flow, and an active recirculation drive flow of 17.73 Mlb_m/hr.

5 Oscillation Power Range Monitor (OPRM) Setpoint

(Technical Specification 3.3.1.1)

Technical Specification Table 3.3.1.1-1, Function 2f, identifies the OPRM upscale function.

Instrument setpoints are established, such that the reactor will be tripped before an oscillation can grow to the point where the SLMCPR is exceeded. An Option III stability analysis is performed for each reload core to determine allowable OLMCPR's as a function of OPRM setpoint. Analyses consider both steady state startup operation, and the case of a two recirculation pump trip from rated power.

The resulting stability based OLMCPR's are reported in Reference 1. The OPRM setpoint (*sometimes referred to as the Amplitude Trip, S_p*) is selected, such that required margin to the SLMCPR is provided without stability being a limiting event. Analyses are based on cycle specific DIVOM analyses performed per Reference 22. The calculated OLMCPR's are shown in Table 5.1. Review of results shown in Table 4.2 indicates an OPRM setpoint of **1.14** may be used. The successive confirmation count (*sometimes referred to as N_p*) is provided in Table 5.2, per Reference 27.

Table 5.1 OPRM Setpoint Range*

OPRM Setpoint	OLMCPR (SS)	OLMCPR (2PT)
1.05	1.18	1.19
1.06	1.20	1.21
1.07	1.22	1.23
1.08	1.24	1.25
1.09	1.26	1.27
1.10	1.28	1.29
1.11	1.30	1.31
1.12	1.32	1.33
1.13	1.34	1.36
1.14	1.36	1.38
1.15	1.39	1.40

Table 5.2 OPRM Successive Confirmation Count Setpoint

Count	OPRM Setpoint
6	≥ 1.04
8	≥ 1.05
10	≥ 1.07
12	≥ 1.09
14	≥ 1.11
16	≥ 1.14
18	≥ 1.18
20	≥ 1.24

* Extrapolation beyond a setpoint of 1.15 is not allowed

6 APRM Flow Biased Rod Block Trip Settings

(Technical Requirements Manual Section 5.3.1 and Table 3.3.4-1)

The APRM rod block trip setting is based upon References 23 & 24, and is defined by the following:

SRB $\leq$ (0.66(W-ΔW) + 61%)	Allowable Value
SRB $\leq$ (0.66(W-ΔW) + 59%)	Nominal Trip Setpoint (NTSP)

where:

SRB	=	Rod Block setting in percent of rated thermal power (3458 MW_t)
W	=	Loop recirculation flow rate in percent of rated
ΔW	=	Difference between two-loop and single-loop effective recirculation flow at the same core flow (ΔW=0.0 for two-loop operation)

The APRM rod block trip setting is clamped at a maximum allowable value of 115% (corresponding to a NTSP of 113%).

7 Rod Block Monitor (RBM) Trip Setpoints and Operability

(Technical Specification Table 3.3.2.1-1)

The RBM trip setpoints and applicable power ranges, based on References 23 & 24, are shown in Table 7.1. Setpoints are based on an HTSP, unfiltered analytical limit of 114%. Unfiltered setpoints are consistent with a nominal RBM filter setting of 0.0 seconds; filtered setpoints are consistent with a nominal RBM filter setting less than 0.5 seconds. Cycle specific CRWE analyses of OLMCPR are documented in Reference 1, superseding values reported in References 23, 24, and 26.

Table 7.1 Analytical RBM Trip Setpoints*

RBM Trip Setpoint		Allowable Value (AV)	Nominal Trip Setpoint (NTSP)
LPSP		27%	25%
IPSP		62%	60%
HPSP		82%	80%
LTSP	- unfiltered	121.7%	120.0%
	- filtered	120.7%	119.0%
ITSP	- unfiltered	116.7%	115.0%
	- filtered	115.7%	114.0%
HTSP	- unfiltered	111.7%	110.0%
	- filtered	110.9%	109.2%
DTSP		90%	92%

As a result of cycle specific CRWE analyses, RBM setpoints in Technical Specification Table 3.3.2.1-1 are applicable as shown in Table 7.2. Cycle specific setpoint analysis results are shown in Table 7.3, per Reference 1.

Table 7.2 RBM Setpoint Applicability

Thermal Power (% Rated)	Applicable MCPR†	Notes from Table 3.3.2.1-1	Comment
> 27% and < 90%	< 1.74	(a), (b), (f), (h)	two loop operation
	< 1.77	(a), (b), (f), (h)	single loop operation
≥ 90%	< 1.43	(g)	two loop operation‡

* Values are considered maximums. Using lower values, due to RBM system hardware/software limitations, is conservative, and acceptable.

† MCPR values shown correspond with, (support), SLMPCR values identified in Reference 1.

‡ Greater than 90% rated power is not attainable in single loop operation.

Table 7.3 Control Rod Withdrawal Error Results

RBM HTSP Analytical Limit	CRWE OLMCPR
Unfiltered	
107	1.28
111	1.31
114	1.33
117	1.35

Results, compared against the base case OLMCPR results of Table 4.2, indicate SLMCPR remains protected for RBM inoperable conditions (i.e., 114% unblocked).

8 Shutdown Margin Limit

(Technical Specification 3.1.1)

Assuming the strongest OPERABLE control blade is fully withdrawn, and all other OPERABLE control blades are fully inserted, the core shall be sub-critical and meet the following minimum shutdown margin:

SDM > 0.38% dk/k

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Browns Ferry Unit 3 Cycle 18

Core Operating Limits Report, (105% OLTP)

TVA-COLR-BF3C18 **Revision 0 (Final)**
(Revision Log, Page v)

March 2016

Prepared: ______________________ Date: 3/6/16
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Approved: ______________________ Date: 3-7-16
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Nomenclature

APLHGR	Average Planar LHGR
APRM	Average Power Range Monitor
AREVA NP	Vendor (Framatome, Siemens)
BOC	Beginning of Cycle
BSP	Backup Stability Protection
BWR	Boiling Water Reactor
CAVEX	Core Average Exposure
CD	Coast Down
CMSS	Core Monitoring System Software
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRWE	Control Rod Withdrawal Error
CSDM	Cold SDM
DIVOM	Delta CPR over Initial CPR vs. Oscillation Magnitude
EOC	End of Cycle
EOCLB	End-of-Cycle Licensing Basis
EOOS	Equipment OOS
FFTR	Final Feedwater Temperature Reduction
FFWTR	Final Feedwater Temperature Reduction
FHOOS	Feedwater Heaters OOS
ft	Foot: english unit of measure for length
GNF	Vendor (General Electric, Global Nuclear Fuels)
GWd	Giga Watt Day
HTSP	High TSP
ICA	Interim Corrective Action
ICF	Increased Core Flow (beyond rated)
IS	In-Service
kW	kilo watt: SI unit of measure for power.
LCO	License Condition of Operation
LFWH	Loss of Feedwater Heating
LHGRFAC	LHGR Multiplier (Power or Flow dependent)
LPRM	Low Power Range Monitor
LRNB	Generator Load Reject, No Bypass
MAPFAC	MAPLHGR multiplier (Power or Flow dependent)
MCPR	Minimum CPR

MSRV	Moisture Separator Reheater Valve
MSRVOOS	MSRV OOS
MTU	Metric Ton Uranium
MWd/MTU	Mega Watt Day per Metric Ton Uranium
NEOC	Near EOC
NRC	United States Nuclear Regulatory Commission
NSS	Nominal Scram Speed
NTSP	Nominal TSP
OLMCPR	MCPR Operating Limit
OOS	Out-Of-Service
OPRM	Oscillation Power Range Monitor
OSS	Optimum Scram Speed
PBDA	Period Based Detection Algorithm
Pbypass	Power, below which TSV Position and TCV Fast Closure Scrams are Bypassed
PLU	Power Load Unbalance
PLUOOS	PLU OOS
PRNM	Power Range Neutron Monitor
RBM	Rod Block Monitor
RCPOOS	Recirculation Pump OOS (*SLO*)
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPTOOS	RPT OOS
SDM	Shutdown Margin
SLMCPR	MCPR Safety Limit
SLO	Single Loop Operation
TBV	Turbine Bypass Valve
TBVIS	TBV IS
TBVOOS	Turbine Bypass Valves OOS
TIP	Transversing In-core Probe
TIPOOS	TIP OOS
TLO	Two Loop Operation
TSP	Trip Setpoint
TSSS	Technical Specification Scram Speed
TVA	Tennessee Valley Authority

References

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Methodology References

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

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

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

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

1 Introduction

In anticipation of cycle startup, it is necessary to describe the expected limits of operation.

1.1 Purpose

The primary purpose of this document is to satisfy requirements identified by unit technical specification section 5.6.5. This document may be provided, upon final approval, to the NRC.

1.2 Scope

This version of the COLR is specifically intended to support the refueling outage. Consequently, this document only covers MODE 3, 4 & 5 operation of the unit. Expanding operation for MODE 1, and 2 will be addressed in a future COLR revision. This document will discuss the following areas:

- Shutdown Margin (SDM) Limit
  (Technical Specification 3.1.1)
  Applicability: All Modes

1.3 Fuel Loading

The core will contain previously exposed AREVA NP, Inc., ATRIUM-10 fuel, along with fresh ATRIUM-10XM. Nuclear fuel types used in the core loading are shown in Table 1.1. The planned outage will consist of a full core off-load for maintenance. The final core loading was evaluated for BOC cold shutdown margin performance as documented per Reference 1.

1.4 Acceptability

Limits discussed in this document were generated based on NRC approved methodologies per References 2 through 21.

Table 1.1 Nuclear Fuel Types*

Fuel Description	Original Cycle	Number of Assemblies	Nuclear Fuel Type (NFT)	Fuel Names (Range)
ATRIUM-10 A10-3440B-11GV80-FCE	16	70	12	FCE002-FCE144
ATRIUM-10 A10-3826B-13GV80-FCE	16	39	13	FCE145-FCE188
ATRIUM-10 A10-4075B-13GV80-FCE	16	15	14	FCE190-FCE235
ATRIUM-10 A10-4081B-12GV80-FCE	16	48	15	FCE237-FCE284
ATRIUM-10 A10-3849B-13GV80-FCF	17	176	16	FCF301-FCF476
ATRIUM-10 A10-3882B-10GV70-FCF	17	40	17	FCF477-FCF516
ATRIUM-10 A10-4116B-12GV70-FCF	17	72	18	FCF517-FCF588
ATRIUM-10XM XMLC-4105B-11GV70-FCG	18	72	19	FCG601-FCG672
ATRIUM-10XM XMLC-4096B-12GV80-FCG	18	136	20	FCG673-FCG808
ATRIUM-10XM XMLC-4055B-13GV70-FCG	18	96	21	FCG809-FCG904

* The table identifies the expected fuel type breakdown in anticipation of final core loading. The final composition of the core depends upon uncertainties during the outage such as discovering a failed fuel bundle, or other bundle damage. Minor core loading changes, due to unforeseen events, will conform to the safety and monitoring requirements identified in this document.

2 Shutdown Margin Limit

(Technical Specification 3.1.1)

Assuming the strongest OPERABLE control blade is fully withdrawn, and all other OPERABLE control blades are fully inserted, the core shall be sub-critical and meet the following minimum shutdown margin:

SDM > 0.38% dk/k

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Verified:	B. C. Mitchell, Engineer	Date:	3/15/16
Approved:	G. C. Storey, Manager, BWR Fuel Engineering	Date:	3/15/16
Reviewed:	D. D. Coffey, Manager, Reactor Engineering	Date:	3/15/16
Approved:	Chairman, PORC	Date:	3/16/16 ~~3/15/16~~ 3/16/16
Approved:	Plant Manager	Date:	3.16.16

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0-R1	All	This is a major revision supporting all modes of operation, but is limited to BOC through NEOC cycle exposure. A future Revision will extend operation to expected end of cycle, and final feedwater temperature reduction and power coastdown. Cross reference to CR 1145408.
1-R1	3-37	Previous Section 2 becomes Section 8. New material for Sections 2 through 7.
0-R0	All	New document.

Nomenclature

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LHGRFAC	LHGR Multiplier (Power or Flow dependent)
LPRM	Low Power Range Monitor
LRNB	Generator Load Reject, No Bypass
MAPFAC	MAPLHGR multiplier (Power or Flow dependent)
MCPR	Minimum CPR

MSRV	Moisture Separator Reheater Valve
MSRVOOS	MSRV OOS
MTU	Metric Ton Uranium
MWd/MTU	Mega Watt Day per Metric Ton Uranium
NEOC	Near EOC
NRC	United States Nuclear Regulatory Commission
NSS	Nominal Scram Speed
NTSP	Nominal TSP
OLMCPR	MCPR Operating Limit
OOS	Out-Of-Service
OPRM	Oscillation Power Range Monitor
OSS	Optimum Scram Speed
PBDA	Period Based Detection Algorithm
Pbypass	Power, below which TSV Position and TCV Fast Closure Scrams are Bypassed
PLU	Power Load Unbalance
PLUOOS	PLU OOS
PRNM	Power Range Neutron Monitor
RBM	Rod Block Monitor
RCPOOS	Recirculation Pump OOS (*SLO*)
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPTOOS	RPT OOS
SDM	Shutdown Margin
SLMCPR	MCPR Safety Limit
SLO	Single Loop Operation
TBV	Turbine Bypass Valve
TBVIS	TBV IS
TBVOOS	Turbine Bypass Valves OOS
TIP	Transversing In-core Probe
TIPOOS	TIP OOS
TLO	Two Loop Operation
TSP	Trip Setpoint
TSSS	Technical Specification Scram Speed
TVA	Tennessee Valley Authority

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1 Introduction

In anticipation of cycle startup, it is necessary to describe the expected limits of operation.

1.1 Purpose

The primary purpose of this document is to satisfy requirements identified by unit technical specification section 5.6.5. This document may be provided, upon final approval, to the NRC.

1.2 Scope

This document will discuss the following areas:

- Average Planar Linear Heat Generation Rate (APLHGR) Limit
  (Technical Specifications 3.2.1 and 3.7.5)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Linear Heat Generation Rate (LHGR) Limit
  (Technical Specification 3.2.3, 3.3.4.1, and 3.7.5)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Minimum Critical Power Ratio Operating Limit (OLMCPR)
  (Technical Specifications 3.2.2, 3.3.4.1, 3.7.5 and Table 3.3.2.1-1)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Oscillation Power Range Monitor (OPRM) Setpoint
  (Technical Specification Table 3.3.1.1)
  Applicability: Mode 1, ≥ (as specified in Technical Specifications Table 3.3.1.1-1)
- Average Power Range Monitor (APRM) Flow Biased Rod Block Trip Setting
  (Technical Requirements Manual Section 5.3.1 and Table 3.3.4-1)
  Applicability: Mode 1, ≥ (as specified in Technical Requirements Manuals Table 3.3.4-1)
- Rod Block Monitor (RBM) Trip Setpoints and Operability
  (Technical Specification Table 3.3.2.1-1)
  Applicability: Mode 1, ≥ % RTP as specified in Table 3.3.2.1-1 (TS definition of RTP)
- Shutdown Margin (SDM) Limit
  (Technical Specification 3.1.1)
  Applicability: All Modes

1.3 Fuel Loading

The core will contain previously exposed AREVA NP, Inc., ATRIUM-10 fuel, along with fresh ATRIUM-10XM. Nuclear fuel types used in the core loading are shown in Table 1.1. The planned outage will consist of a full core off-load for maintenance. The final core loading was evaluated for BOC cold shutdown margin performance as documented per Reference 5.

Table 1.1 Nuclear Fuel Types*

Fuel Description	Original Cycle	Number of Assemblies	Nuclear Fuel Type (NFT)	Fuel Names (Range)
ATRIUM-10 A10-3440B-11GV80-FCE	16	70	12	FCE002-FCE144
ATRIUM-10 A10-3826B-13GV80-FCE	16	39	13	FCE145-FCE188
ATRIUM-10 A10-4075B-13GV80-FCE	16	15	14	FCE190-FCE235
ATRIUM-10 A10-4081B-12GV80-FCE	16	48	15	FCE237-FCE284
ATRIUM-10 A10-3849B-13GV80-FCF	17	176	16	FCF301-FCF476
ATRIUM-10 A10-3882B-10GV70-FCF	17	40	17	FCF477-FCF516
ATRIUM-10 A10-4116B-12GV70-FCF	17	72	18	FCF517-FCF588
ATRIUM-10XM XMLC-4105B-11GV70-FCG	18	72	19	FCG601-FCG672
ATRIUM-10XM XMLC-4096B-12GV80-FCG	18	136	20	FCG673-FCG808
ATRIUM-10XM XMLC-4055B-13GV70-FCG	18	96	21	FCG809-FCG904

1.4 Acceptability

Limits discussed in this document were generated based on NRC approved methodologies per References 6 through 25.

* The table identifies the expected fuel type breakdown in anticipation of final core loading. The final composition of the core depends upon uncertainties during the outage such as discovering a failed fuel bundle, or other bundle damage. Minor core loading changes, due to unforeseen events, will conform to the safety and monitoring requirements identified in this document.

2 APLHGR Limits

(Technical Specifications 3.2.1 & 3.7.5)

The APLHGR limit is determined by adjusting the rated power APLHGR limit for off-rated power, off-rated flow, and SLO conditions. The most limiting of these is then used as follows:

$$\text{APLHGR limit} = \text{MIN}\ (\text{APLHGR}_P\ ,\ \text{APLHGR}_F,\ \text{APLHGR}_{SLO}\ )$$

where:

$APLHGR_P$	off-rated power APLHGR limit	[$APLHGR_{RATED}$ * $MAPFAC_P$]
$APLHGR_F$	off-rated flow APLHGR limit	[$APLHGR_{RATED}$ * $MAPFAC_F$]
$APLHGR_{SLO}$	SLO APLHGR limit	[$APLHGR_{RATED}$ * SLO Multiplier]

2.1 Rated Power and Flow Limit: $APLHGR_{RATED}$

The rated conditions APLHGR for ATRIUM-10 fuel is identified in Reference 1 and shown in Figure 2.1. The rated conditions APLHGR for ATRIUM-10XM are shown in Figure 2.2.

2.2 Off-Rated Power Dependent Limit: $APLHGR_P$

Reference 1 does not specify a power dependent APLHGR. Therefore, $MAPFAC_P$ is set to a value of **1.0**.

2.2.1 Startup without Feedwater Heaters

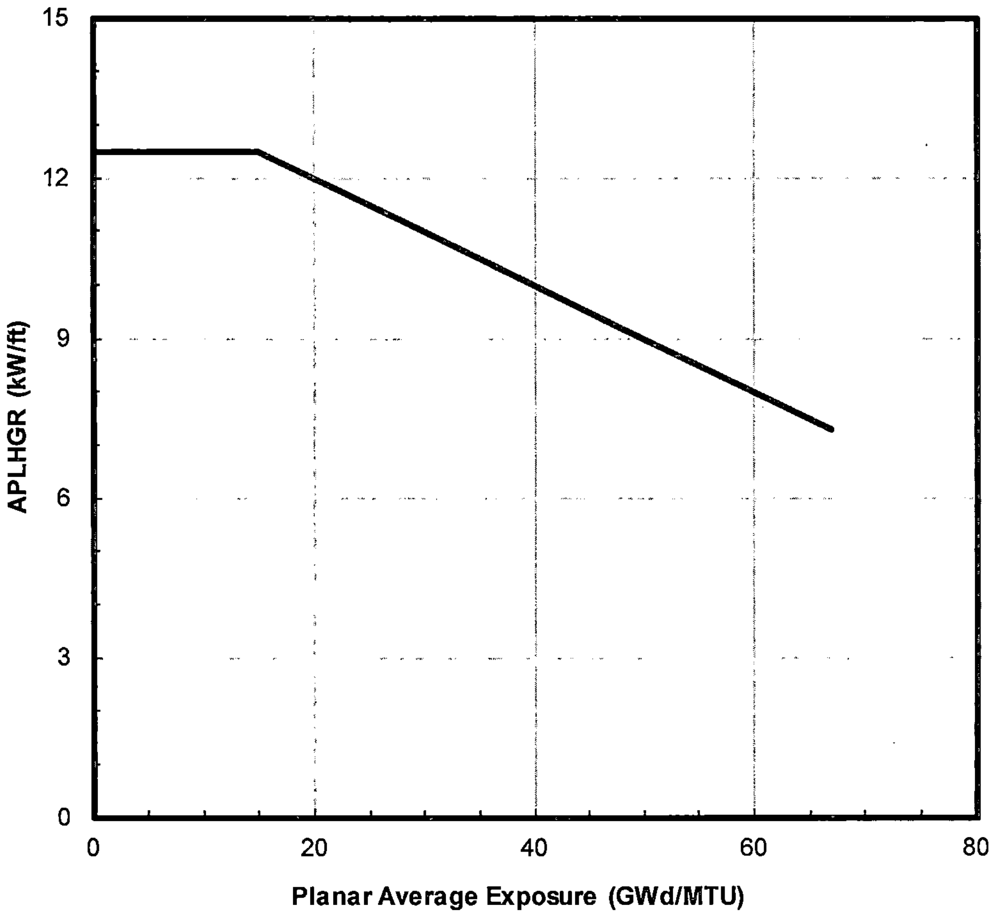
There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. No additional power dependent limitation is required.

2.3 Off-Rated Flow Dependent Limit: $APLHGR_F$

Reference 1 does not specify a flow dependent APLHGR. Therefore, $MAPFAC_F$ is set to a value of **1.0**.

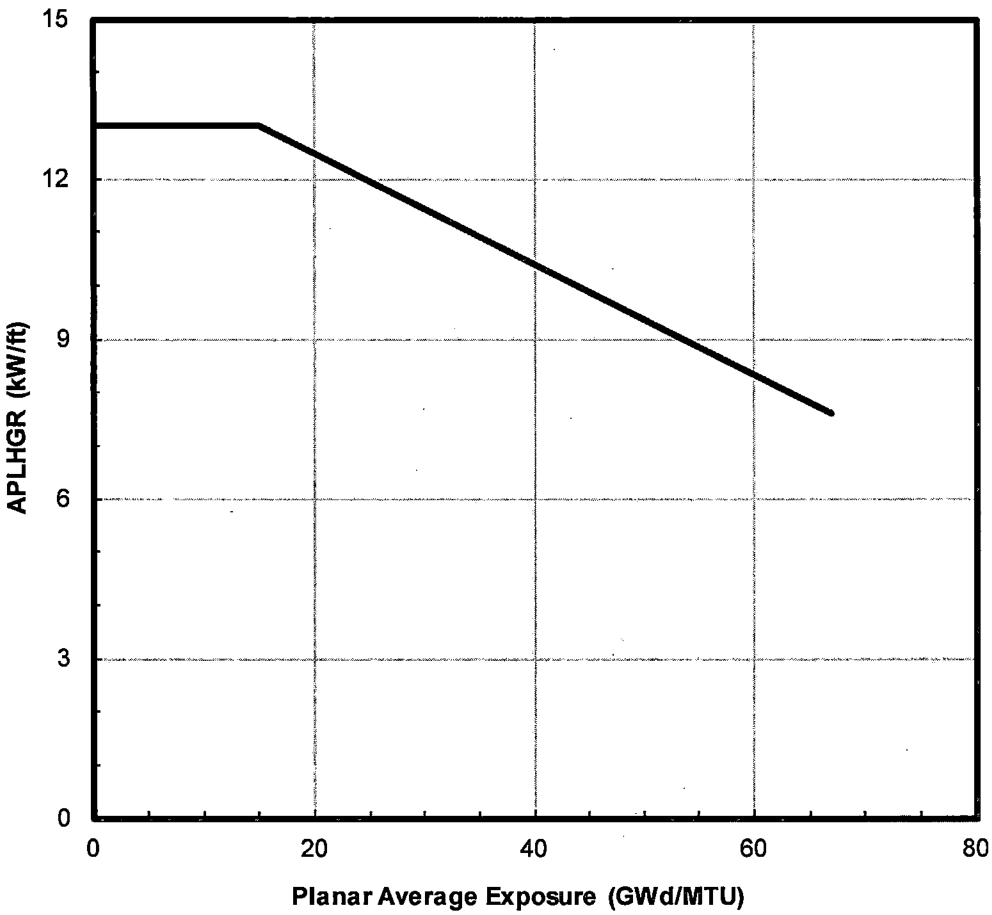
2.4 Single Loop Operation Limit: $APLHGR_{SLO}$

The single loop operation multiplier for ATRIUM-10, and ATRIUM-10XM fuel is **0.85**, per Reference 1.



Planar Avg. Exposure (GWd/MTU)	APLHGR Limit (kW/ft)
0.0	12.5
15.0	12.5
67.0	7.3

Figure 2.1 APLHGR$_{RATED}$ for ATRIUM-10 Fuel



Planar Avg. Exposure (GWd/MTU)	APLHGR Limit (kW/ft)
0.0	13.0
15.0	13.0
67.0	7.6

Figure 2.2 APLHGR$_{RATED}$ for ATRIUM-10XM Fuel

2.5 Equipment Out-Of-Service Corrections

The limits shown in Figure 2.1 and Figure 2.2 are applicable for operation with all equipment In-Service as well as the following Equipment Out-Of-Service (EOOS) options; including combinations of the options.

In-Service	All equipment In-Service*
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
PLUOOS	Power Load Unbalance Out-Of-Service
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service or Final Feedwater Temperature Reduction
RCPOOS	One Recirculation Pump Out-Of-Service

* All equipment service conditions assume 1 SRVOOS.

3 LHGR Limits

(Technical Specification 3.2.3, 3.3.4.1, & 3.7.5)

The LHGR limit is determined by adjusting the rated power LHGR limit for off-rated power and off-rated flow conditions. The most limiting of these is then used as follows:

$$\text{LHGR limit} = \text{MIN}\ (\ LHGR_P,\ LHGR_F\ )$$

where:

$LHGR_P$	off-rated power LHGR limit	$[LHGR_{RATED} * LHGRFAC_P]$
$LHGR_F$	off-rated flow LHGR limit	$[LHGR_{RATED} * LHGRFAC_F]$

3.1 Rated Power and Flow Limit: $LHGR_{RATED}$

The rated conditions LHGR for ATRIUM-10 fuel is identified in Reference 1 and shown in Figure 3.1. The rated conditions LHGR for ATRIUM-10XM fuel is shown in Figure 3.2. The LHGR limit is consistent with References 2 and 3.

3.2 Off-Rated Power Dependent Limit: $LHGR_P$

LHGR limits are adjusted for off-rated power conditions using the $LHGRFAC_P$ multiplier provided in Reference 1. The multiplier is split into two sub cases: turbine bypass valves in and out-of-service. The base case multipliers are shown in Figure 3.3 and Figure 3.4.

3.2.1 Startup without Feedwater Heaters

There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. Additional limits are shown in Figure 3.7 through Figure 3.10, based on temperature conditions identified in Table 3.1.

Table 3.1 Startup Feedwater Temperature Basis

	Temperature	
Power	**Range 1**	**Range 2**
(% Rated)	**(°F)**	**(°F)**
25	160.0	155.0
30	165.0	160.0
40	175.0	170.0
50	185.0	180.0

3.3 Off-Rated Flow Dependent Limit: $LHGR_F$

LHGR limits are adjusted for off-rated flow conditions using the $LHGRFAC_F$ multiplier provided in Reference 1. Multipliers are shown in Figure 3.5 and Figure 3.6.

3.4 Equipment Out-Of-Service Corrections

The limits shown in Figure 3.1 and Figure 3.2 are applicable for operation with all equipment In-Service as well as the following Equipment Out-Of-Service (EOOS) options; including combinations of the options.*

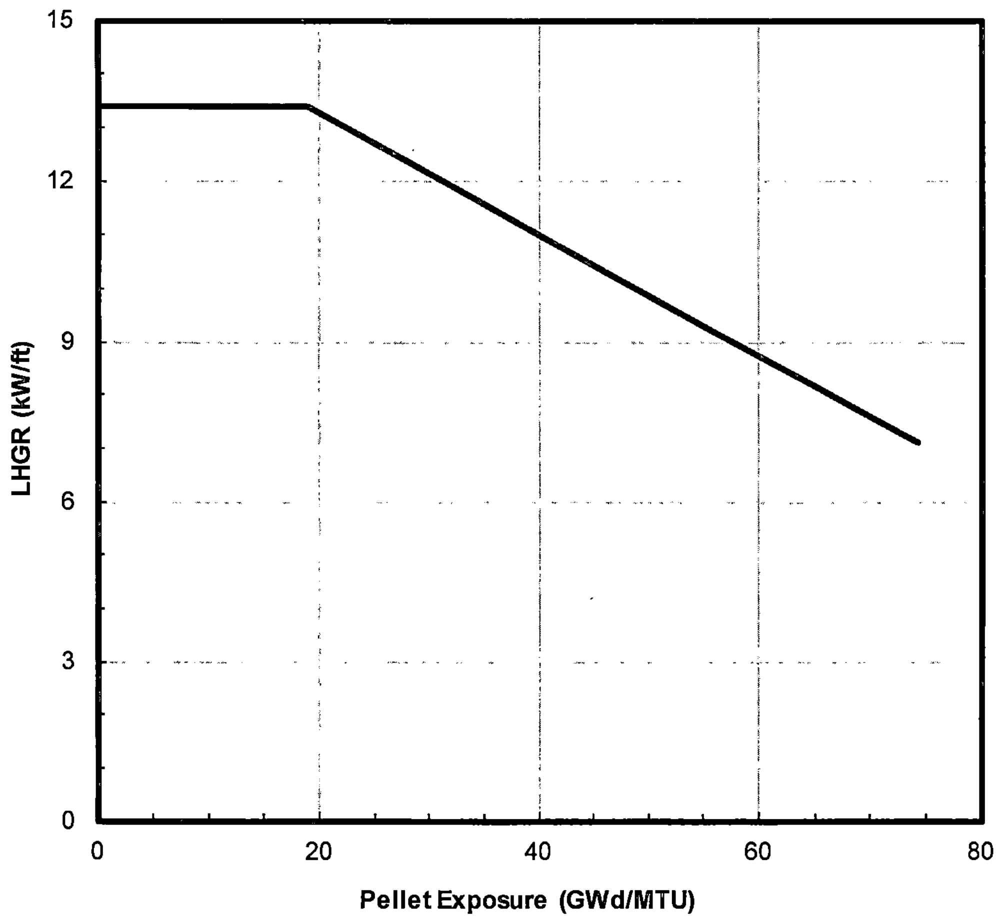
In-Service	All equipment In-Service
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
PLUOOS	Power Load Unbalance Out-Of-Service
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service or Final Feedwater Temperature Reduction
RCPOOS	One Recirculation Pump Out-Of-Service

Off-rated power corrections shown in Figure 3.3 and Figure 3.4 are dependent on operation of the Turbine Bypass Valve system. For this reason, separate limits are to be applied for TBVIS or TBVOOS operation. The limits have no dependency on RPTOOS, PLUOOS, FHOOS/FFWTR, or SLO.

Off-rated flow corrections shown in Figure 3.5 and Figure 3.6 are bounding for all EOOS conditions.

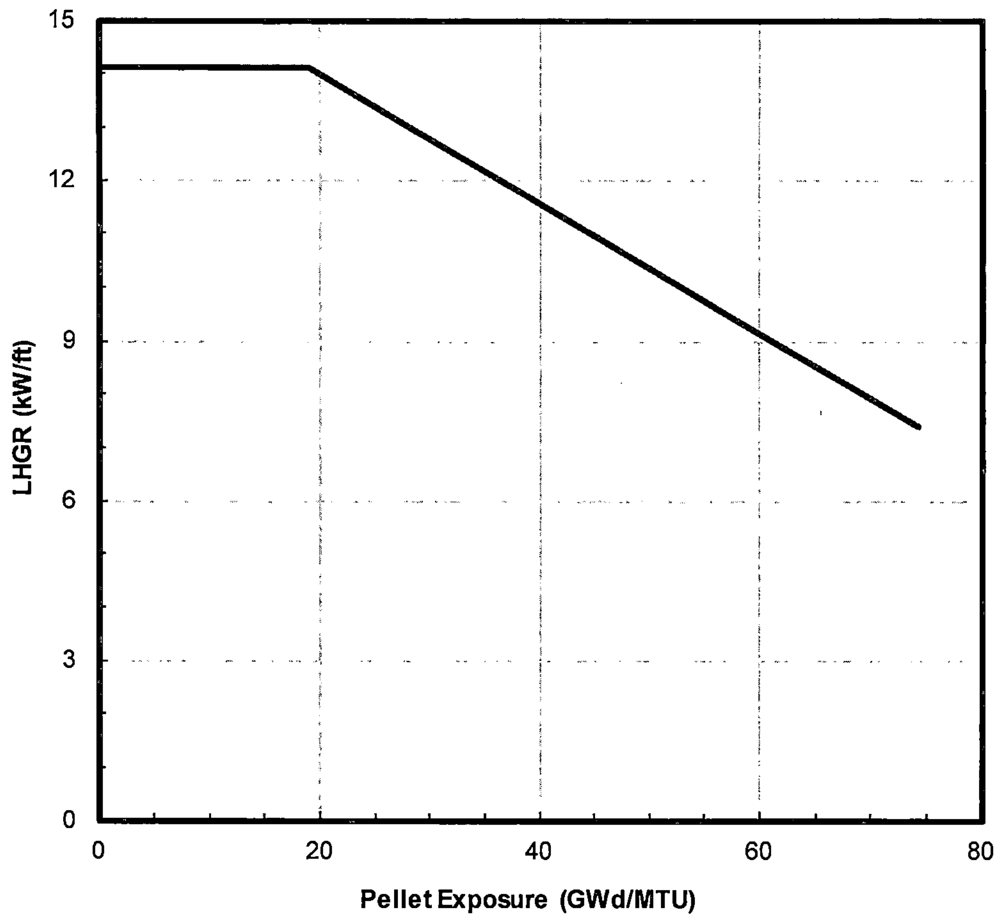
Off-rated power corrections shown in Figure 3.7 through Figure 3.10 are also dependent on operation of the Turbine Bypass Valve system. In this case, limits support FHOOS operation during startup. These limits have no dependency on RPTOOS, PLUOOS, or SLO.

* All equipment service conditions assume 1 SRVOOS.



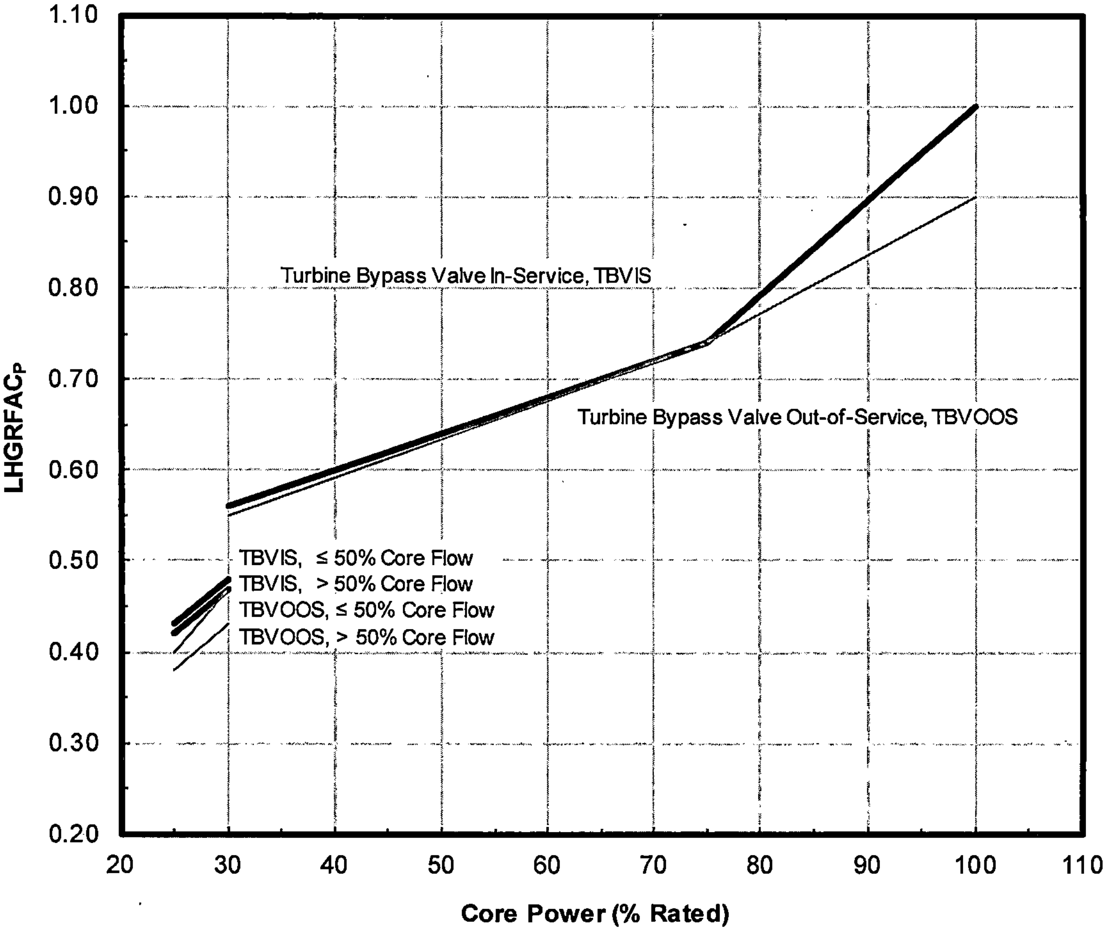
Pellet Exposure (GWd/MTU)	LHGR Limit (kW/ft)
0.0	13.4
18.9	13.4
74.4	7.1

Figure 3.1 $LHGR_{RATED}$ for ATRIUM-10 Fuel



Pellet Exposure (GWd/MTU)	LHGR Limit (kW/ft)
0.0	14.1
18.9	14.1
74.4	7.4

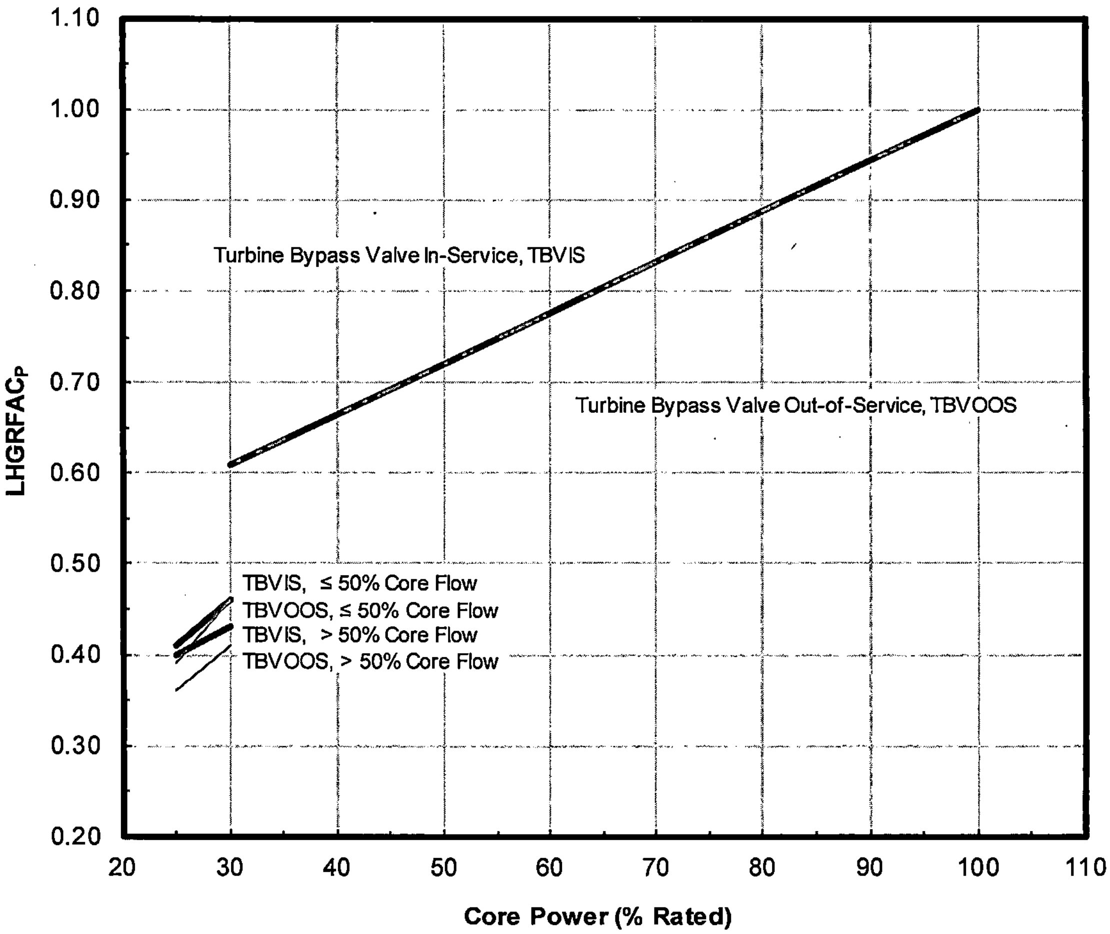
Figure 3.2 $LHGR_{RATED}$ for ATRIUM-10XM Fuel



Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$	*Turbine Bypass Out-of-Service* Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00	100.0	0.90
75.0	0.74	75.0	0.74
30.0	0.56	30.0	0.55
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.47	30.0	0.43
25.0	0.42	25.0	0.38
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.48	30.0	0.47
25.0	0.43	25.0	0.40

Figure 3.3 Base Operation LHGRFAC$_P$ for ATRIUM-10 Fuel

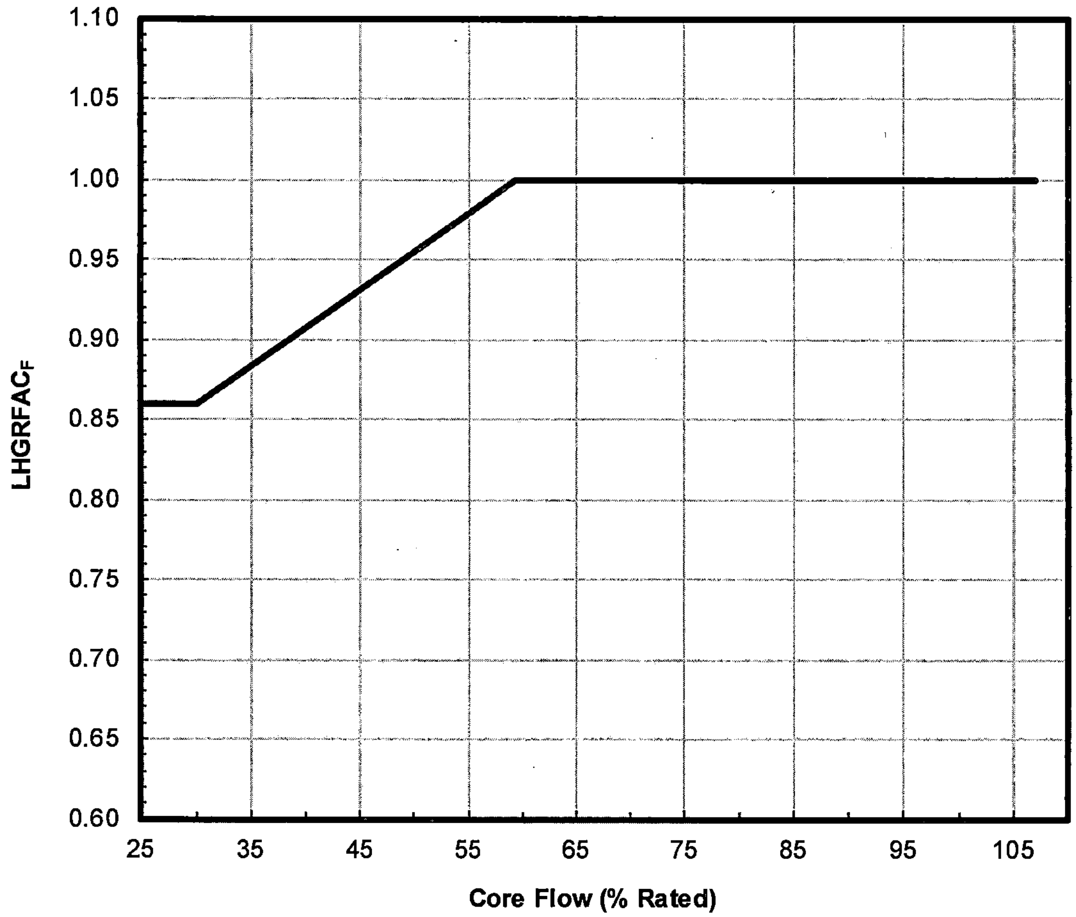
(*Independent of other EOOS conditions*)



Turbine Bypass In-Service Core Power	LHGRFAC$_P$	*Turbine Bypass Out-of-Service* Core Power	LHGRFAC$_P$
(% Rated)		(% Rated)	
100.0	1.00	100.0	1.00
30.0	0.61	30.0	0.61
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.43	30.0	0.41
25.0	0.40	25.0	0.36
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.46	30.0	0.46
25.0	0.41	25.0	0.39

Figure 3.4 Base Operation LHGRFAC$_P$ for ATRIUM-10XM Fuel

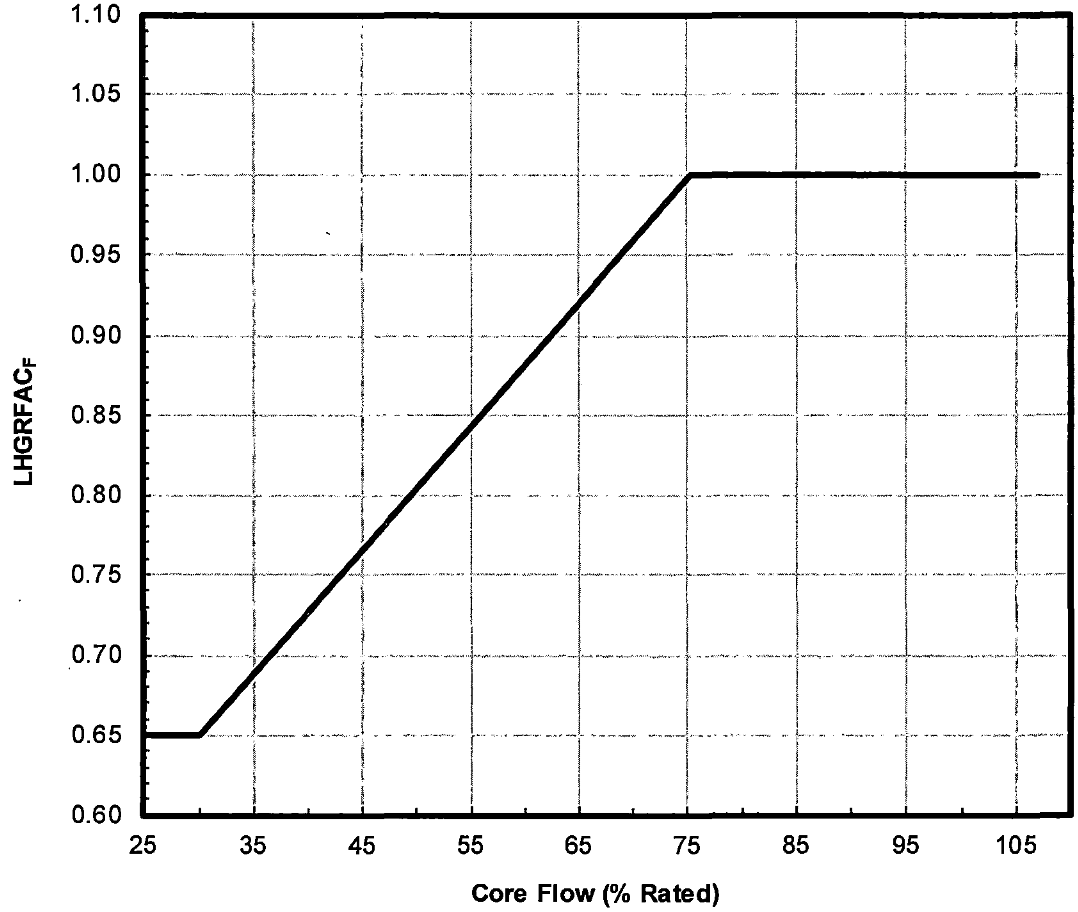
(*Independent of other EOOS conditions*)



Core Flow (% Rated)	$LHGRFAC_F$
0.0	0.86
30.0	0.86
59.3	1.00
107.0	1.00

Figure 3.5 $LHGRFAC_F$ for ATRIUM-10 Fuel
(*Values bound all EOOS conditions*)

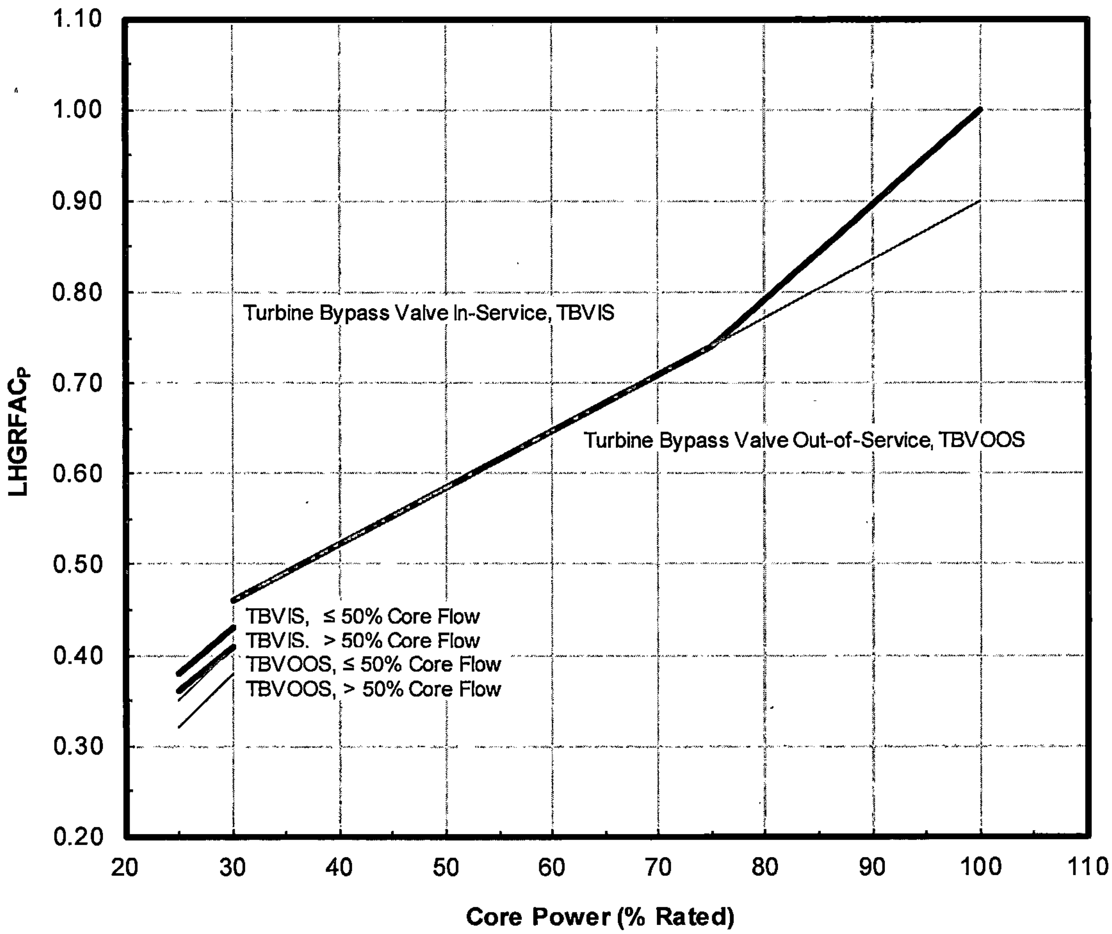
(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)



Core Flow (% Rated)	LHGRFAC$_F$
0.0	0.65
30.0	0.65
75.2	1
107.0	1

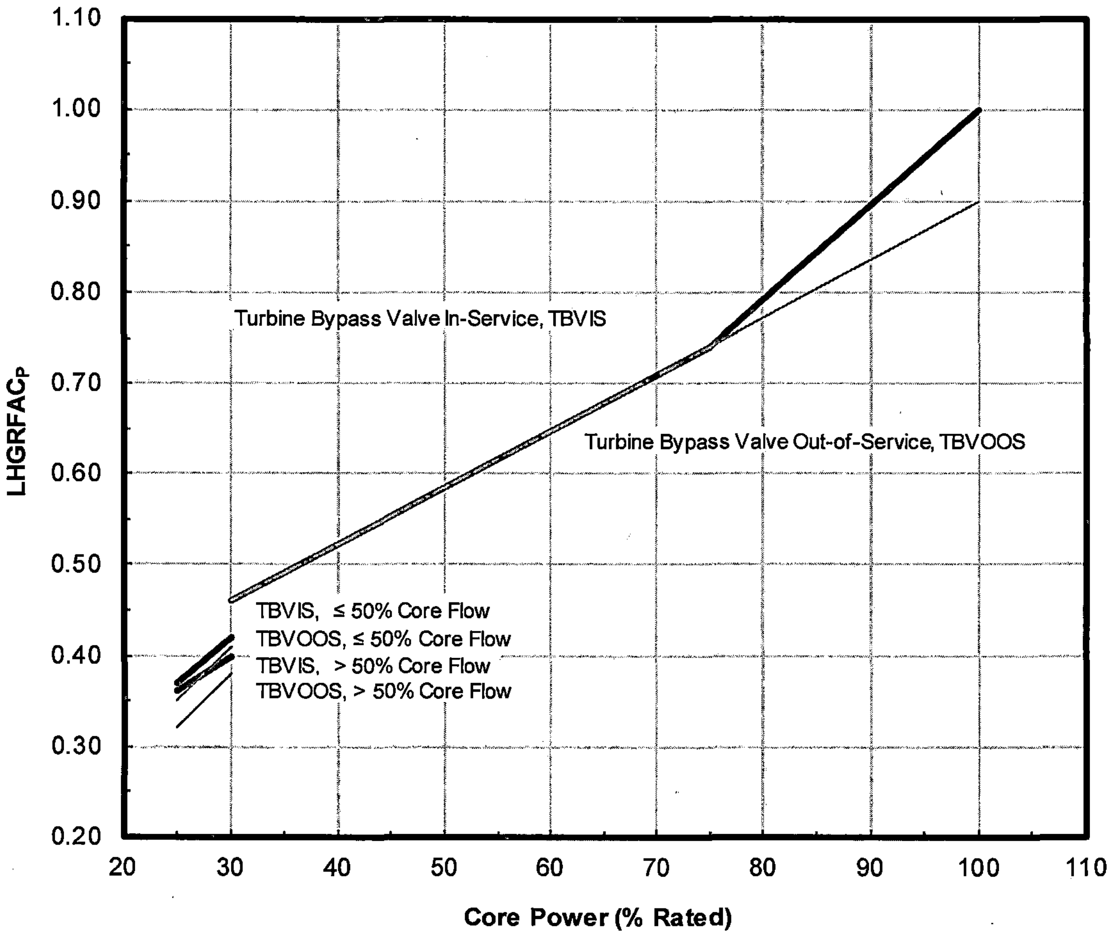
Figure 3.6 LHGRFAC$_F$ for ATRIUM-10XM Fuel
(*Values bound all EOOS conditions*)

(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)



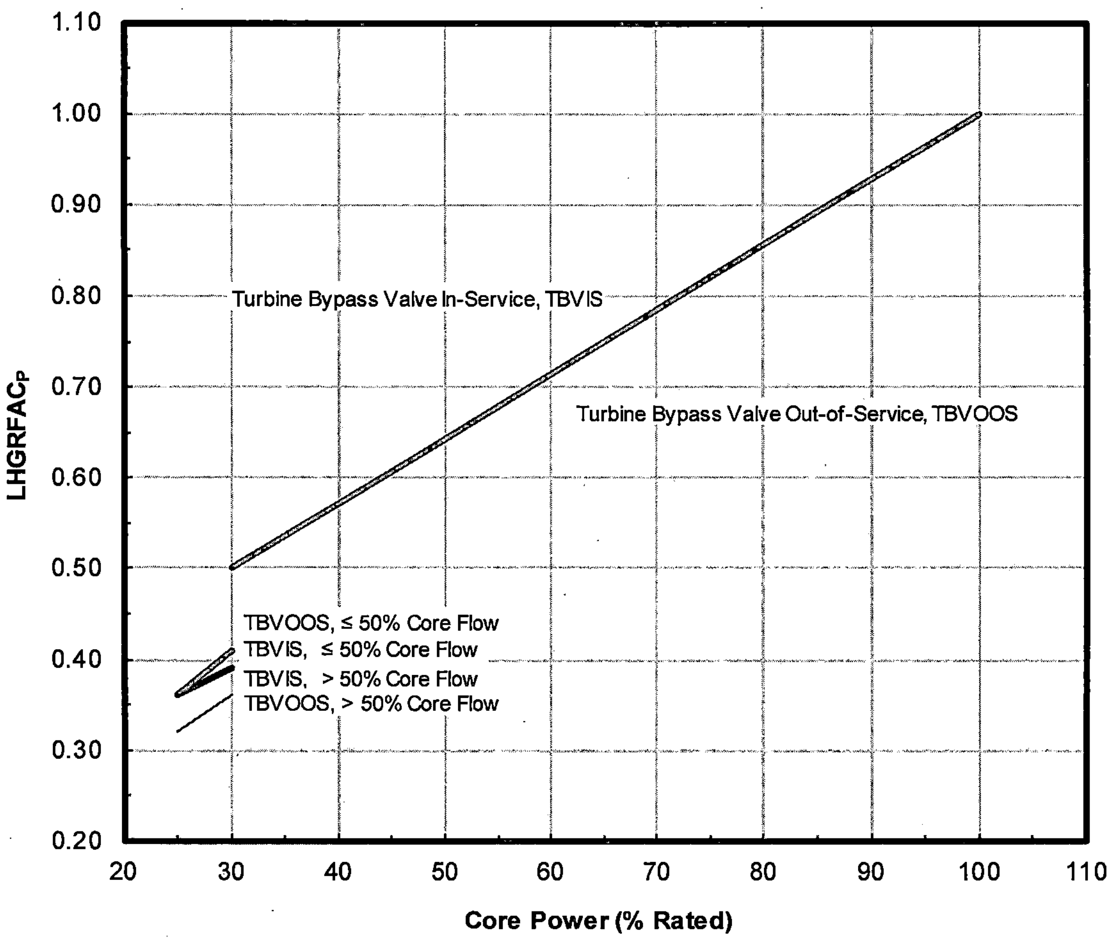
Turbine Bypass In-Service		*Turbine Bypass Out-of-Service*	
Core Power	**LHGRFAC$_P$**	**Core Power**	**LHGRFAC$_P$**
(% Rated)		**(% Rated)**	
100.0	1.00	100.0	0.90
75.0	0.74	75.0	0.74
30.0	0.46	30.0	0.46
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.41	30.0	0.38
25.0	0.36	25.0	0.32
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.43	30.0	0.41
25.0	0.38	25.0	0.35

Figure 3.7 Startup Operation LHGRFAC$_P$ for ATRIUM-10 Fuel:
Table 3.1 Temperature Range 1
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



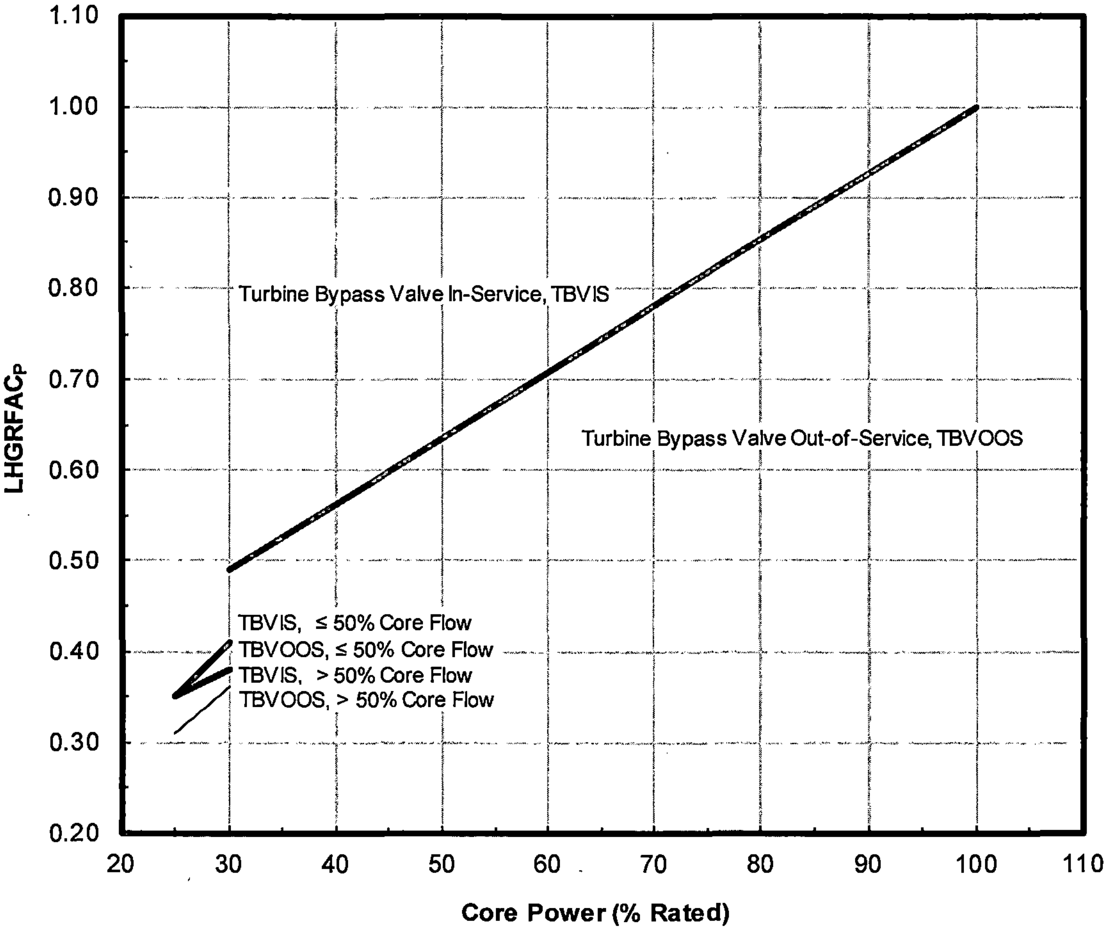
Turbine Bypass In-Service Core Power (%Rated)	LHGRFAC$_P$	*Turbine Bypass Out-of-Service* Core Power (%Rated)	LHGRFAC$_P$
100.0	1.00	100.0	0.90
75.0	0.74	75.0	0.74
30.0	0.46	30.0	0.46
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.40	30.0	0.38
25.0	0.36	25.0	0.32
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.42	30.0	0.41
25.0	0.37	25.0	0.35

Figure 3.8 Startup Operation LHGRFAC$_P$ for ATRIUM-10 Fuel:
Table 3.1 Temperature Range 2
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$	*Turbine Bypass Out-of-Service* Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00	100.0	1.00
30.0	0.50	30.0	0.50
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.39	30.0	0.36
25.0	0.36	25.0	0.32
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.41	30.0	0.41
25.0	0.36	25.0	0.36

Figure 3.9 Startup Operation LHGRFAC$_P$ for ATRIUM-10XM Fuel:
Table 3.1 Temperature Range 1
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



Turbine Bypass In-Service		*Turbine Bypass Out-of-Service*	
Core Power	**$LHGRFAC_P$**	**Core Power**	**$LHGRFAC_P$**
(% Rated)		**(% Rated)**	
100.0	1.00	100.0	1.00
30.0	0.49	30.0	0.49
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.38	30.0	0.36
25.0	0.35	25.0	0.31
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.41	30.0	0.41
25.0	0.35	25.0	0.35

Figure 3.10 Startup Operation $LHGRFAC_P$ for ATRIUM-10XM
Fuel: Table 3.1 Temperature Range 2
(no Feedwater heating during startup)
(Limits valid at and below 50% power)

4 OLMCPR Limits

(Technical Specification 3.2.2, 3.3.4.1, & 3.7.5)

OLMCPR is calculated to be the most limiting of the flow or power dependent values

$$\text{OLMCPR limit} = \text{MAX}\ (\ MCPR_F\ ,\ MCPR_P\ )$$

where:

$MCPR_F$	core flow-dependent MCPR limit
$MCPR_P$	power-dependent MCPR limit

4.1 Flow Dependent MCPR Limit: $MCPR_F$

$MCPR_F$ limits are dependent upon core flow (% of Rated), and the max core flow limit, (Rated or Increased Core Flow, ICF). $MCPR_F$ limits are shown in Figure 4.1, per Reference 1. Limits are valid for all EOOS combinations. No adjustment is required for SLO conditions.

4.2 Power Dependent MCPR Limit: $MCPR_P$

$MCPR_P$ limits are dependent upon:

- Core Power Level (% of Rated)
- Technical Specification Scram Speed (TSSS), Nominal Scram Speed (NSS), or Optimum Scram Speed (OSS)
- Cycle Operating Exposure (NEOC, EOC, and CD - as defined in this section)
- Equipment Out-Of-Service Options
- Two or Single recirculation Loop Operation (TLO vs. SLO)

The $MCPR_P$ limits are provided in Table 4.2 through Table 4.9, where each table contains the limits for all fuel types and EOOS options (for a specified scram speed and exposure range). The CMSS determines $MCPR_P$ limits, from these tables, based on linear interpolation between the specified powers.

4.2.1 <u>Startup without Feedwater Heaters</u>

There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. Additional power dependent limits are shown in Table 4.5 through Table 4.8 based on temperature conditions identified in Table 3.1.

4.2.2 Scram Speed Dependent Limits (TSSS vs. NSS vs. OSS)

$MCPR_P$ limits are provided for three different sets of assumed scram speeds. The Technical Specification Scram Speed (TSSS) $MCPR_P$ limits are applicable at all times, as long as the scram time surveillance demonstrates the times in Technical Specification Table 3.1.4-1 are met. Both Nominal Scram Speeds (NSS) and/or Optimum Scram Speeds (OSS) may be used, as long as the scram time surveillance demonstrates Table 4.1 times are applicable.*†

Table 4.1 Nominal Scram Time Basis

Notch Position (index)	Nominal Scram Timing (seconds)	Optimum Scram Timing (seconds)
46	0.420	0.380
36	0.980	0.875
26	1.600	1.465
6	2.900	2.900

In demonstrating compliance with the NSS and/or OSS scram time basis, surveillance requirements from Technical Specification 3.1.4 apply; accepting the definition of SLOW rods should conform to scram speeds shown in Table 4.1. If conformance is not demonstrated, TSSS based $MCPR_P$ limits are applied.

On initial cycle startup, TSSS limits are used until the successful completion of scram timing confirms NSS and/or OSS based limits are applicable.

4.2.3 Exposure Dependent Limits

Exposures are tracked on a Core Average Exposure basis (CAVEX, not Cycle Exposure). Higher exposure $MCPR_P$ limits are always more limiting and may be used for any Core Average Exposure up to the ending exposure. Per Reference 1, $MCPR_P$ limits are provided for the following exposure ranges:

BOC to NEOC NEOC corresponds to **28,825.7 MWd / MTU**

* Reference 1 analysis results are based on information identified in Reference 4.

† Drop out times consistent with method used to perform actual timing measurements (i.e., including pickup/dropout effects).

NEOC refers to a Near EOC exposure point. At this time, the NEOC exposure is the maximum value supported by Reference 1, as a consequence of CR 1145408. A future Revision to Reference 1 will be required in order to extend operation to the full design energy.

4.2.4 Equipment Out-Of-Service (EOOS) Options

EOOS options* covered by $MCPR_P$ limits are given by the following:

In-Service	All equipment In-Service
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
RPTOOS+TBVOOS	Combined RPTOOS and TBVOOS
PLUOOS	Power Load Unbalance Out-Of-Service
PLUOOS+RPTOOS	Combined PLUOOS and RPTOOS
PLUOOS+TBVOOS	Combined PLUOOS and TBVOOS
PLUOOS+TBVOOS+RPTOOS	Combined PLUOOS, RPTOOS, and TBVOOS
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service (or Final Feedwater Temperature Reduction)
RCPOOS	One Recirculation Pump Out-Of-Service

For exposure ranges up to NEOC additional combinations of $MCPR_P$ limits are also provided including FHOOS.

4.2.5 Single-Loop-Operation (SLO) Limits

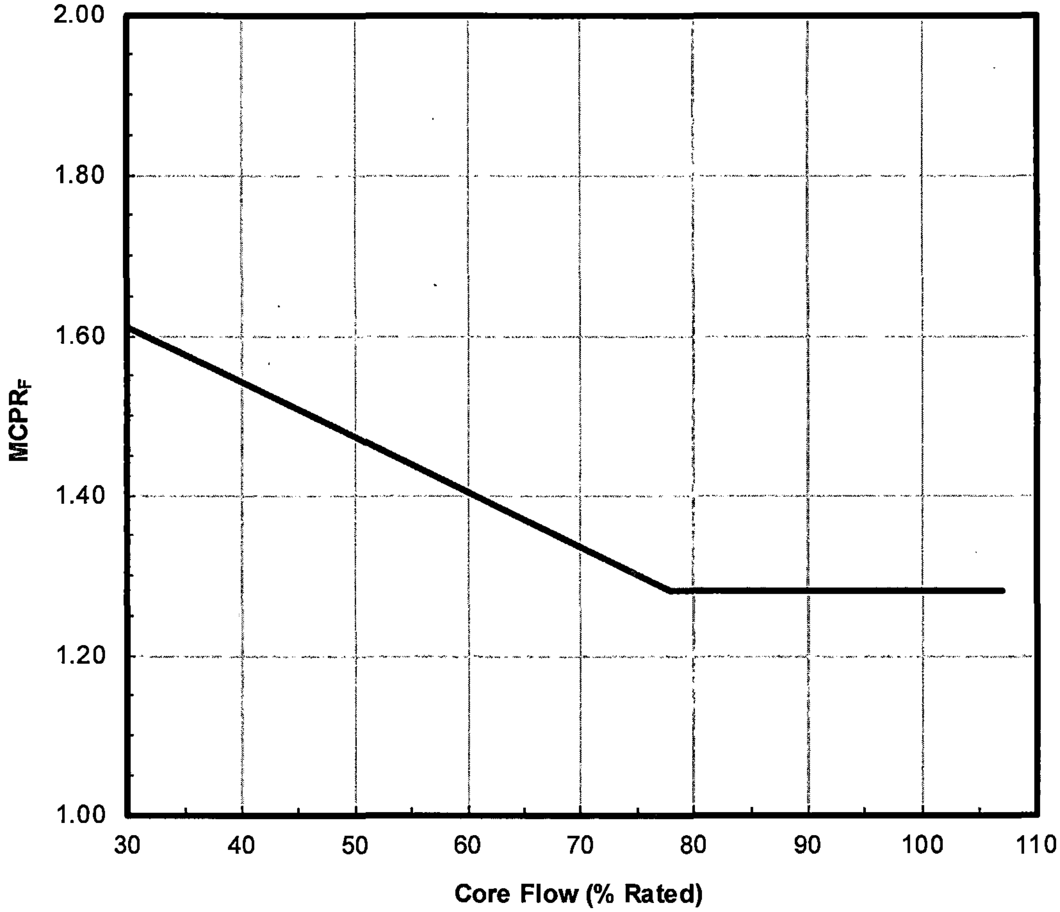
When operating in RCPOOS conditions, MCPRp limits are constructed differently from the normal operating RCP conditions. The limiting event for RCPOOS is a pump seizure scenario, which sets the upper bound for allowed core power and flow†. This event is not impacted by scram time assumptions. Specific $MCPR_P$ limits are shown in Table 4.9.

4.2.6 Below Pbypass Limits

Below Pbypass (30% rated power), $MCPR_P$ limits depend upon core flow. One set of $MCPR_P$ limits applies for core flow above 50% of rated; a second set applies if the core flow is less than or equal to 50% rated.

* All equipment service conditions assume 1 SRVOOS.

† RCPOOS limits are only valid up to 50% rated core power, 50% rated core flow, and an active recirculation drive flow of 17.73 Mlb_m/hr.



Core Flow (% Rated)	$MCPR_F$
30.0	1.61
78.0	1.28
107.0	1.28

Figure 4.1 $MCPR_F$ for All Fuel Types
(*Values bound all EOOS conditions*)

(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)

Table 4.2 MCPR$_P$ Limits for All Fuel Types: Optimum Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
Base Case	100	1.47	---	---	1.41	---	---
	75	1.61	---	---	1.53	---	---
	65	1.69	---	---	1.61	---	---
	50	1.86	---	---	1.75	---	---
	50	1.96	---	---	1.81	---	---
	40	2.12	---	---	2.00	---	---
	30	2.48	---	---	2.31	---	---
	30 at > 50%F	2.64	---	---	2.64	---	---
	25 at > 50%F	2.93	---	---	2.73	---	---
	30 at ≤ 50%F	2.57	---	---	2.37	---	---
	25 at ≤ 50%F	2.82	---	---	2.61	---	---
FHOOS	100	1.50	---	---	1.43	---	---
	75	1.65	---	---	1.58	---	---
	65	1.75	---	---	1.65	---	---
	50	1.95	---	---	---	---	---
	50	1.97	---	---	1.82	---	---
	40	2.24	---	---	2.10	---	---
	30	2.63	---	---	2.45	---	---
	30 at > 50%F	2.77	---	---	2.79	---	---
	25 at > 50%F	3.12	---	---	2.86	---	---
	30 at ≤ 50%F	2.74	---	---	2.47	---	---
	25 at ≤ 50%F	3.00	---	---	2.74	---	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR/FHOOS is supported for the BOC to End of Coast limits.

Table 4.3 MCPR$_P$ Limits for All Fuel Types: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
Base Case	100	1.49	---	---	1.42	---	---
	75	1.63	---	---	1.55	---	---
	65	1.71	---	---	1.62	---	---
	50	1.91	---	---	1.77	---	---
	50	1.97	---	---	1.81	---	---
	40	2.17	---	---	2.03	---	---
	30	2.54	---	---	2.34	---	---
	30 at > 50%F	2.64	---	---	2.64	---	---
	25 at > 50%F	2.93	---	---	2.73	---	---
	30 at ≤ 50%F	2.57	---	---	2.37	---	---
	25 at ≤ 50%F	2.82	---	---	2.61	---	---
TBVOOS	100	1.55	---	---	1.46	---	---
	75	1.68	---	---	1.58	---	---
	65	1.76	---	---	1.66	---	---
	50	1.92	---	---	1.80	---	---
	50	1.97	---	---	1.81	---	---
	40	2.17	---	---	2.03	---	---
	30	2.54	---	---	2.34	---	---
	30 at > 50%F	3.18	---	---	3.12	---	---
	25 at > 50%F	3.59	---	---	3.36	---	---
	30 at ≤ 50%F	2.89	---	---	2.64	---	---
	25 at ≤ 50%F	3.33	---	---	3.06	---	---
FHOOS	100	1.52	---	---	1.45	---	---
	75	1.67	---	---	1.58	---	---
	65	1.76	---	---	1.66	---	---
	50	---	---	---	---	---	---
	50	2.00	---	---	1.85	---	---
	40	2.29	---	---	2.13	---	---
	30	2.68	---	---	2.48	---	---
	30 at > 50%F	2.77	---	---	2.79	---	---
	25 at > 50%F	3.12	---	---	2.86	---	---
	30 at ≤ 50%F	2.74	---	---	2.48	---	---
	25 at ≤ 50%F	3.00	---	---	2.74	---	---
PLUOOS	100	1.49	---	---	1.42	---	---
	75	1.63	---	---	1.55	---	---
	65	1.86	---	---	1.72	---	---
	50	---	---	---	---	---	---
	50	1.97	---	---	1.81	---	---
	40	2.17	---	---	2.03	---	---
	30	2.54	---	---	2.34	---	---
	30 at > 50%F	2.64	---	---	2.64	---	---
	25 at > 50%F	2.93	---	---	2.73	---	---
	30 at ≤ 50%F	2.57	---	---	2.37	---	---
	25 at ≤ 50%F	2.82	---	---	2.61	---	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.3 MCPRP Limits for All Fuel Types: Nominal Scram Time Basis (*continued*)*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
TBVOOS FHOOS	100	1.57	---	---	1.48	---	---
	75	1.72	---	---	1.61	---	---
	65	1.82	---	---	1.69	---	---
	50	---	---	---	---	---	---
	50	2.01	---	---	1.86	---	---
	40	2.29	---	---	2.13	---	---
	30	2.68	---	---	2.48	---	---
	30 at > 50%F	3.29	---	---	3.26	---	---
	25 at > 50%F	3.72	---	---	3.48	---	---
	30 at ≤ 50%F	3.02	---	---	2.77	---	---
	25 at ≤ 50%F	3.49	---	---	3.22	---	---
TBVOOS PLUOOS	100	1.55	---	---	1.46	---	---
	75	1.68	---	---	1.58	---	---
	65	1.86	---	---	1.72	---	---
	50	---	---	---	---	---	---
	50	1.97	---	---	1.81	---	---
	40	2.17	---	---	2.03	---	---
	30	2.54	---	---	2.34	---	---
	30 at > 50%F	3.18	---	---	3.12	---	---
	25 at > 50%F	3.59	---	---	3.36	---	---
	30 at ≤ 50%F	2.89	---	---	2.64	---	---
	25 at ≤ 50%F	3.33	---	---	3.06	---	---
FHOOS PLUOOS	100	1.52	---	---	1.45	---	---
	75	1.67	---	---	1.58	---	---
	65	1.86	---	---	1.72	---	---
	50	---	---	---	---	---	---
	50	2.00	---	---	1.85	---	---
	40	2.29	---	---	2.13	---	---
	30	2.68	---	---	2.48	---	---
	30 at > 50%F	2.77	---	---	2.79	---	---
	25 at > 50%F	3.12	---	---	2.86	---	---
	30 at ≤ 50%F	2.74	---	---	2.48	---	---
	25 at ≤ 50%F	3.00	---	---	2.74	---	---
TBVOOS FHOOS PLUOOS	100	1.57	---	---	1.48	---	---
	75	1.72	---	---	1.61	---	---
	65	1.86	---	---	1.72	---	---
	50	---	---	---	---	---	---
	50	2.01	---	---	1.86	---	---
	40	2.29	---	---	2.13	---	---
	30	2.68	---	---	2.48	---	---
	30 at > 50%F	3.29	---	---	3.26	---	---
	25 at > 50%F	3.72	---	---	3.48	---	---
	30 at ≤ 50%F	3.02	---	---	2.77	---	---
	25 at ≤ 50%F	3.49	---	---	3.22	---	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.4 MCPRP Limits for All Fuel Types: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
Base Case	100	1.51	---	---	1.44	---	---
	75	1.64	---	---	1.56	---	---
	65	1.73	---	---	1.63	---	---
	50	1.95	---	---	1.80	---	---
	50	1.98	---	---	1.82	---	---
	40	2.21	---	---	2.06	---	---
	30	2.59	---	---	2.37	---	---
	30 at > 50%F	2.64	---	---	2.64	---	---
	25 at > 50%F	2.93	---	---	2.73	---	---
	30 at ≤ 50%F	2.59	---	---	2.37	---	---
	25 at ≤ 50%F	2.82	---	---	2.61	---	---
TBVOOS	100	1.58	---	---	1.47	---	---
	75	1.70	---	---	1.60	---	---
	65	1.77	---	---	1.67	---	---
	50	1.97	---	---	1.81	---	---
	50	1.98	---	---	1.82	---	---
	40	2.21	---	---	2.06	---	---
	30	2.59	---	---	2.37	---	---
	30 at > 50%F	3.18	---	---	3.12	---	---
	25 at > 50%F	3.59	---	---	3.36	---	---
	30 at ≤ 50%F	2.89	---	---	2.64	---	---
	25 at ≤ 50%F	3.33	---	---	3.06	---	---
FHOOS	100	1.54	---	---	1.46	---	---
	75	1.68	---	---	1.59	---	---
	65	1.80	---	---	1.68	---	---
	50	---	---	---	---	---	---
	50	2.05	---	---	1.89	---	---
	40	2.33	---	---	2.16	---	---
	30	2.74	---	---	2.52	---	---
	30 at > 50%F	2.77	---	---	2.79	---	---
	25 at > 50%F	3.12	---	---	2.86	---	---
	30 at ≤ 50%F	2.74	---	---	2.52	---	---
	25 at ≤ 50%F	3.00	---	---	2.74	---	---
PLUOOS	100	1.51	---	---	1.44	---	---
	75	1.64	---	---	1.56	---	---
	65	1.88	---	---	1.73	---	---
	50	---	---	---	---	---	---
	50	1.98	---	---	1.82	---	---
	40	2.21	---	---	2.06	---	---
	30	2.59	---	---	2.37	---	---
	30 at > 50%F	2.64	---	---	2.64	---	---
	25 at > 50%F	2.93	---	---	2.73	---	---
	30 at ≤ 50%F	2.59	---	---	2.37	---	---
	25 at ≤ 50%F	2.82	---	---	2.61	---	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.4 MCPR$_P$ Limits for All Fuel Types: Technical Specification Scram Time Basis (*continued*)*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
TBVOOS FHOOS	100	1.59	---	---	1.49	---	---
	75	1.73	---	---	1.64	---	---
	65	1.83	---	---	1.71	---	---
	50	---	---	---	---	---	---
	50	2.06	---	---	1.89	---	---
	40	2.33	---	---	2.16	---	---
	30	2.74	---	---	2.52	---	---
	30 at > 50%F	3.29	---	---	3.26	---	---
	25 at > 50%F	3.72	---	---	3.48	---	---
	30 at ≤ 50%F	3.02	---	---	2.77	---	---
	25 at ≤ 50%F	3.49	---	---	3.22	---	---
TBVOOS PLUOOS	100	1.58	---	---	1.47	---	---
	75	1.70	---	---	1.60	---	---
	65	1.88	---	---	1.73	---	---
	50	---	---	---	---	---	---
	50	1.98	---	---	1.82	---	---
	40	2.21	---	---	2.06	---	---
	30	2.59	---	---	2.37	---	---
	30 at > 50%F	3.18	---	---	3.12	---	---
	25 at > 50%F	3.59	---	---	3.36	---	---
	30 at ≤ 50%F	2.89	---	---	2.64	---	---
	25 at ≤ 50%F	3.33	---	---	3.06	---	---
FHOOS PLUOOS	100	1.54	---	---	1.46	---	---
	75	1.68	---	---	1.59	---	---
	65	1.88	---	---	1.73	---	---
	50	---	---	---	---	---	---
	50	2.05	---	---	1.89	---	---
	40	2.33	---	---	2.16	---	---
	30	2.74	---	---	2.52	---	---
	30 at > 50%F	2.77	---	---	2.79	---	---
	25 at > 50%F	3.12	---	---	2.86	---	---
	30 at ≤ 50%F	2.74	---	---	2.52	---	---
	25 at ≤ 50%F	3.00	---	---	2.74	---	---
TBVOOS FHOOS PLUOOS	100	1.59	---	---	1.49	---	---
	75	1.73	---	---	1.64	---	---
	65	1.88	---	---	1.73	---	---
	50	---	---	---	---	---	---
	50	2.06	---	---	1.89	---	---
	40	2.33	---	---	2.16	---	---
	30	2.74	---	---	2.52	---	---
	30 at > 50%F	3.29	---	---	3.26	---	---
	25 at > 50%F	3.72	---	---	3.48	---	---
	30 at ≤ 50%F	3.02	---	---	2.77	---	---
	25 at ≤ 50%F	3.49	---	---	3.22	---	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.5 Startup Operation MCPR$_P$ Limits for Table 3.1 Temperature Range 1 for All Fuel Types: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
TBVIS	100	1.52	---	---	1.45	---	---
	75	1.67	---	---	1.58	---	---
	65	1.86	---	---	1.72	---	---
	50	---	---	---	---	---	---
	50	2.22	---	---	2.06	---	---
	40	2.57	---	---	2.40	---	---
	30	3.04	---	---	2.83	---	---
	30 at > 50%F	3.11	---	---	3.03	---	---
	25 at > 50%F	3.50	---	---	3.15	---	---
	30 at ≤ 50%F	3.11	---	---	2.83	---	---
	25 at ≤ 50%F	3.48	---	---	3.04	---	---
TBVOOS	100	1.57	---	---	1.48	---	---
	75	1.72	---	---	1.61	---	---
	65	1.86	---	---	1.72	---	---
	50	---	---	---	---	---	---
	50	2.22	---	---	2.06	---	---
	40	2.57	---	---	2.40	---	---
	30	3.04	---	---	2.83	---	---
	30 at > 50%F	3.50	---	---	3.49	---	---
	25 at > 50%F	4.02	---	---	3.73	---	---
	30 at ≤ 50%F	3.31	---	---	2.99	---	---
	25 at ≤ 50%F	3.85	---	---	3.49	---	---

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.6 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 2 for All Fuel Types: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
TBVIS	100	1.52	---	---	1.45	---	---
	75	1.67	---	---	1.58	---	---
	65	1.86	---	---	1.72	---	---
	50	---	---	---	---	---	---
	50	2.23	---	---	2.07	---	---
	40	2.58	---	---	2.42	---	---
	30	3.06	---	---	2.86	---	---
	30 at > 50%F	3.14	---	---	3.04	---	---
	25 at > 50%F	3.52	---	---	3.17	---	---
	30 at ≤ 50%F	3.14	---	---	2.86	---	---
	25 at ≤ 50%F	3.50	---	---	3.06	---	---
TBVOOS	100	1.57	---	---	1.48	---	---
	75	1.72	---	---	1.61	---	---
	65	1.86	---	---	1.72	---	---
	50	---	---	---	---	---	---
	50	2.23	---	---	2.07	---	---
	40	2.58	---	---	2.42	---	---
	30	3.06	---	---	2.86	---	---
	30 at > 50%F	3.52	---	---	3.51	---	---
	25 at > 50%F	4.03	---	---	3.75	---	---
	30 at ≤ 50%F	3.36	---	---	3.01	---	---
	25 at ≤ 50%F	3.91	---	---	3.50	---	---

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.7 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 1 for All Fuel Types: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
TBVIS	100	1.54	---	---	1.46	---	---
	75	1.68	---	---	1.59	---	---
	65	1.88	---	---	1.73	---	---
	50	---	---	---	---	---	---
	50	2.27	---	---	2.10	---	---
	40	2.62	---	---	2.44	---	---
	30	3.11	---	---	2.87	---	---
	30 at > 50%F	3.11	---	---	3.03	---	---
	25 at > 50%F	3.50	---	---	3.15	---	---
	30 at ≤ 50%F	3.11	---	---	2.87	---	---
	25 at ≤ 50%F	3.48	---	---	3.04	---	---
TBVOOS	100	1.59	---	---	1.49	---	---
	75	1.73	---	---	1.64	---	---
	65	1.88	---	---	1.73	---	---
	50	---	---	---	---	---	---
	50	2.27	---	---	2.10	---	---
	40	2.62	---	---	2.44	---	---
	30	3.11	---	---	2.87	---	---
	30 at > 50%F	3.50	---	---	3.49	---	---
	25 at > 50%F	4.02	---	---	3.73	---	---
	30 at ≤ 50%F	3.31	---	---	2.99	---	---
	25 at ≤ 50%F	3.85	---	---	3.49	---	---

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.8 Startup Operation MCPRP Limits for Table 3.1 Temperature Range 2 for All Fuel Types: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
TBVIS	100	1.54	---	---	1.46	---	---
	75	1.68	---	---	1.59	---	---
	65	1.88	---	---	1.73	---	---
	50	---	---	---	---	---	---
	50	2.28	---	---	2.11	---	---
	40	2.64	---	---	2.46	---	---
	30	3.13	---	---	2.90	---	---
	30 at > 50%F	3.14	---	---	3.04	---	---
	25 at > 50%F	3.52	---	---	3.17	---	---
	30 at ≤ 50%F	3.14	---	---	2.90	---	---
	25 at ≤ 50%F	3.50	---	---	3.06	---	---
TBVOOS	100	1.59	---	---	1.49	---	---
	75	1.73	---	---	1.64	---	---
	65	1.88	---	---	1.73	---	---
	50	---	---	---	---	---	---
	50	2.28	---	---	2.11	---	---
	40	2.64	---	---	2.46	---	---
	30	3.13	---	---	2.90	---	---
	30 at > 50%F	3.52	---	---	3.51	---	---
	25 at > 50%F	4.03	---	---	3.75	---	---
	30 at ≤ 50%F	3.36	---	---	3.01	---	---
	25 at ≤ 50%F	3.91	---	---	3.50	---	---

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.9 MCPR$_P$ Limits for All Fuel Types: Single Loop Operation for All Scram Times*

Operating Condition	Power (% of rated)	BOC to NEOC	
		ATRIUM-10	ATRIUM-10XM
RCPOOS FHOOS	100	2.07	2.04
	50	2.07	2.04
	40	2.35	2.18
	30	2.76	2.54
	30 at > 50%F	2.79	2.81
	25 at > 50%F	3.14	2.88
	30 at ≤ 50%F	2.76	2.54
	25 at ≤ 50%F	3.02	2.76
RCPOOS TBVOOS PLUOOS FHOOS	100	2.08	2.04
	50	2.08	2.04
	40	2.35	2.18
	30	2.76	2.54
	30 at > 50%F	3.31	3.28
	25 at > 50%F	3.74	3.50
	30 at ≤ 50%F	3.04	2.79
	25 at ≤ 50%F	3.51	3.24
RCPOOS TBVOOS FHOOS1	100	2.29	2.12
	50	2.29	2.12
	40	2.64	2.46
	30	3.13	2.89
	30 at > 50%F	3.52	3.51
	25 at > 50%F	4.04	3.75
	30 at ≤ 50%F	3.33	3.01
	25 at ≤ 50%F	3.87	3.51
RCPOOS TBVOOS FHOOS2	100	2.30	2.13
	50	2.30	2.13
	40	2.66	2.48
	30	3.15	2.92
	30 at > 50%F	3.54	3.53
	25 at > 50%F	4.05	3.77
	30 at ≤ 50%F	3.38	3.03
	25 at ≤ 50%F	3.93	3.52

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop.

RCPOOS limits are only valid up to 50% rated core power, 50% rated core flow, and an active recirculation drive flow of 17.73 Mlbm/hr.

5 Oscillation Power Range Monitor (OPRM) Setpoint

(Technical Specification 3.3.1.1)

Technical Specification Table 3.3.1.1-1, Function 2f, identifies the OPRM upscale function.

Instrument setpoints are established, such that the reactor will be tripped before an oscillation can grow to the point where the SLMCPR is exceeded. An Option III stability analysis is performed for each reload core to determine allowable OLMCPR's as a function of OPRM setpoint. Analyses consider both steady state startup operation, and the case of a two recirculation pump trip from rated power.

The resulting stability based OLMCPR's are reported in Reference 1. The OPRM setpoint (*sometimes referred to as the Amplitude Trip, S_p*) is selected, such that required margin to the SLMCPR is provided without stability being a limiting event. Analyses are based on cycle specific DIVOM analyses performed per Reference 22. The calculated OLMCPR's are shown in Table 5.1. Review of results shown in COLR Table 4.2 indicates an OPRM setpoint of **1.14** may be used. The successive confirmation count (*sometimes referred to as N_p*) is provided in Table 5.2, per Reference 30.

Table 5.1 OPRM Setpoint Range*

OPRM Setpoint	OLMCPR (SS)	OLMCPR (2PT)
1.05	1.19	1.16
1.06	1.20	1.17
1.07	1.20	1.18
1.08	1.21	1.19
1.09	1.22	1.20
1.10	1.24	1.22
1.11	1.26	1.24
1.12	1.28	1.26
1.13	1.30	1.28
1.14	1.33	1.30
1.15	1.35	1.32

Table 5.2 OPRM Successive Confirmation Count Setpoint

Count	OPRM Setpoint
6	≥ 1.04
8	≥ 1.05
10	≥ 1.07
12	≥ 1.09
14	≥ 1.11
16	≥ 1.14
18	≥ 1.18
20	≥ 1.24

* Extrapolation beyond a setpoint of 1.15 is not allowed

6 APRM Flow Biased Rod Block Trip Settings

(Technical Requirements Manual Section 5.3.1 and Table 3.3.4-1)

The APRM rod block trip setting is based upon References 26 & 27, and is defined by the following:

SRB $\leq$ (0.66(W-ΔW) + 61%)	Allowable Value
SRB $\leq$ (0.66(W-ΔW) + 59%)	Nominal Trip Setpoint (NTSP)

where:

SRB	=	Rod Block setting in percent of rated thermal power (3458 MW_t)
W	=	Loop recirculation flow rate in percent of rated
ΔW	=	Difference between two-loop and single-loop effective recirculation flow at the same core flow (ΔW=0.0 for two-loop operation)

The APRM rod block trip setting is clamped at a maximum allowable value of 115% (corresponding to a NTSP of 113%).

7 Rod Block Monitor (RBM) Trip Setpoints and Operability

(Technical Specification Table 3.3.2.1-1)

The RBM trip setpoints and applicable power ranges, based on References 26 & 27, are shown in Table 7.1. Setpoints are based on an HTSP, unfiltered analytical limit of 114%. Unfiltered setpoints are consistent with a nominal RBM filter setting of 0.0 seconds; filtered setpoints are consistent with a nominal RBM filter setting less than 0.5 seconds. Cycle specific CRWE analyses of OLMCPR are documented in Reference 1, superseding values reported in References 26, 27, and 29.

Table 7.1 Analytical RBM Trip Setpoints*

RBM Trip Setpoint	Allowable Value (AV)	Nominal Trip Setpoint (NTSP)
LPSP	27%	25%
IPSP	62%	60%
HPSP	82%	80%
LTSP - unfiltered - filtered	121.7% 120.7%	120.0% 119.0%
ITSP - unfiltered - filtered	116.7% 115.7%	115.0% 114.0%
HTSP - unfiltered - filtered	111.7% 110.9%	110.0% 109.2%
DTSP	90%	92%

As a result of cycle specific CRWE analyses, RBM setpoints in Technical Specification Table 3.3.2.1-1 are applicable as shown in Table 7.2. Cycle specific setpoint analysis results are shown in Table 7.3, per Reference 1.

Table 7.2 RBM Setpoint Applicability

Thermal Power (% Rated)	Applicable MCPR†	Notes from Table 3.3.2.1-1	Comment
> 27% and < 90%	< 1.74	(a), (b), (f), (h)	two loop operation
	< 1.78	(a), (b), (f), (h)	single loop operation
≥ 90%	< 1.41	(g)	two loop operation‡

* Values are considered maximums. Using lower values, due to RBM system hardware/software limitations, is conservative, and acceptable.

† MCPR values shown correspond with, (support), SLMPCR values identified in Reference 1.

‡ Greater than 90% rated power is not attainable in single loop operation.

Table 7.3 Control Rod Withdrawal Error Results

RBM HTSP Analytical Limit	CRWE OLMCPR
Unfiltered	
107	1.30
111	1.33
114	1.36
117	1.38

Results, compared against the base case OLMCPR results of Table 4.2, indicate SLMCPR remains protected for RBM inoperable conditions (i.e., 114% unblocked).

8 Shutdown Margin Limit

(Technical Specification 3.1.1)

Assuming the strongest OPERABLE control blade is fully withdrawn, and all other OPERABLE control blades are fully inserted, the core shall be sub-critical and meet the following minimum shutdown margin:

SDM > 0.38% dk/k

EDMS L32 160607 800
QA Document
BFE-4008, Revision 2

TVA NPG

Reactor Engineering and Fuels - BWRFE
1101 Market Street, Chattanooga, TN 37402

Browns Ferry Unit 3 Cycle 18

Core Operating Limits Report, (105% OLTP)

TVA-COLR-BF3C18 Revision 2 (Final)
(Revision Log, Page v)

June 2016

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Reactor Engineering and Fuels - BWRFE
1101 Market Street, Chattanooga TN 37402

Date: June 7, 2016

Revision Log

Number	Page	Description
0-R2	All	This is a major revision supporting all modes of operation. Cross reference to CR 1145408-012. Full cycle exposure operation is supported.
1-R2	viii	Updated References 1 and 4.
2-R2	11-18	Section 3: Updated $LHGRFAC_p$ Figures 3.3, 3.4, 3.7, 3.8, 3.9, & 3.10.
3-R2	20-21	Section 4.2.3: Reinstated the EOCLB and End of Coast Exposures and Descriptions
4-R2	23-32	Section 4: Updated $MCPR_p$ Tables 4.2, 4.3, 4.4, 4.5, 4.6, 4.7, 4.8, & 4.9
0-R1	All	This is a major revision supporting all modes of operation, but is limited to BOC through NEOC cycle exposure. A future Revision will extend operation to expected end of cycle, and final feedwater temperature reduction and power coastdown. Cross reference to CR 1145408.
1-R1	3-37	Previous Section 2 becomes Section 8. New material for Sections 2 through 7.
0-R0	All	New document.

Nomenclature

APLHGR	Average Planar LHGR
APRM	Average Power Range Monitor
AREVA NP	Vendor (Framatome, Siemens)
BOC	Beginning of Cycle
BSP	Backup Stability Protection
BWR	Boiling Water Reactor
CAVEX	Core Average Exposure
CD	Coast Down
CMSS	Core Monitoring System Software
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRWE	Control Rod Withdrawal Error
CSDM	Cold SDM
DIVOM	Delta CPR over Initial CPR vs. Oscillation Magnitude
EOC	End of Cycle
EOCLB	End-of-Cycle Licensing Basis
EOOS	Equipment OOS
FFTR	Final Feedwater Temperature Reduction
FFWTR	Final Feedwater Temperature Reduction
FHOOS	Feedwater Heaters OOS
ft	Foot: English unit of measure for length
GNF	Vendor (General Electric, Global Nuclear Fuels)
GWd	Giga Watt Day
HTSP	High TSP
ICA	Interim Corrective Action
ICF	Increased Core Flow (beyond rated)
IS	In-Service
kW	kilo watt: SI unit of measure for power.
LCO	License Condition of Operation
LFWH	Loss of Feedwater Heating
LHGRFAC	LHGR Multiplier (Power or Flow dependent)
LPRM	Low Power Range Monitor
LRNB	Generator Load Reject, No Bypass
MAPFAC	MAPLHGR multiplier (Power or Flow dependent)
MCPR	Minimum CPR

MSRV	Moisture Separator Reheater Valve
MSRVOOS	MSRV OOS
MTU	Metric Ton Uranium
MWd/MTU	Mega Watt Day per Metric Ton Uranium
NEOC	Near EOC
NRC	United States Nuclear Regulatory Commission
NSS	Nominal Scram Speed
NTSP	Nominal TSP
OLMCPR	MCPR Operating Limit
OOS	Out-Of-Service
OPRM	Oscillation Power Range Monitor
OSS	Optimum Scram Speed
PBDA	Period Based Detection Algorithm
Pbypass	Power, below which TSV Position and TCV Fast Closure Scrams are Bypassed
PLU	Power Load Unbalance
PLUOOS	PLU OOS
PRNM	Power Range Neutron Monitor
RBM	Rod Block Monitor
RCPOOS	Recirculation Pump OOS (*SLO*)
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPTOOS	RPT OOS
SDM	Shutdown Margin
SLMCPR	MCPR Safety Limit
SLO	Single Loop Operation
TBV	Turbine Bypass Valve
TBVIS	TBV IS
TBVOOS	Turbine Bypass Valves OOS
TIP	Transversing In-core Probe
TIPOOS	TIP OOS
TLO	Two Loop Operation
TSP	Trip Setpoint
TSSS	Technical Specification Scram Speed
TVA	Tennessee Valley Authority

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1 Introduction

In anticipation of cycle startup, it is necessary to describe the expected limits of operation.

1.1 Purpose

The primary purpose of this document is to satisfy requirements identified by unit technical specification section 5.6.5. This document may be provided, upon final approval, to the NRC.

1.2 Scope

This document will discuss the following areas:

- Average Planar Linear Heat Generation Rate (APLHGR) Limit
  (Technical Specifications 3.2.1 and 3.7.5)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Linear Heat Generation Rate (LHGR) Limit
  (Technical Specification 3.2.3, 3.3.4.1, and 3.7.5)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Minimum Critical Power Ratio Operating Limit (OLMCPR)
  (Technical Specifications 3.2.2, 3.3.4.1, 3.7.5 and Table 3.3.2.1-1)
  Applicability: Mode 1, ≥ 25% RTP (Technical Specifications definition of RTP)
- Oscillation Power Range Monitor (OPRM) Setpoint
  (Technical Specification Table 3.3.1.1)
  Applicability: Mode 1, ≥ (as specified in Technical Specifications Table 3.3.1.1-1)
- Average Power Range Monitor (APRM) Flow Biased Rod Block Trip Setting
  (Technical Requirements Manual Section 5.3.1 and Table 3.3.4-1)
  Applicability: Mode 1, ≥ (as specified in Technical Requirements Manuals Table 3.3.4-1)
- Rod Block Monitor (RBM) Trip Setpoints and Operability
  (Technical Specification Table 3.3.2.1-1)
  Applicability: Mode 1, ≥ % RTP as specified in Table 3.3.2.1-1 (TS definition of RTP)
- Shutdown Margin (SDM) Limit
  (Technical Specification 3.1.1)
  Applicability: All Modes

1.3 Fuel Loading

The core will contain previously exposed AREVA NP, Inc., ATRIUM-10 fuel, along with fresh ATRIUM-10XM. Nuclear fuel types used in the core loading are shown in Table 1.1. The planned outage will consist of a full core off-load for maintenance. The final core loading was evaluated for BOC cold shutdown margin performance as documented per Reference 5.

Table 1.1 Nuclear Fuel Types*

Fuel Description	Original Cycle	Number of Assemblies	Nuclear Fuel Type (NFT)	Fuel Names (Range)
ATRIUM-10 A10-3440B-11GV80-FCE	16	70	12	FCE002-FCE144
ATRIUM-10 A10-3826B-13GV80-FCE	16	39	13	FCE145-FCE188
ATRIUM-10 A10-4075B-13GV80-FCE	16	15	14	FCE190-FCE235
ATRIUM-10 A10-4081B-12GV80-FCE	16	48	15	FCE237-FCE284
ATRIUM-10 A10-3849B-13GV80-FCF	17	176	16	FCF301-FCF476
ATRIUM-10 A10-3882B-10GV70-FCF	17	40	17	FCF477-FCF516
ATRIUM-10 A10-4116B-12GV70-FCF	17	72	18	FCF517-FCF588
ATRIUM-10XM XMLC-4105B-11GV70-FCG	18	72	19	FCG601-FCG672
ATRIUM-10XM XMLC-4096B-12GV80-FCG	18	136	20	FCG673-FCG808
ATRIUM-10XM XMLC-4055B-13GV70-FCG	18	96	21	FCG809-FCG904

1.4 Acceptability

Limits discussed in this document were generated based on NRC approved methodologies per References 6 through 25.

* The table identifies the expected fuel type breakdown in anticipation of final core loading. The final composition of the core depends upon uncertainties during the outage such as discovering a failed fuel bundle, or other bundle damage. Minor core loading changes, due to unforeseen events, will conform to the safety and monitoring requirements identified in this document.

2 APLHGR Limits

(Technical Specifications 3.2.1 & 3.7.5)

The APLHGR limit is determined by adjusting the rated power APLHGR limit for off-rated power, off-rated flow, and SLO conditions. The most limiting of these is then used as follows:

$$\text{APLHGR limit} = \text{MIN} \left(\text{APLHGR}_P , \text{APLHGR}_F, \text{APLHGR}_{SLO} \right)$$

where:

$APLHGR_P$	off-rated power APLHGR limit	[$APLHGR_{RATED}$ * $MAPFAC_P$]
$APLHGR_F$	off-rated flow APLHGR limit	[$APLHGR_{RATED}$ * $MAPFAC_F$]
$APLHGR_{SLO}$	SLO APLHGR limit	[$APLHGR_{RATED}$ * SLO Multiplier]

2.1 Rated Power and Flow Limit: $APLHGR_{RATED}$

The rated conditions APLHGR for ATRIUM-10 fuel is identified in Reference 1 and shown in Figure 2.1. The rated conditions APLHGR for ATRIUM-10XM are shown in Figure 2.2.

2.2 Off-Rated Power Dependent Limit: $APLHGR_P$

Reference 1 does not specify a power dependent APLHGR. Therefore, $MAPFAC_P$ is set to a value of **1.0**.

2.2.1 *Startup without Feedwater Heaters*

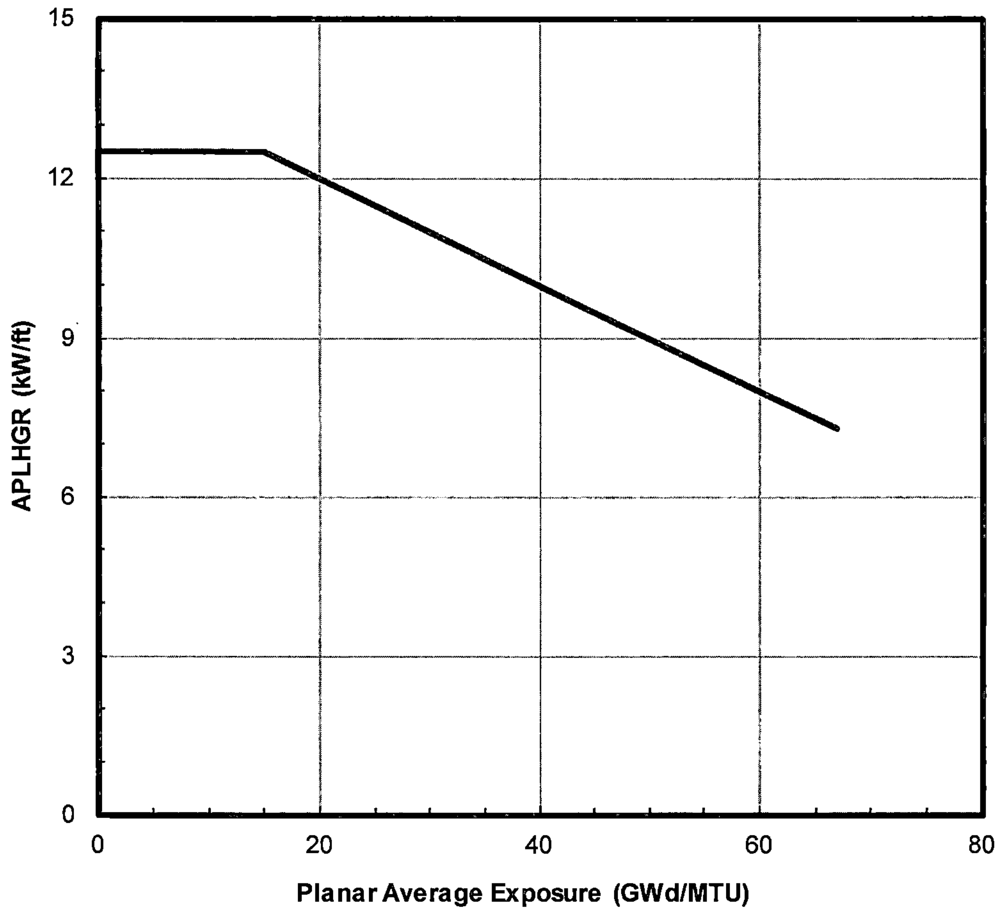
There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. No additional power dependent limitation is required.

2.3 Off-Rated Flow Dependent Limit: $APLHGR_F$

Reference 1 does not specify a flow dependent APLHGR. Therefore, $MAPFAC_F$ is set to a value of **1.0**.

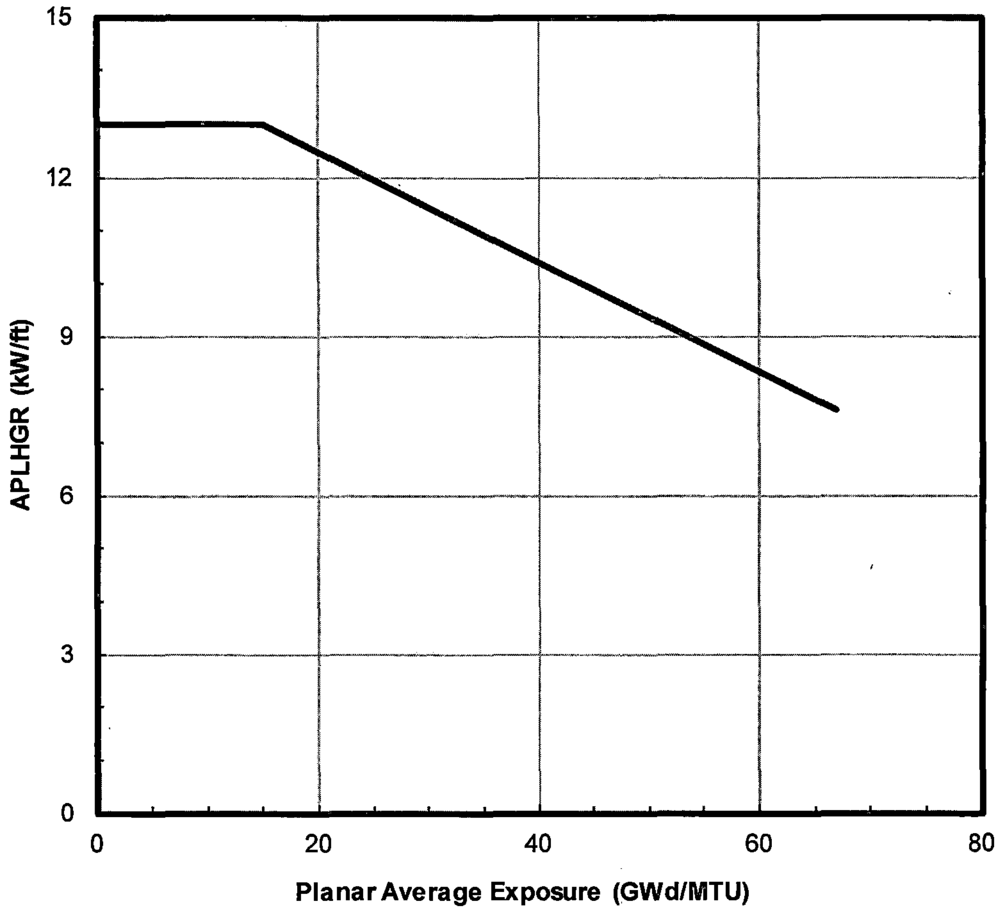
2.4 Single Loop Operation Limit: $APLHGR_{SLO}$

The single loop operation multiplier for ATRIUM-10, and ATRIUM-10XM fuel is **0.85**, per Reference 1.



Planar Avg. Exposure (GWd/MTU)	APLHGR Limit (kW/ft)
0.0	12.5
15.0	12.5
67.0	7.3

Figure 2.1 APLHGR$_{RATED}$ for ATRIUM-10 Fuel



Planar Avg. Exposure (GWd/MTU)	APLHGR Limit (kW/ft)
0.0	13.0
15.0	13.0
67.0	7.6

Figure 2.2 $APLHGR_{RATED}$ for ATRIUM-10XM Fuel

2.5 Equipment Out-Of-Service Corrections

The limits shown in Figure 2.1 and Figure 2.2 are applicable for operation with all equipment In-Service as well as the following Equipment Out-Of-Service (EOOS) options; including combinations of the options.

In-Service	All equipment In-Service*
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
PLUOOS	Power Load Unbalance Out-Of-Service
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service or Final Feedwater Temperature Reduction
RCPOOS	One Recirculation Pump Out-Of-Service

* All equipment service conditions assume 1 SRVOOS.

3 LHGR Limits

(Technical Specification 3.2.3, 3.3.4.1, & 3.7.5)

The LHGR limit is determined by adjusting the rated power LHGR limit for off-rated power and off-rated flow conditions. The most limiting of these is then used as follows:

LHGR limit = MIN ($LHGR_P$, $LHGR_F$)

where:

$LHGR_P$	off-rated power LHGR limit	[$LHGR_{RATED}$ * $LHGRFAC_P$]
$LHGR_F$	off-rated flow LHGR limit	[$LHGR_{RATED}$ * $LHGRFAC_F$]

3.1 Rated Power and Flow Limit: $LHGR_{RATED}$

The rated conditions LHGR for ATRIUM-10 fuel is identified in Reference 1 and shown in Figure 3.1. The rated conditions LHGR for ATRIUM-10XM fuel is shown in Figure 3.2. The LHGR limit is consistent with References 2 and 3.

3.2 Off-Rated Power Dependent Limit: $LHGR_P$

LHGR limits are adjusted for off-rated power conditions using the $LHGRFAC_P$ multiplier provided in Reference 1. The multiplier is split into two sub cases: turbine bypass valves in and out-of-service. The base case multipliers are shown in Figure 3.3 and Figure 3.4.

3.2.1 Startup without Feedwater Heaters

There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. Additional limits are shown in Figure 3.7 through Figure 3.10, based on temperature conditions identified in Table 3.1.

Table 3.1 Startup Feedwater Temperature Basis

Power	Temperature Range 1	Temperature Range 2
(% Rated)	(°F)	(°F)
25	160.0	155.0
30	165.0	160.0
40	175.0	170.0
50	185.0	180.0

3.3 Off-Rated Flow Dependent Limit: $LHGR_F$

LHGR limits are adjusted for off-rated flow conditions using the $LHGRFAC_F$ multiplier provided in Reference 1. Multipliers are shown in Figure 3.5 and Figure 3.6.

3.4 Equipment Out-Of-Service Corrections

The limits shown in Figure 3.1 and Figure 3.2 are applicable for operation with all equipment In-Service as well as the following Equipment Out-Of-Service (EOOS) options; including combinations of the options.*

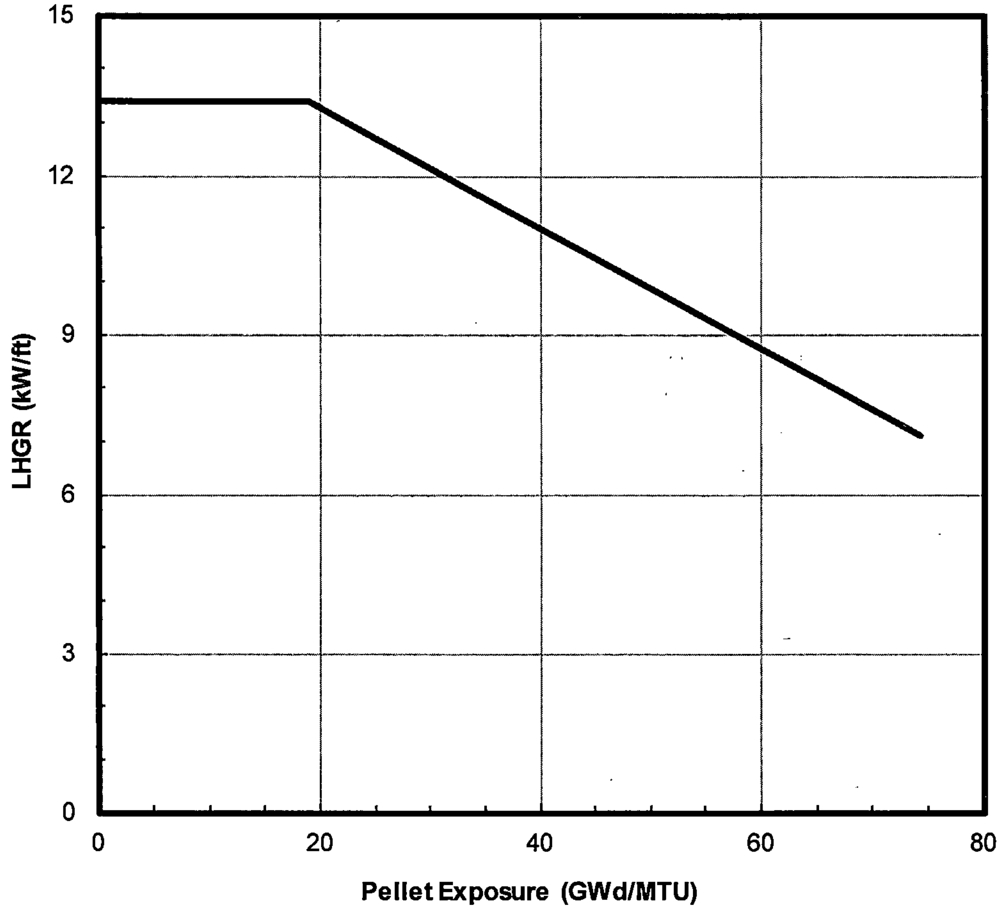
In-Service	All equipment In-Service
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
PLUOOS	Power Load Unbalance Out-Of-Service
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service or Final Feedwater Temperature Reduction
RCPOOS	One Recirculation Pump Out-Of-Service

Off-rated power corrections shown in Figure 3.3 and Figure 3.4 are dependent on operation of the Turbine Bypass Valve system. For this reason, separate limits are to be applied for TBVIS or TBVOOS operation. The limits have no dependency on RPTOOS, PLUOOS, FHOOS/FFWTR, or SLO.

Off-rated flow corrections shown in Figure 3.5 and Figure 3.6 are bounding for all EOOS conditions.

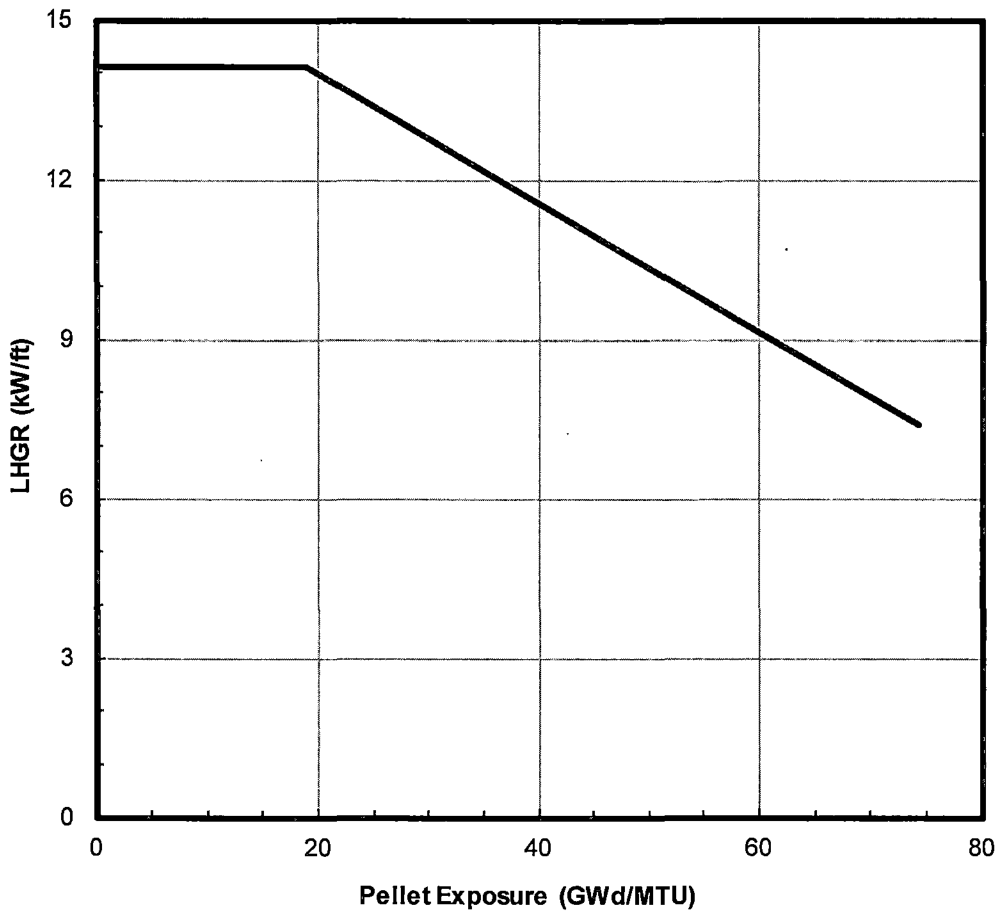
Off-rated power corrections shown in Figure 3.7 through Figure 3.10 are also dependent on operation of the Turbine Bypass Valve system. In this case, limits support FHOOS operation during startup. These limits have no dependency on RPTOOS, PLUOOS, or SLO.

* All equipment service conditions assume 1 SRVOOS.



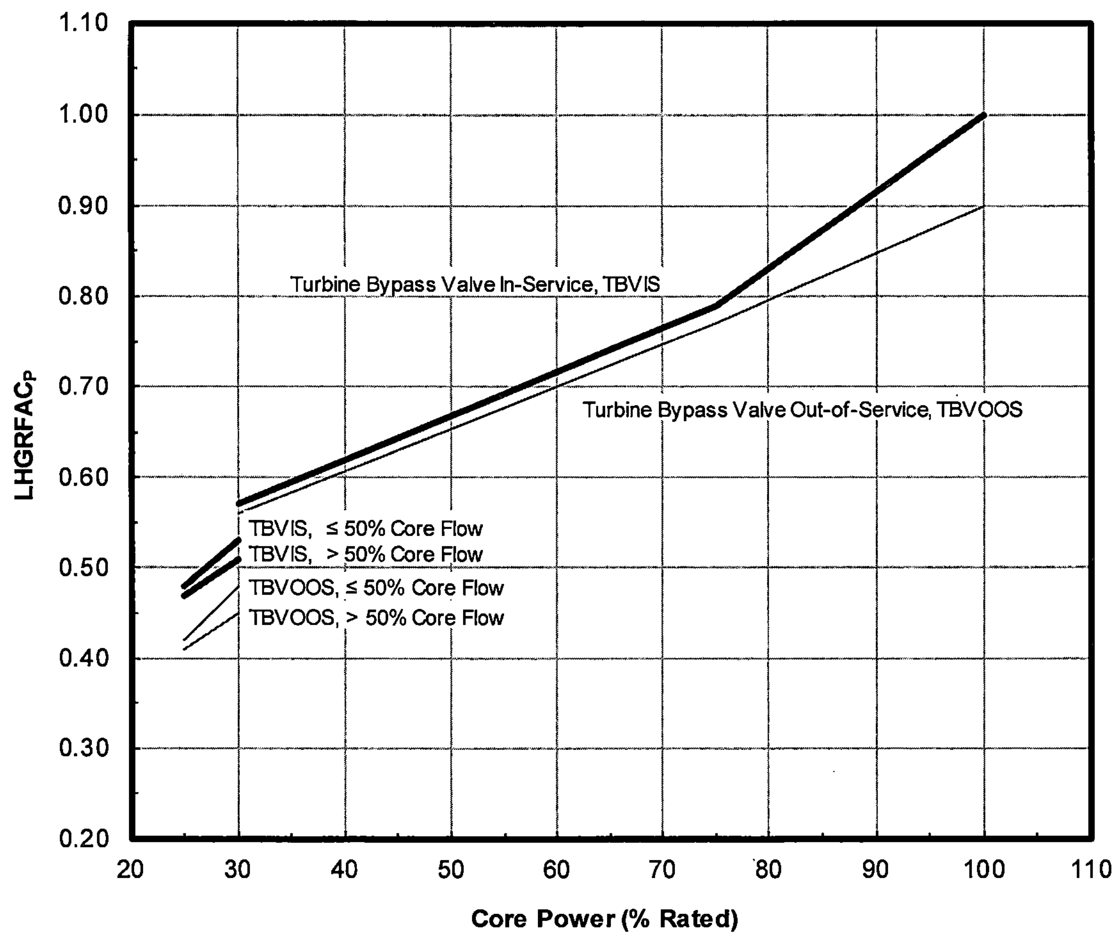
Pellet Exposure (GWd/MTU)	LHGR Limit (kW/ft)
0.0	13.4
18.9	13.4
74.4	7.1

Figure 3.1 LHGR$_{RATED}$ for ATRIUM-10 Fuel



Pellet Exposure (GWd/MTU)	LHGR Limit (kW/ft)
0.0	14.1
18.9	14.1
74.4	7.4

Figure 3.2 $LHGR_{RATED}$ for ATRIUM-10XM Fuel

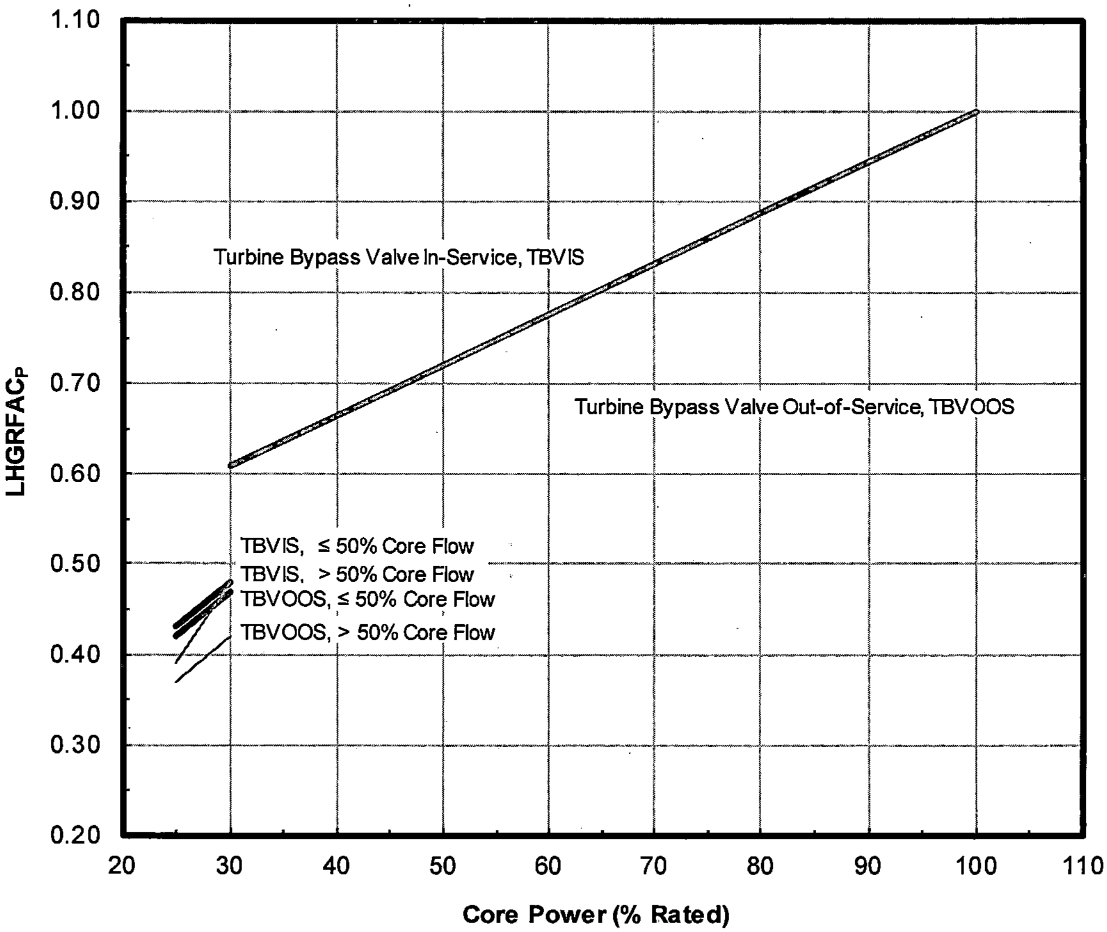


Turbine Bypass In-Service Core Power (% Rated)	$LHGRFAC_P$
100.0	1.00
75.0	0.79
30.0	0.57
Core Flow > 50% Rated	
30.0	0.51
25.0	0.47
Core Flow ≤ 50% Rated	
30.0	0.53
25.0	0.48

Turbine Bypass Out-of-Service Core Power (% Rated)	$LHGRFAC_P$
100.0	0.90
75.0	0.77
30.0	0.56
Core Flow > 50% Rated	
30.0	0.45
25.0	0.41
Core Flow ≤ 50% Rated	
30.0	0.48
25.0	0.42

Figure 3.3 Base Operation $LHGRFAC_P$ for ATRIUM-10 Fuel

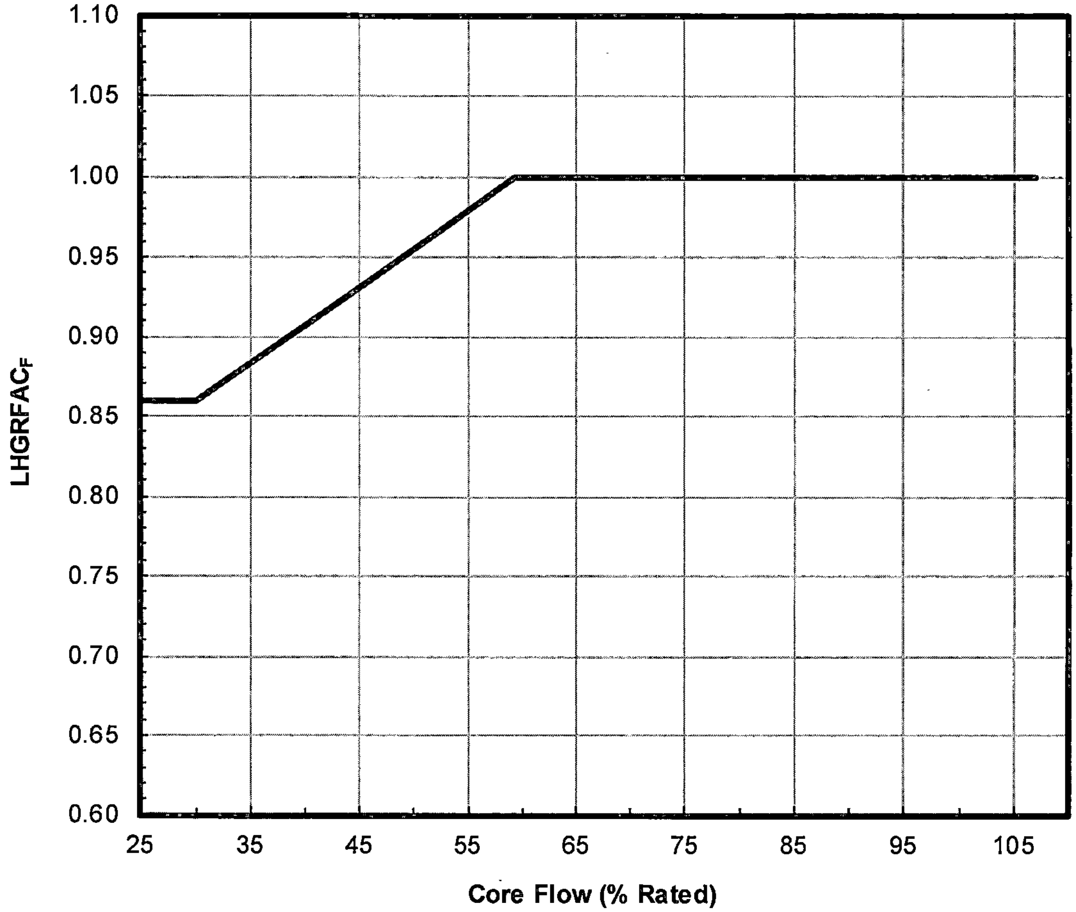
(*Independent of other EOOS conditions*)



Turbine Bypass In-Service		*Turbine Bypass Out-of-Service*	
Core Power (% Rated)	**$LHGRFAC_P$**	**Core Power (% Rated)**	**$LHGRFAC_P$**
100.0	1.00	100.0	1.00
30.0	0.61	30.0	0.61
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.47	30.0	0.42
25.0	0.42	25.0	0.37
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.48	30.0	0.48
25.0	0.43	25.0	0.39

Figure 3.4 Base Operation $LHGRFAC_P$ for ATRIUM-10XM Fuel

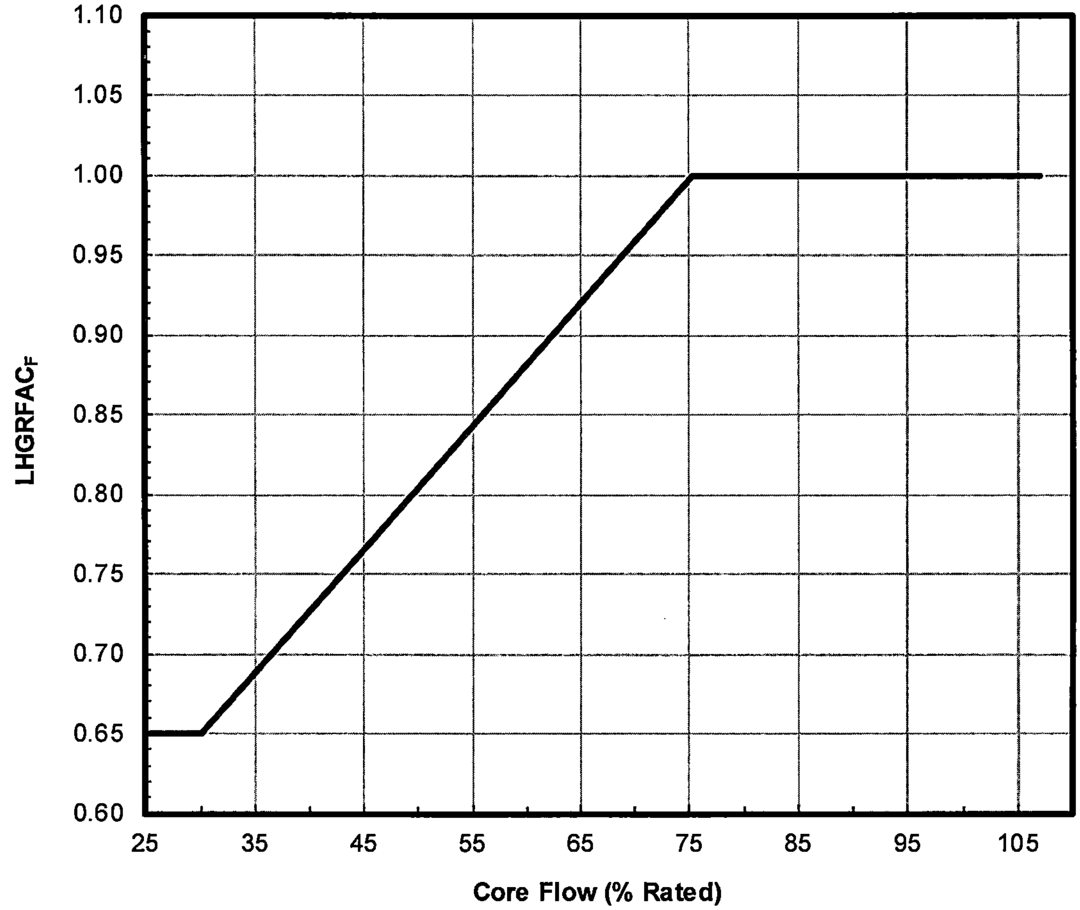
(*Independent of other EOOS conditions*)



Core Flow (% Rated)	$LHGRFAC_F$
0.0	0.86
30.0	0.86
59.3	1.00
107.0	1.00

Figure 3.5 $LHGRFAC_F$ for ATRIUM-10 Fuel
(*Values bound all EOOS conditions*)

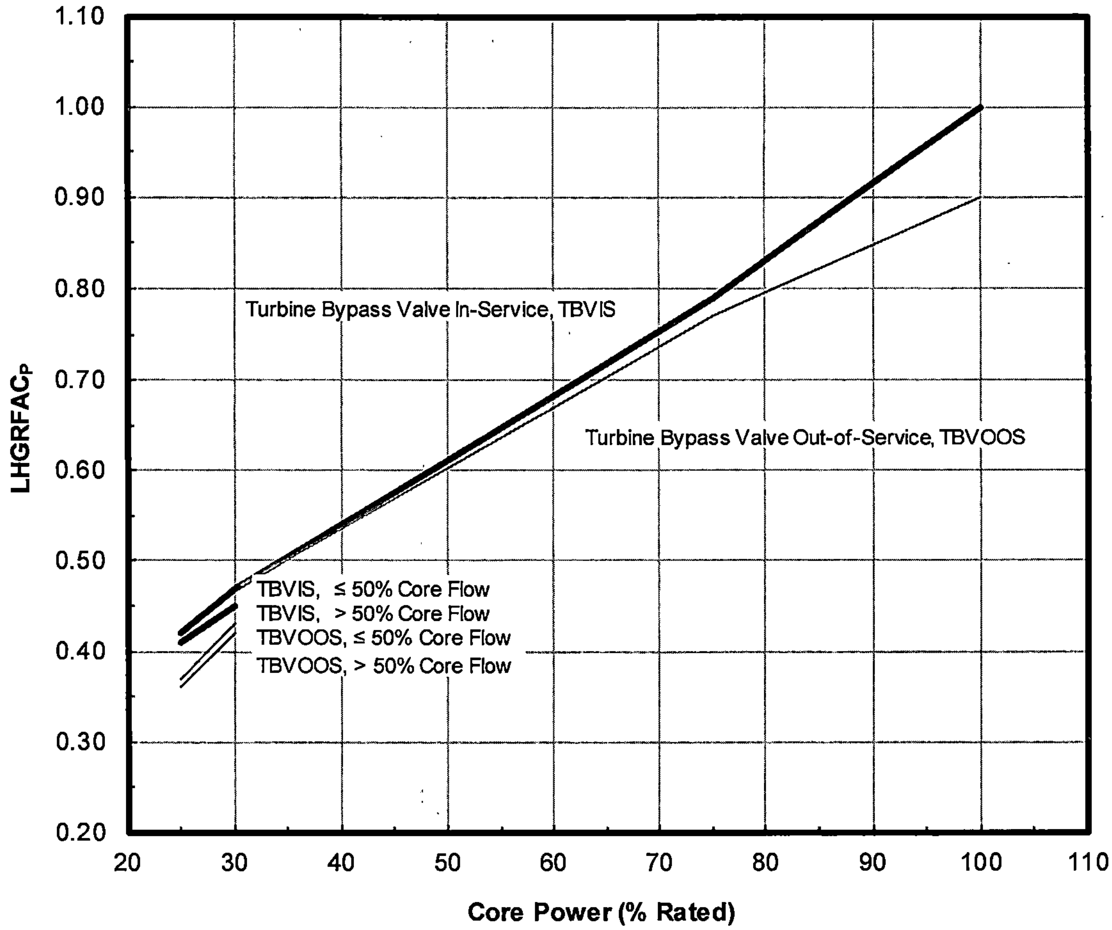
(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)



Core Flow (% Rated)	$LHGRFAC_F$
0.0	0.65
30.0	0.65
75.2	1
107.0	1

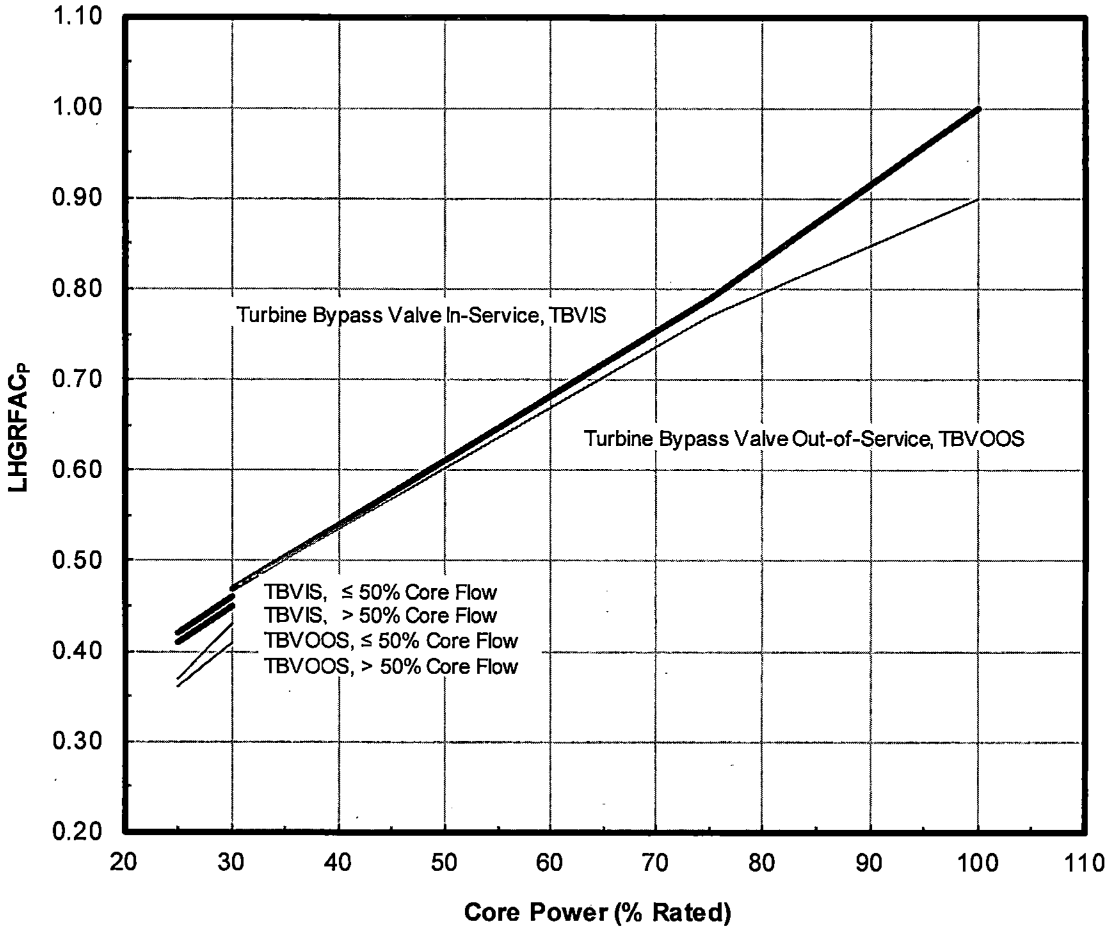
Figure 3.6 $LHGRFAC_F$ for ATRIUM-10XM Fuel
(*Values bound all EOOS conditions*)

(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)



Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$	*Turbine Bypass Out-of-Service* Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00	100.0	0.90
75.0	0.79	75.0	0.77
30.0	0.47	30.0	0.47
Core Flow > 50% Rated		**Core Flow > 50% Rated**	
30.0	0.45	30.0	0.42
25.0	0.41	25.0	0.36
Core Flow ≤ 50% Rated		**Core Flow ≤ 50% Rated**	
30.0	0.47	30.0	0.43
25.0	0.42	25.0	0.37

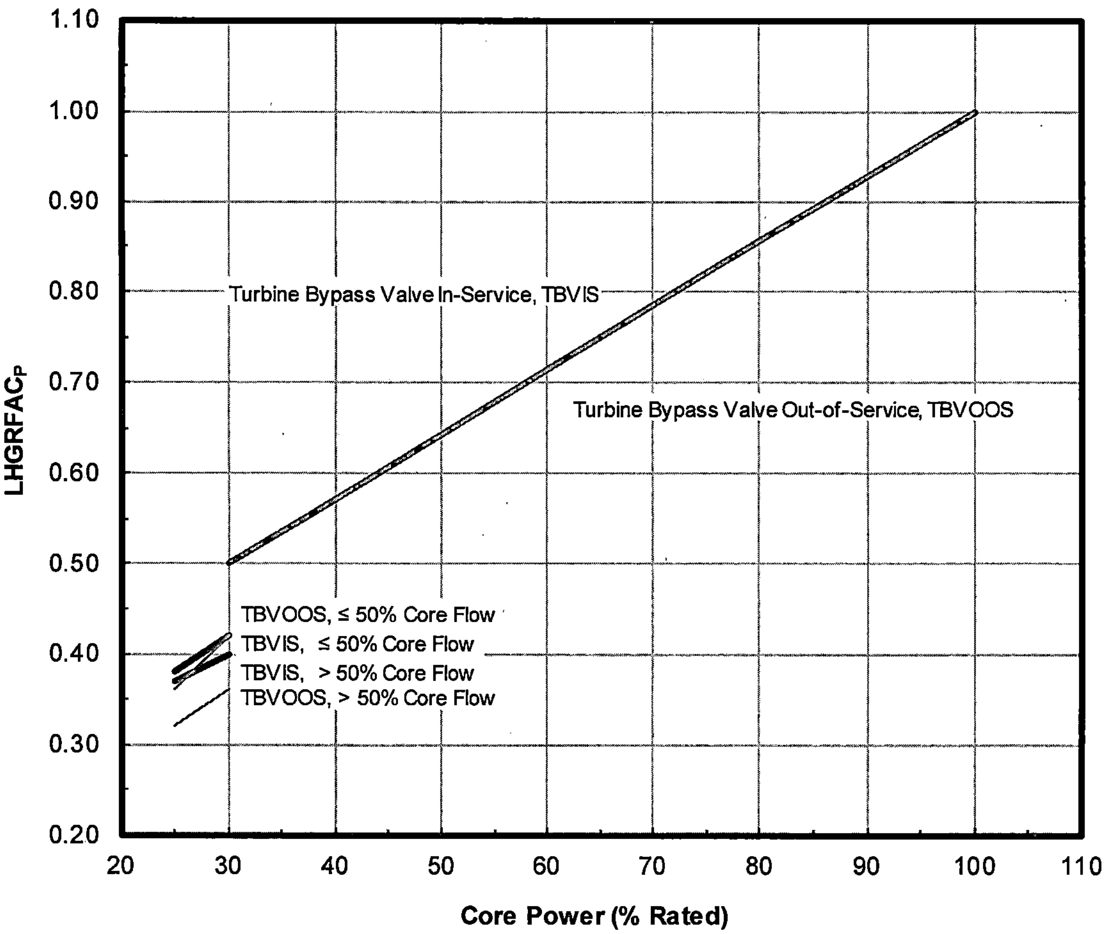
Figure 3.7 Startup Operation LHGRFAC$_P$ for ATRIUM-10 Fuel:
Table 3.1 Temperature Range 1
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



Turbine Bypass In-Service	
Core Power	**LHGRFAC$_P$**
(% Rated)	
100.0	1.00
75.0	0.79
30.0	0.47
Core Flow > 50% Rated	
30.0	0.45
25.0	0.41
Core Flow ≤ 50% Rated	
30.0	0.46
25.0	0.42

Turbine Bypass Out-of-Service	
Core Power	**LHGRFAC$_P$**
(% Rated)	
100.0	0.90
75.0	0.77
30.0	0.47
Core Flow > 50% Rated	
30.0	0.41
25.0	0.36
Core Flow ≤ 50% Rated	
30.0	0.43
25.0	0.37

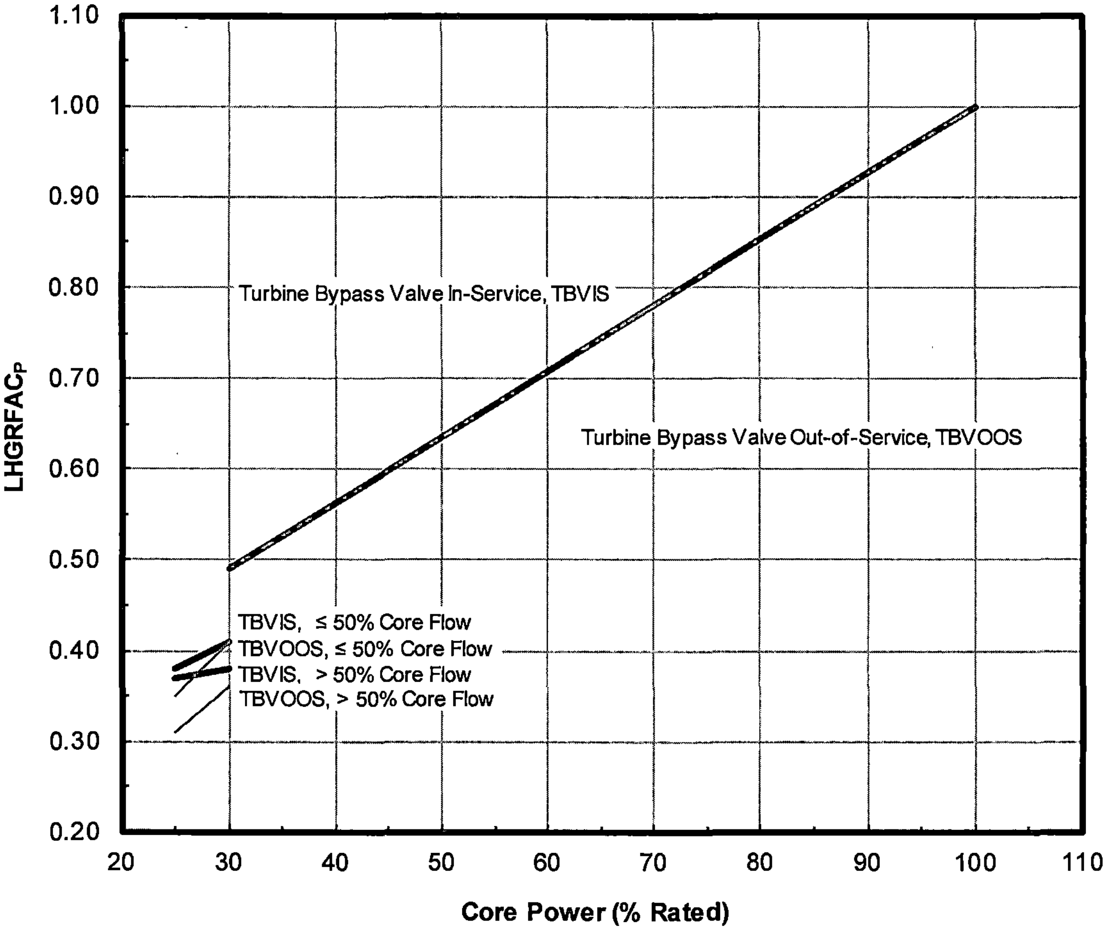
Figure 3.8 Startup Operation LHGRFAC$_P$ for ATRIUM-10 Fuel:
Table 3.1 Temperature Range 2
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



Turbine Bypass In-Service Core Power	LHGRFAC$_P$
(% Rated)	
100.0	1.00
30.0	0.50
Core Flow > 50% Rated	
30.0	0.40
25.0	0.37
Core Flow ≤ 50% Rated	
30.0	0.42
25.0	0.38

Turbine Bypass Out-of-Service Core Power	LHGRFAC$_P$
(% Rated)	
100.0	1.00
30.0	0.50
Core Flow > 50% Rated	
30.0	0.36
25.0	0.32
Core Flow ≤ 50% Rated	
30.0	0.42
25.0	0.36

Figure 3.9 Startup Operation LHGRFAC$_P$ for ATRIUM-10XM Fuel:
Table 3.1 Temperature Range 1
(no Feedwater heating during startup)
(Limits valid at and below 50% power)



Turbine Bypass In-Service Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00
30.0	0.49
Core Flow > 50% Rated	
30.0	0.38
25.0	0.37
Core Flow ≤ 50% Rated	
30.0	0.41
25.0	0.38

Turbine Bypass Out-of-Service Core Power (% Rated)	LHGRFAC$_P$
100.0	1.00
30.0	0.49
Core Flow > 50% Rated	
30.0	0.36
25.0	0.31
Core Flow ≤ 50% Rated	
30.0	0.41
25.0	0.35

Figure 3.10 Startup Operation LHGRFAC$_P$ for ATRIUM-10XM
Fuel: Table 3.1 Temperature Range 2
(no Feedwater heating during startup)
(Limits valid at and below 50% power)

4 OLMCPR Limits

(Technical Specification 3.2.2, 3.3.4.1, & 3.7.5)

OLMCPR is calculated to be the most limiting of the flow or power dependent values

$$\text{OLMCPR limit} = \text{MAX} (\text{MCPR}_F , \text{MCPR}_P)$$

where:

$MCPR_F$	core flow-dependent MCPR limit
$MCPR_P$	power-dependent MCPR limit

4.1 Flow Dependent MCPR Limit: $MCPR_F$

$MCPR_F$ limits are dependent upon core flow (% of Rated), and the max core flow limit, (Rated or Increased Core Flow, ICF). $MCPR_F$ limits are shown in Figure 4.1, per Reference 1. Limits are valid for all EOOS combinations. No adjustment is required for SLO conditions.

4.2 Power Dependent MCPR Limit: $MCPR_P$

$MCPR_P$ limits are dependent upon:

- Core Power Level (% of Rated)
- Technical Specification Scram Speed (TSSS), Nominal Scram Speed (NSS), or Optimum Scram Speed (OSS)
- Cycle Operating Exposure (NEOC, EOC, and CD - as defined in this section)
- Equipment Out-Of-Service Options
- Two or Single recirculation Loop Operation (TLO vs. SLO)

The $MCPR_P$ limits are provided in Table 4.2 through Table 4.9, where each table contains the limits for all fuel types and EOOS options (for a specified scram speed and exposure range). The CMSS determines $MCPR_P$ limits, from these tables, based on linear interpolation between the specified powers.

4.2.1 <u>Startup without Feedwater Heaters</u>

There is a range of operation during startup when the feedwater heaters are not placed into service until after the unit has reached a significant operating power level. Additional power dependent limits are shown in Table 4.5 through Table 4.8 based on temperature conditions identified in Table 3.1.

4.2.2 Scram Speed Dependent Limits (TSSS vs. NSS vs. OSS)

$MCPR_P$ limits are provided for three different sets of assumed scram speeds. The Technical Specification Scram Speed (TSSS) $MCPR_P$ limits are applicable at all times, as long as the scram time surveillance demonstrates the times in Technical Specification Table 3.1.4-1 are met. Both Nominal Scram Speeds (NSS) and/or Optimum Scram Speeds (OSS) may be used, as long as the scram time surveillance demonstrates Table 4.1 times are applicable.*†

Table 4.1 Nominal Scram Time Basis

Notch Position	Nominal Scram Timing	Optimum Scram Timing
(index)	(seconds)	(seconds)
46	0.420	0.380
36	0.980	0.875
26	1.600	1.465
6	2.900	2.900

In demonstrating compliance with the NSS and/or OSS scram time basis, surveillance requirements from Technical Specification 3.1.4 apply; accepting the definition of SLOW rods should conform to scram speeds shown in Table 4.1. If conformance is not demonstrated, TSSS based $MCPR_P$ limits are applied.

On initial cycle startup, TSSS limits are used until the successful completion of scram timing confirms NSS and/or OSS based limits are applicable.

4.2.3 Exposure Dependent Limits

Exposures are tracked on a Core Average Exposure basis (CAVEX, not Cycle Exposure). Higher exposure $MCPR_P$ limits are always more limiting and may be used for any Core Average Exposure up to the ending exposure. Per Reference 1, $MCPR_P$ limits are provided for the following exposure ranges:

BOC to NEOC	NEOC corresponds to	**28,825.7 MWd / MTU**
BOC to EOCLB	EOCLB corresponds to	**33,065.8 MWd / MTU**
BOC to End of Coast	End of Coast	**34,368.7 MWd / MTU**

NEOC refers to a Near EOC exposure point.

* Reference 1 analysis results are based on information identified in Reference 4.

† Drop out times consistent with method used to perform actual timing measurements (i.e., including pickup/dropout effects).

The EOCLB exposure point is not the true End-Of-Cycle Exposure. Instead it corresponds to a licensing exposure window exceeding expected end-of-full-power-life.

The End of Coast exposure point represents a licensing exposure point exceeding the expected end-of-cycle exposure including cycle extension options.

4.2.4 *Equipment Out-Of-Service (EOOS) Options*

EOOS options* covered by $MCPR_P$ limits are given by the following:

In-Service	All equipment In-Service
RPTOOS	EOC-Recirculation Pump Trip Out-Of-Service
TBVOOS	Turbine Bypass Valve(s) Out-Of-Service
RPTOOS+TBVOOS	Combined RPTOOS and TBVOOS
PLUOOS	Power Load Unbalance Out-Of-Service
PLUOOS+RPTOOS	Combined PLUOOS and RPTOOS
PLUOOS+TBVOOS	Combined PLUOOS and TBVOOS
PLUOOS+TBVOOS+RPTOOS	Combined PLUOOS, RPTOOS, and TBVOOS
FHOOS (or FFWTR)	Feedwater Heaters Out-Of-Service (or Final Feedwater Temperature Reduction)
RCPOOS	One Recirculation Pump Out-Of-Service

For exposure ranges up to NEOC additional combinations of $MCPR_P$ limits are also provided including FHOOS.

4.2.5 *Single-Loop-Operation (SLO) Limits*

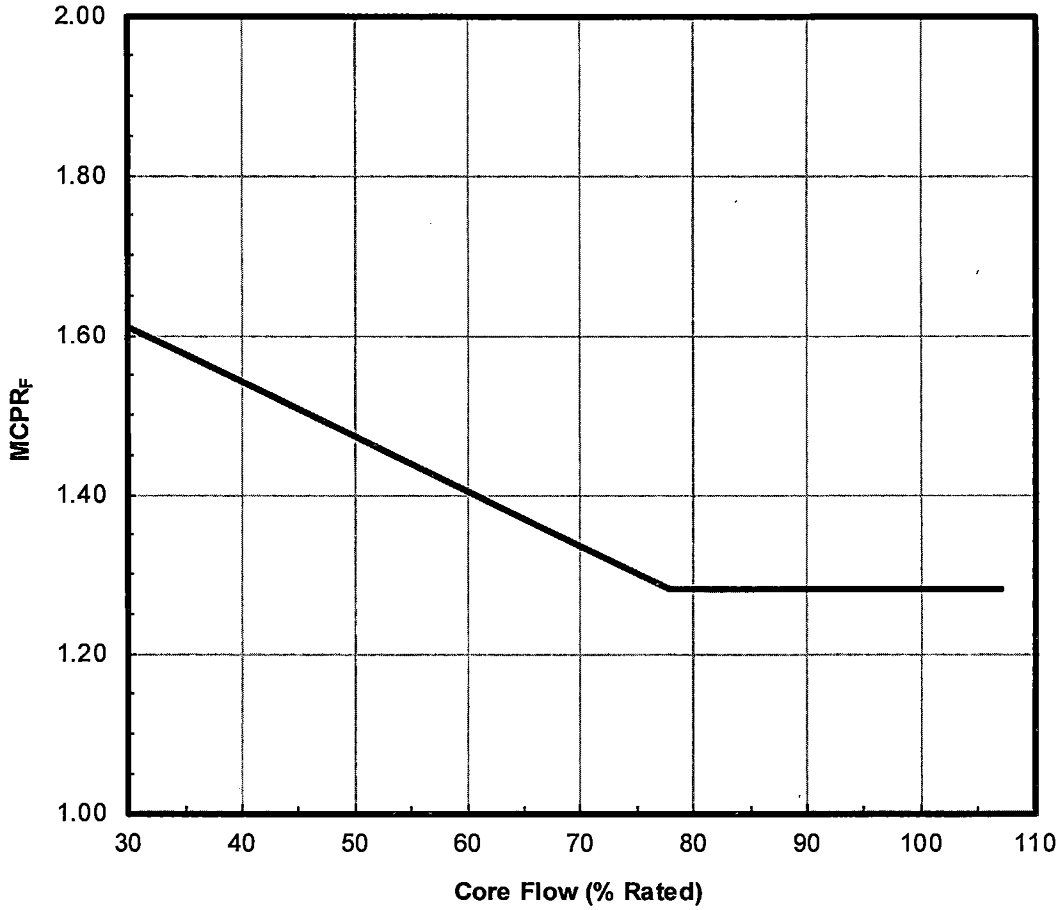
When operating in RCPOOS conditions, MCPRp limits are constructed differently from the normal operating RCP conditions. The limiting event for RCPOOS is a pump seizure scenario, which sets the upper bound for allowed core power and flow†. This event is not impacted by scram time assumptions. Specific $MCPR_P$ limits are shown in Table 4.9.

4.2.6 *Below Pbypass Limits*

Below Pbypass (30% rated power), $MCPR_P$ limits depend upon core flow. One set of $MCPR_P$ limits applies for core flow above 50% of rated; a second set applies if the core flow is less than or equal to 50% rated.

* All equipment service conditions assume 1 SRVOOS.

† RCPOOS limits are only valid up to 50% rated core power, 50% rated core flow, and an active recirculation drive flow of 17.73 Mlb_m/hr.



Core Flow (% Rated)	$MCPR_F$
30.0	1.61
78.0	1.28
107.0	1.28

Figure 4.1 $MCPR_F$ for All Fuel Types
(*Values bound all EOOS conditions*)

(*107.0% maximum core flow line is used to support 105% rated flow operation, ICF*)

Table 4.2 MCPRp Limits for All Fuel Types: Optimum Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
Base Case	100	1.47	1.49	1.51	1.41	1.43	1.45
	75	1.61	1.61	1.65	1.54	1.54	1.58
	65	1.68	1.68	1.73	1.60	1.60	1.64
	50	1.83	1.83	1.92	1.73	1.73	1.80
	50	1.96	1.96	1.97	1.81	1.81	1.82
	40	2.08	2.08	2.19	1.97	1.97	2.07
	30	2.45	2.45	2.58	2.29	2.29	2.42
	30 at > 50%F	2.49	2.49	2.60	2.48	2.48	2.65
	25 at > 50%F	2.74	2.74	2.86	2.57	2.57	2.69
	30 at ≤ 50%F	2.45	2.45	2.58	2.29	2.29	2.42
	25 at ≤ 50%F	2.66	2.66	2.76	2.46	2.46	2.58
FHOOS	100	1.50	1.51	---	1.43	1.45	---
	75	1.65	1.65	---	1.58	1.58	---
	65	1.73	1.73	---	1.64	1.64	---
	50	1.92	1.92	---	1.80	1.80	---
	50	1.97	1.97	---	1.82	1.82	---
	40	2.19	2.19	---	2.07	2.07	---
	30	2.58	2.58	---	2.42	2.42	---
	30 at > 50%F	2.60	2.60	---	2.65	2.65	---
	25 at > 50%F	2.86	2.86	---	2.69	2.69	---
	30 at ≤ 50%F	2.58	2.58	---	2.42	2.42	---
	25 at ≤ 50%F	2.76	2.76	---	2.58	2.58	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR/FHOOS is supported for the BOC to End of Coast limits.

Table 4.3 MCPRP Limits for All Fuel Types: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
Base Case	100	1.50	1.51	1.53	1.43	1.44	1.46
	75	1.63	1.63	1.66	1.55	1.55	1.60
	65	1.69	1.69	1.74	1.61	1.61	1.65
	50	1.88	1.88	1.97	1.76	1.76	---
	50	1.97	1.97	1.98	1.81	1.81	1.83
	40	2.13	2.13	2.24	2.00	2.00	2.10
	30	2.50	2.50	2.64	2.31	2.31	2.45
	30 at > 50%F	2.50	2.50	2.64	2.48	2.48	2.65
	25 at > 50%F	2.74	2.74	2.86	2.57	2.57	2.69
	30 at ≤ 50%F	2.50	2.50	2.64	2.31	2.31	2.45
	25 at ≤ 50%F	2.66	2.66	2.76	2.46	2.46	2.58
TBVOOS	100	1.56	1.57	1.58	1.46	1.47	1.49
	75	1.69	1.69	1.72	1.59	1.59	1.63
	65	1.75	1.75	1.80	1.65	1.65	1.69
	50	1.90	1.90	---	1.78	1.78	---
	50	1.97	1.97	1.98	1.81	1.81	1.84
	40	2.13	2.13	2.24	2.00	2.00	2.10
	30	2.50	2.50	2.64	2.31	2.31	2.45
	30 at > 50%F	3.06	3.06	3.17	3.02	3.02	3.13
	25 at > 50%F	3.44	3.44	3.53	3.23	3.23	3.32
	30 at ≤ 50%F	2.86	2.86	2.98	2.61	2.61	2.73
	25 at ≤ 50%F	3.26	3.26	3.41	3.00	3.00	3.15
FHOOS	100	1.52	1.53	---	1.45	1.46	---
	75	1.66	1.66	---	1.60	1.60	---
	65	1.74	1.74	---	1.65	1.65	---
	50	1.97	1.97	---	---	---	---
	50	1.98	1.98	---	1.83	1.83	---
	40	2.24	2.24	---	2.10	2.10	---
	30	2.64	2.64	---	2.45	2.45	---
	30 at > 50%F	2.64	2.64	---	2.65	2.65	---
	25 at > 50%F	2.86	2.86	---	2.69	2.69	---
	30 at ≤ 50%F	2.64	2.64	---	2.45	2.45	---
	25 at ≤ 50%F	2.76	2.76	---	2.58	2.58	---
PLUOOS	100	1.50	1.51	1.53	1.43	1.44	1.46
	75	1.63	1.63	1.66	1.55	1.55	1.60
	65	1.86	1.86	1.86	1.72	1.74	1.74
	50	---	---	---	---	---	---
	50	1.97	1.97	1.98	1.81	1.81	1.83
	40	2.13	2.13	2.24	2.00	2.00	2.10
	30	2.50	2.50	2.64	2.31	2.31	2.45
	30 at > 50%F	2.50	2.50	2.64	2.48	2.48	2.65
	25 at > 50%F	2.74	2.74	2.86	2.57	2.57	2.69
	30 at ≤ 50%F	2.50	2.50	2.64	2.31	2.31	2.45
	25 at ≤ 50%F	2.66	2.66	2.76	2.46	2.46	2.58

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.3 MCPR$_P$ Limits for All Fuel Types: Nominal Scram Time Basis (*continued*)*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
TBVOOS FHOOS	100	1.57	1.58	---	1.48	1.49	---
	75	1.72	1.72	---	1.63	1.63	---
	65	1.80	1.80	---	1.69	1.69	---
	50	---	---	---	---	---	---
	50	1.98	1.98	---	1.84	1.84	---
	40	2.24	2.24	---	2.10	2.10	---
	30	2.64	2.64	---	2.45	2.45	---
	30 at > 50%F	3.17	3.17	---	3.13	3.13	---
	25 at > 50%F	3.53	3.53	---	3.32	3.32	---
	30 at ≤ 50%F	2.98	2.98	---	2.73	2.73	---
	25 at ≤ 50%F	3.41	3.41	---	3.15	3.15	---
TBVOOS PLUOOS	100	1.56	1.57	1.58	1.46	1.47	1.49
	75	1.69	1.69	1.72	1.59	1.59	1.63
	65	1.86	1.86	1.86	1.72	1.74	1.74
	50	---	---	---	---	---	---
	50	1.97	1.97	1.98	1.81	1.81	1.84
	40	2.13	2.13	2.24	2.00	2.00	2.10
	30	2.50	2.50	2.64	2.31	2.31	2.45
	30 at > 50%F	3.06	3.06	3.17	3.02	3.02	3.13
	25 at > 50%F	3.44	3.44	3.53	3.23	3.23	3.32
	30 at ≤ 50%F	2.86	2.86	2.98	2.61	2.61	2.73
	25 at ≤ 50%F	3.26	3.26	3.41	3.00	3.00	3.15
FHOOS PLUOOS	100	1.52	1.53	---	1.45	1.46	---
	75	1.66	1.66	---	1.60	1.60	---
	65	1.86	1.86	---	1.72	1.74	---
	50	---	---	---	---	---	---
	50	1.98	1.98	---	1.83	1.83	---
	40	2.24	2.24	---	2.10	2.10	---
	30	2.64	2.64	---	2.45	2.45	---
	30 at > 50%F	2.64	2.64	---	2.65	2.65	---
	25 at > 50%F	2.86	2.86	---	2.69	2.69	---
	30 at ≤ 50%F	2.64	2.64	---	2.45	2.45	---
	25 at ≤ 50%F	2.76	2.76	---	2.58	2.58	---
TBVOOS FHOOS PLUOOS	100	1.57	1.58	---	1.48	1.49	---
	75	1.72	1.72	---	1.63	1.63	---
	65	1.86	1.86	---	1.72	1.74	---
	50	---	---	---	---	---	---
	50	1.98	1.98	---	1.84	1.84	---
	40	2.24	2.24	---	2.10	2.10	---
	30	2.64	2.64	---	2.45	2.45	---
	30 at > 50%F	3.17	3.17	---	3.13	3.13	---
	25 at > 50%F	3.53	3.53	---	3.32	3.32	---
	30 at ≤ 50%F	2.98	2.98	---	2.73	2.73	---
	25 at ≤ 50%F	3.41	3.41	---	3.15	3.15	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.4 MCPR$_P$ Limits for All Fuel Types: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
Base Case	100	1.52	1.52	1.54	1.44	1.45	1.47
	75	1.64	1.64	1.68	1.56	1.56	1.60
	65	1.71	1.71	1.78	1.62	1.62	1.67
	50	1.92	1.92	---	1.79	1.79	---
	50	1.98	1.98	2.01	1.82	1.82	1.86
	40	2.17	2.17	2.29	2.03	2.03	2.13
	30	2.55	2.55	2.69	2.34	2.34	2.48
	30 at > 50%F	2.55	2.55	2.69	2.48	2.48	2.65
	25 at > 50%F	2.74	2.74	2.86	2.57	2.57	2.69
	30 at ≤ 50%F	2.55	2.55	2.69	2.34	2.34	2.48
	25 at ≤ 50%F	2.66	2.66	2.76	2.46	2.46	2.58
TBVOOS	100	1.58	1.58	1.60	1.48	1.49	1.50
	75	1.70	1.70	1.73	1.60	1.60	1.63
	65	1.76	1.76	1.81	1.66	1.66	1.70
	50	1.94	1.94	---	1.80	1.80	---
	50	1.98	1.98	2.03	1.82	1.82	1.87
	40	2.17	2.17	2.29	2.03	2.03	2.13
	30	2.55	2.55	2.69	2.34	2.34	2.48
	30 at > 50%F	3.06	3.06	3.17	3.02	3.02	3.13
	25 at > 50%F	3.44	3.44	3.53	3.23	3.23	3.32
	30 at ≤ 50%F	2.86	2.86	2.98	2.61	2.61	2.73
	25 at ≤ 50%F	3.26	3.26	3.41	3.00	3.00	3.15
FHOOS	100	1.54	1.54	---	1.47	1.47	---
	75	1.68	1.68	---	1.60	1.60	---
	65	1.78	1.78	---	1.67	1.67	---
	50	---	---	---	---	---	---
	50	2.01	2.01	---	1.86	1.86	---
	40	2.29	2.29	---	2.13	2.13	---
	30	2.69	2.69	---	2.48	2.48	---
	30 at > 50%F	2.69	2.69	---	2.65	2.65	---
	25 at > 50%F	2.86	2.86	---	2.69	2.69	---
	30 at ≤ 50%F	2.69	2.69	---	2.48	2.48	---
	25 at ≤ 50%F	2.76	2.76	---	2.58	2.58	---
PLUOOS	100	1.52	1.52	1.54	1.44	1.45	1.47
	75	1.64	1.64	1.68	1.56	1.56	1.60
	65	1.88	1.88	1.88	1.73	1.76	1.76
	50	---	---	---	---	---	---
	50	1.98	1.98	2.01	1.82	1.82	1.86
	40	2.17	2.17	2.29	2.03	2.03	2.13
	30	2.55	2.55	2.69	2.34	2.34	2.48
	30 at > 50%F	2.55	2.55	2.69	2.48	2.48	2.65
	25 at > 50%F	2.74	2.74	2.86	2.57	2.57	2.69
	30 at ≤ 50%F	2.55	2.55	2.69	2.34	2.34	2.48
	25 at ≤ 50%F	2.66	2.66	2.76	2.46	2.46	2.58

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.4 $MCPR_P$ Limits for All Fuel Types: Technical Specification Scram Time Basis (*continued*)*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
TBVOOS FHOOS	100	1.60	1.60	---	1.50	1.50	---
	75	1.73	1.73	---	1.63	1.63	---
	65	1.81	1.81	---	1.70	1.70	---
	50	---	---	---	---	---	---
	50	2.03	2.03	---	1.87	1.87	---
	40	2.29	2.29	---	2.13	2.13	---
	30	2.69	2.69	---	2.48	2.48	---
	30 at > 50%F	3.17	3.17	---	3.13	3.13	---
	25 at > 50%F	3.53	3.53	---	3.32	3.32	---
	30 at ≤ 50%F	2.98	2.98	---	2.73	2.73	---
	25 at ≤ 50%F	3.41	3.41	---	3.15	3.15	---
TBVOOS PLUOOS	100	1.58	1.58	1.60	1.48	1.49	1.50
	75	1.70	1.70	1.73	1.60	1.60	1.63
	65	1.88	1.88	1.88	1.73	1.76	1.76
	50	---	---	---	---	---	---
	50	1.98	1.98	2.03	1.82	1.82	1.87
	40	2.17	2.17	2.29	2.03	2.03	2.13
	30	2.55	2.55	2.69	2.34	2.34	2.48
	30 at > 50%F	3.06	3.06	3.17	3.02	3.02	3.13
	25 at > 50%F	3.44	3.44	3.53	3.23	3.23	3.32
	30 at ≤ 50%F	2.86	2.86	2.98	2.61	2.61	2.73
	25 at ≤ 50%F	3.26	3.26	3.41	3.00	3.00	3.15
FHOOS PLUOOS	100	1.54	1.54	---	1.47	1.47	---
	75	1.68	1.68	---	1.60	1.60	---
	65	1.88	1.88	---	1.73	1.76	---
	50	---	---	---	---	---	---
	50	2.01	2.01	---	1.86	1.86	---
	40	2.29	2.29	---	2.13	2.13	---
	30	2.69	2.69	---	2.48	2.48	---
	30 at > 50%F	2.69	2.69	---	2.65	2.65	---
	25 at > 50%F	2.86	2.86	---	2.69	2.69	---
	30 at ≤ 50%F	2.69	2.69	---	2.48	2.48	---
	25 at ≤ 50%F	2.76	2.76	---	2.58	2.58	---
TBVOOS FHOOS PLUOOS	100	1.60	1.60	---	1.50	1.50	---
	75	1.73	1.73	---	1.63	1.63	---
	65	1.88	1.88	---	1.73	1.76	---
	50	---	---	---	---	---	---
	50	2.03	2.03	---	1.87	1.87	---
	40	2.29	2.29	---	2.13	2.13	---
	30	2.69	2.69	---	2.48	2.48	---
	30 at > 50%F	3.17	3.17	---	3.13	3.13	---
	25 at > 50%F	3.53	3.53	---	3.32	3.32	---
	30 at ≤ 50%F	2.98	2.98	---	2.73	2.73	---
	25 at ≤ 50%F	3.41	3.41	---	3.15	3.15	---

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop. Consequently, FHOOS limits are not provided for BOC to End of COAST due to redundancy. Thermal limits for the "BOC to End of COAST" exposure applicability window are developed to conservatively bound FHOOS limits for earlier exposure applicability windows.

Table 4.5 Startup Operation MCPRp Limits for Table 3.1 Temperature Range 1 for All Fuel Types: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
TBVIS	100	1.52	1.53	1.53	1.45	1.46	1.46
	75	1.66	1.66	1.66	1.60	1.60	1.60
	65	1.86	1.86	1.86	1.72	1.74	1.74
	50	---	---	---	---	---	---
	50	2.18	2.18	2.18	2.03	2.03	2.03
	40	2.51	2.51	2.51	2.36	2.36	2.36
	30	2.97	2.97	2.97	2.78	2.78	2.78
	30 at > 50%F	2.97	2.97	2.97	2.87	2.87	2.87
	25 at > 50%F	3.24	3.24	3.24	2.95	2.95	2.95
	30 at ≤ 50%F	2.97	2.97	2.97	2.78	2.78	2.78
	25 at ≤ 50%F	3.12	3.12	3.12	2.86	2.86	2.86
TBVOOS	100	1.57	1.58	1.58	1.48	1.49	1.49
	75	1.72	1.72	1.72	1.63	1.63	1.63
	65	1.86	1.86	1.86	1.72	1.74	1.74
	50	---	---	---	---	---	---
	50	2.18	2.18	2.18	2.03	2.03	2.03
	40	2.51	2.51	2.51	2.36	2.36	2.36
	30	2.97	2.97	2.97	2.78	2.78	2.78
	30 at > 50%F	3.34	3.34	3.34	3.33	3.33	3.33
	25 at > 50%F	3.80	3.80	3.80	3.55	3.55	3.55
	30 at ≤ 50%F	3.18	3.18	3.18	2.94	2.94	2.94
	25 at ≤ 50%F	3.64	3.64	3.64	3.39	3.39	3.39

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.6 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 2 for All Fuel Types: Nominal Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
TBVIS	100	1.52	1.53	1.53	1.45	1.46	1.46
	75	1.66	1.66	1.66	1.60	1.60	1.60
	65	1.86	1.86	1.86	1.72	1.74	1.74
	50	---	---	---	---	---	---
	50	2.19	2.19	2.19	2.04	2.04	2.04
	40	2.52	2.52	2.52	2.37	2.37	2.37
	30	2.99	2.99	2.99	2.80	2.80	2.80
	30 at > 50%F	2.99	2.99	2.99	2.88	2.88	2.88
	25 at > 50%F	3.27	3.27	3.27	2.97	2.97	2.97
	30 at ≤ 50%F	2.99	2.99	2.99	2.80	2.80	2.80
	25 at ≤ 50%F	3.15	3.15	3.15	2.87	2.87	2.87
TBVOOS	100	1.57	1.58	1.58	1.48	1.49	1.49
	75	1.72	1.72	1.72	1.63	1.63	1.63
	65	1.86	1.86	1.86	1.72	1.74	1.74
	50	---	---	---	---	---	---
	50	2.19	2.19	2.19	2.04	2.04	2.04
	40	2.52	2.52	2.52	2.37	2.37	2.37
	30	2.99	2.99	2.99	2.80	2.80	2.80
	30 at > 50%F	3.35	3.35	3.35	3.34	3.34	3.34
	25 at > 50%F	3.81	3.81	3.81	3.57	3.57	3.57
	30 at ≤ 50%F	3.19	3.19	3.19	2.95	2.95	2.95
	25 at ≤ 50%F	3.65	3.65	3.65	3.41	3.41	3.41

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.7 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 1 for All Fuel Types: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
TBVIS	100	1.54	1.54	1.54	1.47	1.47	1.47
	75	1.68	1.68	1.68	1.60	1.60	1.60
	65	1.88	1.88	1.88	1.73	1.76	1.76
	50	---	---	---	---	---	---
	50	2.23	2.23	2.23	2.06	2.06	2.06
	40	2.56	2.56	2.56	2.39	2.39	2.39
	30	3.04	3.04	3.04	2.81	2.81	2.81
	30 at > 50%F	3.04	3.04	3.04	2.87	2.87	2.87
	25 at > 50%F	3.24	3.24	3.24	2.95	2.95	2.95
	30 at ≤ 50%F	3.04	3.04	3.04	2.81	2.81	2.81
	25 at ≤ 50%F	3.12	3.12	3.12	2.86	2.86	2.86
TBVOOS	100	1.60	1.60	1.60	1.50	1.50	1.50
	75	1.73	1.73	1.73	1.63	1.63	1.63
	65	1.88	1.88	1.88	1.73	1.76	1.76
	50	---	---	---	---	---	---
	50	2.23	2.23	2.23	2.06	2.06	2.06
	40	2.56	2.56	2.56	2.39	2.39	2.39
	30	3.04	3.04	3.04	2.81	2.81	2.81
	30 at > 50%F	3.34	3.34	3.34	3.33	3.33	3.33
	25 at > 50%F	3.80	3.80	3.80	3.55	3.55	3.55
	30 at ≤ 50%F	3.18	3.18	3.18	2.94	2.94	2.94
	25 at ≤ 50%F	3.64	3.64	3.64	3.39	3.39	3.39

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.8 Startup Operation $MCPR_P$ Limits for Table 3.1 Temperature Range 2 for All Fuel Types: Technical Specification Scram Time Basis*

Operating Condition	Power (% of rated)	ATRIUM-10 BOC to NEOC	ATRIUM-10 BOC to EOCLB	ATRIUM-10 BOC to End of Coast	ATRIUM-10XM BOC to NEOC	ATRIUM-10XM BOC to EOCLB	ATRIUM-10XM BOC to End of Coast
TBVIS	100	1.54	1.54	1.54	1.47	1.47	1.47
	75	1.68	1.68	1.68	1.60	1.60	1.60
	65	1.88	1.88	1.88	1.73	1.76	1.76
	50	---	---	---	---	---	---
	50	2.24	2.24	2.24	2.07	2.07	2.07
	40	2.57	2.57	2.57	2.40	2.40	2.40
	30	3.06	3.06	3.06	2.83	2.83	2.83
	30 at > 50%F	3.06	3.06	3.06	2.88	2.88	2.88
	25 at > 50%F	3.27	3.27	3.27	2.97	2.97	2.97
	30 at ≤ 50%F	3.06	3.06	3.06	2.83	2.83	2.83
	25 at ≤ 50%F	3.15	3.15	3.15	2.87	2.87	2.87
TBVOOS	100	1.60	1.60	1.60	1.50	1.50	1.50
	75	1.73	1.73	1.73	1.63	1.63	1.63
	65	1.88	1.88	1.88	1.73	1.76	1.76
	50	---	---	---	---	---	---
	50	2.24	2.24	2.24	2.07	2.07	2.07
	40	2.57	2.57	2.57	2.40	2.40	2.40
	30	3.06	3.06	3.06	2.83	2.83	2.83
	30 at > 50%F	3.35	3.35	3.35	3.34	3.34	3.34
	25 at > 50%F	3.81	3.81	3.81	3.57	3.57	3.57
	30 at ≤ 50%F	3.19	3.19	3.19	2.95	2.95	2.95
	25 at ≤ 50%F	3.65	3.65	3.65	3.41	3.41	3.41

* Limits support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

Limits are applicable for all other EOOS scenarios, apart from TBV.

Limits are only valid up to 50% rated core power.

Table 4.9 MCPR$_P$ Limits for All Fuel Types: Single Loop Operation for All Scram Times*

Operating Condition	Power (% of rated)	BOC to End of COAST	
		ATRIUM-10	ATRIUM-10XM
RCPOOS FHOOS	100	2.03	2.04
	50	2.03	2.04
	40	2.31	2.15
	30	2.71	2.50
	30 at > 50%F	2.71	2.67
	25 at > 50%F	2.88	2.71
	30 at ≤ 50%F	2.71	2.50
	25 at ≤ 50%F	2.78	2.60
RCPOOS TBVOOS PLUOOS FHOOS	100	2.05	2.04
	50	2.05	2.04
	40	2.31	2.15
	30	2.71	2.50
	30 at > 50%F	3.19	3.15
	25 at > 50%F	3.55	3.34
	30 at ≤ 50%F	3.00	2.75
	25 at ≤ 50%F	3.43	3.17
RCPOOS TBVOOS FHOOS1	100	2.25	2.08
	50	2.25	2.08
	40	2.58	2.41
	30	3.06	2.83
	30 at > 50%F	3.36	3.35
	25 at > 50%F	3.82	3.57
	30 at ≤ 50%F	3.20	2.96
	25 at ≤ 50%F	3.66	3.41
RCPOOS TBVOOS FHOOS2	100	2.26	2.09
	50	2.26	2.09
	40	2.59	2.42
	30	3.08	2.85
	30 at > 50%F	3.37	3.36
	25 at > 50%F	3.83	3.59
	30 at ≤ 50%F	3.21	2.97
	25 at ≤ 50%F	3.67	3.43

* All limits, including "Base Case," support RPTOOS operation; operation is supported for any combination of 1 MSRVOOS, up to 2 TIPOOS (or the equivalent number of TIP channels), and up to 50% of the LPRMs out-of-service.

FFWTR and FHOOS assume the same value of temperature drop.

RCPOOS limits are only valid up to 50% rated core power, 50% rated core flow, and an active recirculation drive flow of 17.73 Mlbm/hr.

5 Oscillation Power Range Monitor (OPRM) Setpoint

(Technical Specification 3.3.1.1)

Technical Specification Table 3.3.1.1-1, Function 2f, identifies the OPRM upscale function.

Instrument setpoints are established, such that the reactor will be tripped before an oscillation can grow to the point where the SLMCPR is exceeded. An Option III stability analysis is performed for each reload core to determine allowable OLMCPR's as a function of OPRM setpoint. Analyses consider both steady state startup operation, and the case of a two recirculation pump trip from rated power.

The resulting stability based OLMCPR's are reported in Reference 1. The OPRM setpoint (*sometimes referred to as the Amplitude Trip, S_p*) is selected, such that required margin to the SLMCPR is provided without stability being a limiting event. Analyses are based on cycle specific DIVOM analyses performed per Reference 22. The calculated OLMCPR's are shown in Table 5.1. Review of results shown in COLR Table 4.2 indicates an OPRM setpoint of **1.14** may be used. The successive confirmation count (*sometimes referred to as N_p*) is provided in Table 5.2, per Reference 30.

Table 5.1 OPRM Setpoint Range*

OPRM Setpoint	OLMCPR (SS)	OLMCPR (2PT)
1.05	1.19	1.16
1.06	1.20	1.17
1.07	1.20	1.18
1.08	1.21	1.19
1.09	1.22	1.20
1.10	1.24	1.22
1.11	1.26	1.24
1.12	1.28	1.26
1.13	1.30	1.28
1.14	1.33	1.30
1.15	1.35	1.32

Table 5.2 OPRM Successive Confirmation Count Setpoint

Count	OPRM Setpoint
6	≥ 1.04
8	≥ 1.05
10	≥ 1.07
12	≥ 1.09
14	≥ 1.11
16	≥ 1.14
18	≥ 1.18
20	≥ 1.24

* Extrapolation beyond a setpoint of 1.15 is not allowed

6 APRM Flow Biased Rod Block Trip Settings

(Technical Requirements Manual Section 5.3.1 and Table 3.3.4-1)

The APRM rod block trip setting is based upon References 26 & 27, and is defined by the following:

$SRB \leq (0.66(W-\Delta W) + 61\%)$	Allowable Value
$SRB \leq (0.66(W-\Delta W) + 59\%)$	Nominal Trip Setpoint (NTSP)

where:

SRB	=	Rod Block setting in percent of rated thermal power (3458 MW_t)
W	=	Loop recirculation flow rate in percent of rated
ΔW	=	Difference between two-loop and single-loop effective recirculation flow at the same core flow (ΔW=0.0 for two-loop operation)

The APRM rod block trip setting is clamped at a maximum allowable value of 115% (corresponding to a NTSP of 113%).

7 Rod Block Monitor (RBM) Trip Setpoints and Operability

(Technical Specification Table 3.3.2.1-1)

The RBM trip setpoints and applicable power ranges, based on References 26 & 27, are shown in Table 7.1. Setpoints are based on an HTSP, unfiltered analytical limit of 114%. Unfiltered setpoints are consistent with a nominal RBM filter setting of 0.0 seconds; filtered setpoints are consistent with a nominal RBM filter setting less than 0.5 seconds. Cycle specific CRWE analyses of OLMCPR are documented in Reference 1, superseding values reported in References 26, 27, and 29.

Table 7.1 Analytical RBM Trip Setpoints*

RBM Trip Setpoint	Allowable Value (AV)	Nominal Trip Setpoint (NTSP)
LPSP	27%	25%
IPSP	62%	60%
HPSP	82%	80%
LTSP - unfiltered - filtered	121.7% 120.7%	120.0% 119.0%
ITSP - unfiltered - filtered	116.7% 115.7%	115.0% 114.0%
HTSP - unfiltered - filtered	111.7% 110.9%	110.0% 109.2%
DTSP	90%	92%

As a result of cycle specific CRWE analyses, RBM setpoints in Technical Specification Table 3.3.2.1-1 are applicable as shown in Table 7.2. Cycle specific setpoint analysis results are shown in Table 7.3, per Reference 1.

Table 7.2 RBM Setpoint Applicability

Thermal Power (% Rated)	Applicable MCPR†	Notes from Table 3.3.2.1-1	Comment
> 27% and < 90%	< 1.74	(a), (b), (f), (h)	two loop operation
	< 1.78	(a), (b), (f), (h)	single loop operation
≥ 90%	< 1.41	(g)	two loop operation‡

* Values are considered maximums. Using lower values, due to RBM system hardware/software limitations, is conservative, and acceptable.

† MCPR values shown correspond with, (support), SLMPCR values identified in Reference 1.

‡ Greater than 90% rated power is not attainable in single loop operation.

Table 7.3 Control Rod Withdrawal Error Results

RBM HTSP Analytical Limit	CRWE OLMCPR
Unfiltered	
107	1.30
111	1.33
114	1.36
117	1.38

Results, compared against the base case OLMCPR results of Table 4.2, indicate SLMCPR remains protected for RBM inoperable conditions (i.e., 114% unblocked).

8 Shutdown Margin Limit

(Technical Specification 3.1.1)

Assuming the strongest OPERABLE control blade is fully withdrawn, and all other OPERABLE control blades are fully inserted, the core shall be sub-critical and meet the following minimum shutdown margin:

SDM > 0.38% dk/k

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B 3.0-2	0	Initial
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B 3.0-8a	Revision 111	10-10-2014
B 3.0-9	0	Initial
B 3.0-10	Revision 102	10-24-2013
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ACTIONS (continued)

C.1 and C.2

When required control room indication channels are inoperable but the redundant channels for the parameters are still OPERABLE, the required control room indication channels must be returned to OPERABLE status in 30 days (Required Action C.2). However, if both redundant channels for one or more of the associated parameters are not indicating in the control room, the 30 day allowed out of service time is not acceptable and one of the required control room indication channels for each associated parameter must be returned to OPERABLE status in 7 days (Required Action C.1).

D.1, D.2, and D.3

When the Tailpipe Thermocouple Temperature and Acoustic Monitor is inoperable for one or more Main Steam Relief Valves (MSRVs), the torus temperature must be observed once per 12 hours to observe any unexplained temperature increase which might be indicative of an open relief valve (Required Action D.1) and control room indication by either the Tailpipe Thermocouple Temperature or Acoustic Monitor must be returned to OPERABLE status for each relief valve in 30 days (Required Action D.2). The condition must be entered into the Corrective Action Program within 24 hours if control room indication is not restored in 30 days (Required Action D.3).

E.1.1 and E.1.2

When the Wide Range Gaseous Effluent Radiation Monitor and Recorder instrument channel is inoperable, either the inoperable channel must be returned to OPERABLE status in 72 hours (Required Action E.1.1), or the preplanned alternate method of monitoring the parameter must be initiated (Required Action E.1.2). A note is provided to indicate that Required Actions E.1.1 and E.1.2 are not applicable when in MODES 4 and 5.

E.2

The condition must be entered into the Corrective Action Program within 24 hours after the Wide Range Gaseous Effluent Radiation Monitor and Recorder instrument channel has been inoperable for 7 days.

BASES

LCO 3.3.7	The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation dose to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public.
APPLICABILITY	At all times.
ACTIONS	A.1 Seven days to obtain replacement parts and repair is reasonable for these instruments given the requirements to be available for radiological emergencies. B.1 If the instruments cannot be repaired in the allowed time frame, the condition must be entered into the Corrective Action Program within 24 hours.
TECHNICAL SURVEILLANCE REQUIREMENTS	TSR 3.3.7.1 Daily checks of these parameters assures prompt replacement/repair of inoperable or questionable instruments. TSR 3.3.7.2 Surveillance requirement times are based on equipment reliability and engineering judgment and conservatively set to provide adequate assurance of instrument performance.

ACTIONS (continued)

B.1 (continued)

conditions at the locations of these instruments to determine the impact of the vibratory ground motion on structures and equipment in these locations following any required unit shutdown after a seismic event.

C.1

If any Required Action and associated Completion Time of Condition A or B is not met, the failure to restore the inoperable seismic monitoring instrumentation within the required Completion Time must be entered into the Corrective Action Program.

TECHNICAL SURVEILLANCE REQUIREMENTS

The TSRs for each seismic monitoring Function are identified by the Technical Surveillance Requirements column of Table 3.3.8-1.

TSR 3.3.8.1

Performance of a CHANNEL CHECK on the seismic monitoring instrumentation once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a check of external system status indications that the seismic monitoring equipment is in a state of readiness to properly function should an earthquake occur. A CHANNEL CHECK will detect gross system failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL FUNCTIONAL TEST. The Surveillance Frequency of 31 days is based on operating experience related to instrumentation systems, which demonstrates that gross instrumentation system failure in any 31-day interval is a rare event.

TSR 3.3.8.2

A CHANNEL FUNCTIONAL TEST is to be performed on each required channel to ensure the entire channel will perform the intended function. A CHANNEL FUNCTIONAL TEST is the comparison of the response of the instrumentation, including all

BASES

LCO 3.3.11

The primary containment hydrogen monitoring instrumentation allows the operators to detect trends in hydrogen concentration to diagnose the course of beyond design basis accidents. High hydrogen concentration is measured, continuously recorded, and displayed in the control room by a single instrument channel. The analyzer has the capability for sampling both the drywell and the suppression chamber. LCO 3.3.11 requires the primary containment hydrogen analyzer to be OPERABLE.

APPLICABILITY

The primary containment hydrogen analyzer is required to be OPERABLE when primary containment is inerted, except as allowed by the relaxations during startup and shutdown addressed below. The primary containment must be inert in MODE 1, since this is the condition with the highest probability of an event that could produce hydrogen.

Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and de-inerted as soon as possible in the plant shutdown. As long as reactor power is < 15% RTP, the potential for an event that generates significant hydrogen is low and the primary containment need not be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a startup, or within the last 24 hours before a shutdown, is low enough that these "windows," when the primary containment is not inerted, are also justified. The 24 hour time period is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.

ACTIONS

A.1

Seven days to restore the instrument is reasonable given the requirements to be available for use in diagnosing beyond design basis events.

TR 3.7 PLANT SYSTEMS

TR 3.7.4 Snubbers

BASES

BACKGROUND

Snubbers are designed to prevent unrestrained component or system motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to the component or system as a result of a seismic or other event initiating dynamic loads. An inoperable snubber (ex: failed by locked in place) may cause damage to the supported component or system from normal operating modes such as thermal operation. It is, therefore, required that all snubbers required to protect the primary coolant system or any other safety related component or system be OPERABLE during MODES 1, 2, 3, 4, and 5. The Technical Requirements Manual (TRM) action statements establish allowable outage times for components or systems addressed by the Limiting Conditions of Operation (LCO) for snubbers. These time limits are applicable when a snubber must be removed from service to perform required surveillance tests. For snubbers, the allowable outage time is 72 hours. Table 3.7.4-1, "Visual Examination Table" is published in ASME OM Code Subsection ISTD, Table ISTD-4252-1, and is based on previous table issued to all nuclear plant license holders by the Nuclear Regulatory Commission (NRC) under Generic Letter (GL) 90-09. This was added to the old Technical Specification and approved by the NRC under Technical Specification Amendment 183.

APPLICABLE SAFETY ANALYSIS

During MODES 1, 2, 3, 4, and 5 snubbers may be removed from service for functional surveillance testing to satisfy the required testing interval. When a snubber is removed from a component or system, the snubber is declared inoperable since it cannot perform its intended function while removed. This type of inoperability is not a failure. Examples of snubber failures include locked in place, high drag force, does not activate, no lockup, high lockup, low lockup, high bleed, no bleed, and damage to the snubber hardware. If a snubber is determined to be inoperable based on failure to meet the functional test acceptance criteria, an engineering evaluation is performed to establish whether, during

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS

A note is provided to define the term "code snubber," as it is used in the Technical Requirements, and it shall mean snubbers that are identified by BFN ASME Code IST program as ASME Class 1, 2, or 3 equivalent, snubbers. It shall also mean those BFN safety-related snubbers that are not identified as ASME Class 1, 2, or 3 equivalent, but are treated as such.

A note is provided to indicate that each code snubber, identified by those snubbers listed in plant procedures, shall be demonstrated OPERABLE by performance of the following inservice examination and test program requirements, which is derived from ASME OM Code Subsection ISTD.

An additional note is provided to indicate that in this Technical Requirement, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

The augmented inservice inspection program includes the following.

All code snubbers are visually inspected, pin to pin, inclusive, for overall integrity and OPERABILITY. The visual inspection will include verification that no visible indications of damage, leakage, corrosion, degradation, binding, misalignment, deformation, or other external characteristics that may indicate impaired OPERABILITY are present, verification that proper attachment of the snubber to the component or system and structures exist, and that no loose or missing fasteners exist. In addition, hydraulic fluid level is verified. The removal of insulation or the verification of torque values for threaded fasteners is not required for visual inspections.

The visual inspection frequency is based upon maintaining a constant level of snubber protection. In accordance with Table 3.7.4-1, the number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25 percent) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

BASES (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS	When the cause of the rejection of a snubber in a visual inspection is clearly established and remedied for that snubber and for any other snubber(s) that may be generically susceptible and OPERABILITY verified by inservice functional testing, if applicable, that snubber(s) may be reclassified as OPERABLE. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to the rejected snubber, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

vibration. The inspection population or category may be established based on design features, or installed conditions which may be expected to be generic. Each of these inspection populations or categories may be inspected and tested separately unless an engineering analysis indicates the inspection population or category is improperly constituted. All suspect snubbers are subject to inspection and testing regardless of inspection population or category.

To verify snubber OPERABILITY, a functional test shall be performed once per fuel cycle.

These tests will include stroking of the snubbers to verify proper movement, activation, and bleed or release. Ten percent represents an adequate sample for such tests. Observed failures on these samples will require a failure analysis and testing of additional units. If the failure analysis results in the determination that the failure of a snubber to activate or to stroke (i.e., seized components) is the result of a manufacture or design deficiency, all snubbers subject to the same defect shall be functionally tested. Also, an engineering evaluation shall be performed to determine the effects on the supported component or system during the previous unit operating cycle with the snubber inoperable, and to ensure it remains capable of meeting its designed service. A thorough visual inspection of the snubber threaded attachments to the component or system and the anchorage will be made in conjunction with all required functional tests. The stroke setting of the snubbers selected for functional testing also will be verified.

Exemption from Visual Inspection or Functional Tests:

Permanent or other exemptions from visual inspections and/or functional testing for individual snubbers may be granted by the Nuclear Regulatory Commission if a justifiable basis for exemption is presented and if applicable snubber life destructive testing was performed to qualify the snubber OPERABILITY for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall continue to be listed in the plant instructions with footnotes indicating the extent of the exemptions.

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

Snubber Service Life Program:

The service life of snubbers may be extended based on an evaluation of the records of functional tests, maintenance history, and environmental conditions to which the snubbers have been exposed.

The following will be implemented by the augmented inservice inspection program:

TSR 3.7.4.1

Visual Inspections:

At BFN, code snubbers are visually examined as one population or category, regardless of accessibility, in accordance with the schedule and interval determined by Table 3.7.4-1, which is Table ISTD-4252-1. The first inspection interval determined using Table 3.7.4-1 criteria shall be based on the previous inspection interval as established by the requirements in effect before Revision 007 of these Technical Requirements were issued.

Visual Inspection Acceptance Criteria:

Visual inspections shall verify that.

a) The snubber has no visible indications of damage, leakage, corrosion, degradation, or other external characteristics that may indicate impaired OPERABILITY, pin to pin, inclusive.

b) Fasteners for the attachment of the snubber are function.

c) No indications of binding, misalignment, or deformation of the snubber.

d) Hydraulic snubber fluid is at the recommended level and vented reservoir is oriented such that fluid can gravitate to snubber body.

e) The absence of weld arc strikes, paint, weld slag, adhesive, or other deposits on piston rod or support cylinder that could result in unacceptable snubber performance.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.4.1 (continued)

f) Snubber spherical bearing is fully engaged in attachment lug.

g) Snubber position setting is adequate.

Snubbers which appear inoperable as a result of visual inspection shall be classified unacceptable. Snubbers confirmed as unacceptable snubbers are adjusted, repaired, modified, or replaced, and counted in determination of the subsequent examination interval in accordance with Table 3.7.4-1, regardless if the affected snubber is functionally tested in the as-found condition and determined OPERABLE per the functional test acceptance criteria of TSR 3.7.4.2.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.4.1 (continued)

A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable.

Additionally, snubbers attached to sections of safety-related systems that have experienced unexpected potentially damaging transients since the last inspection period shall be evaluated for the possibility of concealed damage and functionally tested, if applicable, to confirm OPERABILITY. Snubbers which have been made inoperable as the result of unexpected transients, isolated damage, or other random events, when the provisions of TSR 3.7.4.5 and 3.7.4.6 have been met and any other appropriate corrective action implemented, shall not be counted in determining the next visual inspection interval.

TSR 3.7.4.2

Functional Test Schedule, Lot Size, and Composition:

Once per fuel cycle, a representative sample of 10% of the total of each type of code snubbers in use in the plant shall be functionally tested either in place or in a bench test. The sample population is rounded up to the next whole integer. Safety-related snubbers that are not ASME Class 1, 2, or 3 equivalent snubbers shall not be included in the snubber population when selecting the initial or additional samples.

As practical, the representative sample selected for functional testing shall include representation from the Defined Test Plan Group (DTPG) based on the significant features (i.e., the various designs, configurations, operating environments, and the range of size and capacity of snubbers within the types) and based on the ratio of the number of snubbers of each significant feature, to the total number of snubbers in the DTPG. The sample shall be generally representative as specified in ISTD-5311(a), but may also be selected from snubbers concurrently scheduled for seal replacement or other similar activity related to service life monitoring. The snubbers shall be tested on a generally rotational basis to coincide with the service life monitoring.

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.4.2 (continued)

The representative sample should be weighed to include more snubbers from severe service areas such as near heavy equipment. After testing, at the time of reinstallation, the snubber shall meet visual examination attributes described in ISTD-4110(a), -4110(c), -4110(d), and -4110(e). The stroke setting shall be verified.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.4.2 (continued)

Functional Test Acceptance Criteria:

The snubber functional test shall verify that:

a. Activation (restraining action) is achieved in both tension and compression within the specified range of velocity or acceleration.

b. Snubber bleed, or release where required, is present in both compression and tension within the specified range.

c. For mechanical snubbers, the drag force is within the specified limits, in tension and compression.

d. For hydraulic snubbers, if required to verify proper assembly, drag force is within specific limits in tension and compression.

e. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

f. Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

TSR 3.7.4.3

Functional Test Failure Analysis and Additional Test Lots:

A failure analysis shall be performed for each failure to meet the functional test acceptance criteria to determine the cause of the failure. The result of this analysis shall be used, if applicable, in selecting snubbers to be tested in the subsequent lot in an effort to determine the OPERABILITY of other snubbers which may be subject to the same failure mode. Selection of snubbers for future testing may also be based on the failure analysis.

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.7.4.3 (continued)

For each snubber that does not meet the functional test acceptance criteria, an additional lot equal to 5 percent of the subject snubber DTPG shall be functionally tested. When establishing additional sample testing, failure analysis results of unacceptable snubbers are to be used to determine if establishing an FMG is appropriate. Additional lot (i.e., sample) population is rounded up to next whole integer. Rounding up satisfies ISTD-5312 requirement that additional samples shall be at least one-half the size of the initial sample from the DTPG. Additional samples selected from a DTPG follow the same composition as the original sample. However, additional samples from FMGs are random samples. Testing shall continue until no additional inoperable snubbers are found within subsequent lots or all snubbers of the original functional test type have been tested or all suspect snubbers identified by the failure analysis have been tested, as applicable.

The discovery of loose or missing attachment fasteners will be evaluated to determine whether the cause may be localized or generic. The result of the evaluation will be used to select other suspect snubbers for verifying the attachment fasteners, as applicable.

TSR 3.7.4.4

(Deleted by TRM Revision 125.)

TSR 3.7.4.5

Functional Test Failure - Supported Component or System Analysis:

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the component or system which is restrained by the snubber(s) due to not meeting their functional test acceptance criteria. The purpose of this engineering evaluation shall be to determine if the component or system restrained by the snubber(s) was adversely affected by the inoperability of the snubber(s), and in order to ensure that the restrained component or system remains capable of meeting the designed service.

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

TSR 3.7.4.6

Functional Testing of Repaired and Spare Snubbers:

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be adjusted, repaired, modified, scrapped, or replaced. Replacement snubbers and snubbers having repairs, adjustments, or modifications which might affect the functional test results shall meet the functional test acceptance criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the functional test must have been performed within 12 months prior to being installed in the unit.

REFERENCES

1. BFN Technical Specifications (version prior to standardized version)

ENCLOSURE 4

Tennessee Valley Authority
Browns Ferry Nuclear Plant
Units 1, 2, and 3

Summary of Revised Commitments

Commitment Tracking Number

111492581
112906273
114569185
114569179
114569214
114569374
114574578
114574790
114574882
114690044
114690158
114698306
114698399
114698542
114703120
114703145
114703194
114703257
114703304
114712500
114712572
114712681
114712713
114772054
114772326
114774135
114776636
114782716
114782752
114785981
114790295
115295456
116201617
116201648
116201658
116201665
116201669
116201672
116633768

COMMITMENT CHANGE EVALUATION FORM (CCEF)

Commitment Tracking Database (CTD) Number: 111492581

Source Document: Letter from TVA to NRC - R08100518051 Date: 05/18/2010

Existing Commitment Description: (*Attach additional page, as necessary*)

Full compliance for this violation (2010-007-001) will be achieved upon completing implementation of the NFPA 805 approach to compliance with fire protection requirements. Full implementation of NFPA 805 will not be completed until at least 2016.

Revised Commitment Description:

Full compliance for this violation (2010-007-001) will be achieved upon completing implementation of the NFPA 805 approach to compliance with fire protection requirements. Full implementation of NFPA 805, including modifications, will not be completed until at least March 31, 2019.

Summarize Justification for Revising Commitment:

This is a change in schedule. The original due date was based on a NFPA 805 submittal of March 4, 2012. The submittal date was subsequently delayed until March 29, 2013 (13 months). NFPA 805 Programatic implementation is required within 240 days after receipt of the NRC SER, received on October 28, 2015. Full compliance with NFPA 805 will be achieved when all modifications are completed, which per the SER must take place by the end of the 2nd refueling outageof the final unit and shared systems being modified (U2 R20). Proposed change to the scheduled due date is March 31, 2019.

(*Refer to Attachments 3 (flow diagram) and 4 (description) for an explanation of the commitment change evaluation process.*)

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-09.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: This is a change in schedule only and does not revise any implementation item previously required.

(Attach additional page, as necessary)

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional page, as necessary)

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in CTD. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in CTD. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in CTD. Notify NRC of revised commitment in next scheduled summary report.

☐ No Revise commitment in CTD. No NRC notification required.

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Supervisor. List any documents (e.g., procedures, NRC submittals) affected by this change below: (*Attach additional page, as necessary*)

Description	**EDMS #**
None	

Prepared by: Baruch Calkin Date: 3/25/2016

APPROVALS

Sign/Print: Jamie Paul / Jamie Paul Date: 3/25/16
Responsible Supervisor

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Once approved by Licensing, this form shall be kept for the life of the plant in the EDMS vault and attached to CTD.
EDMS # ____________ CCEF attached to CTD and emailed to RS: Y / N

Sign/Print: BARUCH CALKIN Date: 3/25/16
Lic Eng, Lic Prog Mgr, Corporate or Site Lic Mgr

COMMITMENT CHANGE EVALUATION FORM (CCEF)

Commitment Tracking Database (CTD) Number: 112906273 Commitment Change for Temporary Diesel Generators(TDGs)

Source Document: (112906273), Response to NRC Request for Additional Information Regarding Extending Allowed Outage Times for Technical Specification 3.8.1 (TAC Nos. ME5036, ME5037, and ME5038) L44411060 1004 as amended by Commitment Change Evaluation Form (CCEF) dated 5/27/2011

Date: 5/31/16

Existing Commitment Description: (*Attach additional page, as necessary*)

See Attached

Revised Commitment Description:

See Attached

Summarize Justification for Revising Commitment:

See Attached

See Attached FPPCE

(*Refer to Attachments 3 (flow diagram) and 4 (description) for an explanation of the commitment change evaluation process.*)

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-09.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: The revised commitment has no adverse impact on the conclusions of the NFPA 805 analyses. This commitment involves changes compensatory measures put in place prior to the implementation of the NFPA 805 FPP.

(Attach additional page, as necessary)

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional page, as necessary)

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☒ Yes Go to 4.2.

☐ No Go to Part V.

4.2 Has the original commitment been implemented?

☒ Yes STOP. You have completed this evaluation. Revise commitment in CTD. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☐ No STOP. You have completed this evaluation. Revise commitment in CTD. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in CTD. Notify NRC of revised commitment in next scheduled summary report.

☐ No Revise commitment in CTD. No NRC notification required.

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Supervisor. List any documents (e.g., procedures, NRC submittals) affected by this change below: (*Attach additional page, as necessary*)

Description	EDMS #
TVA letter, dated 5/27/2011,Response to NRC Request for Additional Information Regarding Extending Allowed Outage Times for Technical Specification 3.8.1 (TAC Nos. ME5036, ME5037, and ME5038)	L444110601004
NRC letter, dated 10/5/2011, " BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3 -ISSUANCE OF AMENDMENTS TO REVISE THE TECHNICAL SPECIFICATIONS TO EXTEND THE EMERGENCY DIESEL GENERATOR ALLOWED OUTAGE TIME (TAC NOS. ME5036, ME5037, AND ME5038) (TS-468)	
0-TI-576 Temporary Diesel Generator (TDG)	
0-SR-3.8.1.1(TDG Implementation)	

Prepared by: R. K. Richter Jr. RKR [signature] Date: 6/6/16

APPROVALS

COMMITMENT CHANGE EVALUATION FORM (CCEF)

Sign/Print: L. T. STAFFORD / [signature] Date: 6/7/16
Responsible Supervisor

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Once approved by Licensing, this form shall be kept for the life of the plant in the EDMS vault and attached to CTD. EDMS # ____________ CCEF attached to CTD and emailed to RS: Y / N

Sign/Print: Jamie Paul / Jamie Paul Date: 6/7/16
Lic Eng / Lic Prog Mgr, Corporate or Site Lic Mgr

Attachment to Commitment Change (112906273), Response to NRC Request for Additional Information Regarding Extending Allowed Outage Times for Technical Specification 3.8.1 (TAC Nos. ME5036, ME5037, and ME5038) L444110601004 as amended by Commitment Change Evaluation Form (CCEF) dated 5/27/2011

Existing Commitment Descriptions:

Commitment 3.7.D from October 5, 2011 License Amendment

D. Licensed Operators and Assistant Unit Operators will be appropriately trained on the purpose and use of the TDGs and the revised Safe Shutdown Instructions (SSIs). A briefing/discussion of the revised TS 3.8.1 and putting TDGs in service will be completed prior to a planned EDG inoperability that exceeds 7 days. Operating crews will be briefed on the EDG work plan and procedural actions regarding loss-of-offsite power and station blackout (SBO).

Commitment 3.7.J from October 5, 2011 License Amendment

J. Compensatory actions for fire protection impairments associated with the Yellow violation will remain in effect. In addition, TVA will implement additional actions that will mitigate fire risk during the extended AOT for each EDG, as follows:

1. TVA will perform a qualitative analysis on electrical circuits/cable routing for each out-of-service (OOS) EDG for the 14-day AOT. This analysis will identify the fire areas where there is a lack of adequate cable separation associated with the use of TDGs.

After the 7th day of the extended 14-day AOT, a continuous fire watch will be provided for the fire areas which lack cable separation, as identified in the Qualitative Analysis. (Amended by 5/27/2011 Commitment Change Evaluation Form (CCEF).

2. TVA will revise the SSIs to incorporate guidance for use of the TDGs as an alternate AC power source for each OOS EDG. Additional or revised operator manual actions associated with the use of the TDGs in the SSIs will be minimized and validated. Actions will include starting the TDGs upon entering any SSI that credits the TDGs as a replacement for an OOS EDG. Operators will be trained on the revised SSIs.

3. After the 7th day of the extended 14-day AOT, an operator will be stationed at the TDGs to start the TDGs when directed, and to ensure reliable operation and equal load sharing of the TDGs.

4. TVA will impose additional restrictions on hot work in the affected fire zones/areas, and on elective work on Appendix R components.

5. Fire Operations personnel will perform walk downs to verify required controls of transient combustibles in affected fire zones/areas that could impact the operable EDGs, the TDGs'

availability, offsite power availability, or the ability to use the Bus Tie Board prior to entering and during each shift for the duration of the extended AOT.

Revised Commitment Description:

D. Licensed Operators and Assistant Unit Operators will be appropriately trained on the purpose and use of the TDGs and the ~~revised~~ Fire Safe Shutdown ~~Instructions~~ (~~SSIs~~FSSs) procedures. A briefing/discussion of the revised TS 3.8.1 and putting TDGs in service will be completed prior to a planned EDG inoperability that exceeds 7 days. Operating crews will be briefed on the EDG work plan and procedural actions regarding loss-of-offsite power and station blackout (SBO).

J. ~~Compensatory actions for fire protection impairments associated with the Yellow violation will remain in effect. In addition,~~ TVA will implement ~~additional~~ compensatory actions that will mitigate fire risk during the extended AOT for each EDG, as follows:

1. Deleted. ~~TVA will perform a qualitative analysis on electrical circuits/cable routing for each out-of-service (OOS) EDG for the 14-day AOT. This analysis will identify the fire areas where there is a lack of adequate cable separation associated with the use of TDGs.~~

~~After the 7th day of the extended 14-day AOT, a continuous fire watch will be provided for the fire areas which lack cable separation, as identified in the Qualitative Analysis.~~

2. Deleted. ~~TVA will revise the SSIs to incorporate guidance for use of the TGs as an alternate AC power source for each OOS EDG. Additional or revised operator manual actions associated with the use of the TDGs in the SSIs will be minimized and validated. Actions will include starting the TDGs upon entering any SSI that credits the TOGs as a replacement for an OOS EDG. Operators will be trained on the revised SSIs.~~

3. After the 7th day of the extended 14-day AOT, an operator will be stationed at the TDGs to start the TDGs when directed, and to ensure reliable operation and equal load sharing of the TDGs.

4. Deleted. ~~TVA will impose additional restrictions on hot work in the affected fire zones/areas, and on elective work on Appendix R components.~~

5. Fire Operations personnel ~~will~~ perform periodic walk downs to verify required controls of transient combustibles are in place. ~~in affected fire zones/areas that could impact the operable EDGs, the TDGs' availability, offsite power availability, or the ability to use the Bus Tie Board prior to entering and during each shift for the duration of the extended AOT.~~

6. BFN will maintain appropriate compensatory measures as described in the NFPA 805 Transition License Condition until all Table S-2, "Plant Modifications," of Tennessee Valley Authority letter CNL-15-191, dated September 8, 2015 are implemented.

Summarize Justification for Revising Commitment:

The commitment has been modified to reflect the new Risk-Informed, Performance-Based Fire Protection Program implemented by the NFPA 805 Program transition.

The basis for many of the compensatory measures defined in the License Amendment were based on the BFN Appendix R analyses and previously recognized Appendix R noncompliant conditions. BFN has transitioned to a risk-informed, performance-based Fire Protection Program in accordance with 10CFR 50.48 (c) and therefore, the Appendix R Program and supporting analyses have been superseded. Within the Appendix R analyses one train and subsequent power supply was credited. Deterministic analyses dictated that loss of a single credited component warranted compensatory measures in areas where out-of-service equipment was credited. Contrary to the Appendix R analyses, the NFPA 805 analyses considers multiple success paths for a given fire area and availability and reliability for each credited component was considered (including the EDGs). The NFPA 805 analysis did not credit the Temporary Diesel Generator (TDG) to achieve NFPA 805 compliance.

The NFPA 805 analysis is based on the Fire PRA which assumes that all NFPA 805 modifications have been installed. Consistent with BFN's NFPA 805 SER and Transition License Condition 2, appropriate compensatory measures will remain in place until all modifications listed in Table S-2,"Implementation Items," of Tennessee Valley Authority letters CNL-15-191, dated September 8, 2015, are implemented. These appropriate compensatory measures provide protective measures until modifications are all implemented.

Although not credited within the Fire PRA or NFPA 805 analyses, implementation of the TDG is performed per AOI-57-1A which is referenced in affected NFPA 805 Fire Safe Shutdown (FSS) instructions (previously defined as SSIs). Instructions for utilizing the TDGs have been incorporated directly into the Main Control Room Abandonment FSS. Although not formally credited, use of the TDG does provide additional means of emergency power and adds to emergency power system reliability.

Because the EDGs are fire risk significant, Risk Management Actions (defined by NPG-SPP-18.4.6) will be implemented within three days upon taking an EDG out of service per NPG-SPP-09.11.1, "*Equipment Out of Service (EOOS) Management.*"

As a result of the NFPA 805 transition, significant Fire Protection Program upgrades have been implemented to enhance the Fire Protection Program. Those items are defined within Attachment S, Table S-3 of the BFN March 2013 LAR (as updated by various submittals). Included in those upgrades were strengthening risk and defense of depth controls for hot work and transient combustibles. Additionally, inspection procedures are in place to verify that hot work and transient combustible controls have been implemented.

In summary, fire protection compensatory actions defined in the EDG License Amendment for the EDGs were based on BFN's Appendix R Program which has been superseded. BFN has implemented a Risk-Informed, Performance-Based Fire Protection Program (NFPA 805) which

does not credit the TDG. Although not required per the NFPA 805 Fire Protection Program, TDGs are required per TS 3.8.1 to support an extended AOT. Procedures are in place to implement the TDGs. As a result of the transition, NFPA 805 Fire Protection Program controls have been upgraded with strengthened hot work and transient combustible processes. Inspections are in place to verify hot work and transient controls are in place, thus enhancing fire prevention controls. The NFPA 805 analyses rely on modifications which have not yet all been implemented. Appropriate compensatory measures are in place to mitigate adverse conditions caused by modifications not yet implemented. Based on the NFPA 805 Fire Protection Program controls and these commitment changes, appropriate fire protection compensatory measures are in place when EDGs are out of service for more than 7 days.

R20 160310 009

COMMITMENT CHANGE EVALUATION FORM (CCEF)

Commitment Tracking Database (CTD) Number: Original-114569185 Hot Work Controls

Source Document: Original - J.W. Shea TVA to USNRC March 27, 2013 License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition) (Technical Specifications Change TS-480) (L44130327002) Date: 12/31/15

Existing Commitment Description: (*Attach additional page, as necessary*)

See Attached

Revised Commitment Description:

See Attached

Summarize Justification for Revising Commitment:

See Attached

See Attached FPPCRR

(*Refer to Attachments 3 (flow diagram) and 4 (description) for an explanation of the commitment change evaluation process.*)

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-09.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: The revised commitment has no impact on the ability to achieve safe shutdown in the event of a fire.. This commitment is an interim compensatory measure put in place to reduce the risk and consequences of a fire.

(*Attach additional page as necessary*)

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis. ______________________________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______________________________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional page. as necessary)

PART III

3.1 Was the original commitment (e.g.. response to NOV, LER) to restore an OBLIGATION (e.g.. rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☒ Yes Go to 4.2.

☐ No Go to Part V

4.2 Has the original commitment been implemented?

☒ Yes STOP You have completed this evaluation Revise commitment in CTD. Notify NRC of revised commitment in summary report

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date)

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☐ No STOP. You have completed this evaluation. Revise commitment in CTD. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in CTD. Notify NRC of revised commitment in next scheduled summary report.

☐ No Revise commitment in CTD. No NRC notification required.

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Supervisor. List any documents (e.g., procedures, NRC submittals) affected by this change below: (*Attach additional page, as necessary*)

Description	EDMS #
TVA to NRC letter dated March 31, 2013	L44130327002)
NPG-SPP-18.4.8	

Prepared by: R. K. Richter Jr. Date: 12/31/2015

APPROVALS Walter Mauer 3/10/16 Walt. Mull

Sign/Print: L. T. Stafford [signature] Date: 2-22-16
Responsible Supervisor

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Once approved by Licensing, this form shall be kept for the life of the plant in the EDMS vault and attached to CTD.
EDMS # L20 160310 009 3/10/16 DHZ CCEF attached to CTD and emailed to RS: Y / N

Sign/Print: T.E. Barrett Date: 3/10/16
Lic Eng, Lic Prog Mgr, Corporate or Site Lic Mgr

Attachment to Commitment Change (114569185), defined in March 31, 2013 NFPA 805 LAR Cover Letter

Existing Commitment Description:

Commitment 5

TVA will implement controls limiting hot work activities (e.g., welding, cutting, and grinding)in FA 01-03, FA 01-04, FA 02-02, FA 02-03, FA 02-04, FA 03-01, FA 03-02, FA 03-03, FA 04, FA 08, FA 09, FA 16, FA 20, FA 21, FA 22, FA 23, FA 24, FA 25-01, FA 26, and FA SWITCH until the modifications described for the listed FAs in Attachment S, Table S-2 "Plant Modifications Committed," of the enclosure to this letter are installed. If hot work activities are necessary, the existing hot work controls specified in NPG-SPP-18.4.8, Control of Ignition Sources, shall be augmented to require that the Fire Brigade or Senior Reactor Operator perform a pre-job briefing in accordance with NPG-SPP-18.2.2, Human Performance Tools, and perform a walkdown for job area familiarization prior to performing the hot work activity. This walkdown shall include, ensuring travel path is clear of obstructions, identifying the location of fire hoses and fire extinguishers, and verifying fire suppression systems (manual and automatic) are properly aligned.

Additionally, TVA letter to NRC dated May 16, 2013, Commitment 19 states:

TVA will inform the NRC 30 days prior to discontinuing each of the interim compensatory measures identified in Commitments 3 through 18 above.

To satisfy this commitment, this commitment change must be submitted to the NRC. Commitment 19 is not being changed but is the basis for required NRC notification.

Revised Commitment Description:

Commitment 5

Withdrawn (upon issuance of NPG-SPP-18.4.8 Revision 6)

Summarize Justification for Revising Commitment:

BFN has incorporated the administrative controls (pre-job briefings and walkdowns referenced in this commitment) into the existing NPG-SPP-18.4.8. Prior to implementation of the risk informed, performance based fire protection program, BFN will have implemented LAR Table S-3 items (06. 43, 45) to further strengthen BFN's controls of ignition sources. Controls are based on industry expectations and will focus on plant areas where risk significant scenarios involving hot work are present. Descriptions of S-3 items are as follows:

Item 06 - add controls on use of electric heaters and prohibition on use of portable fuel-fired heaters in plant areas containing equipment important to nuclear safety or where there is a potential for radiological releases resulting from a fire

Summarize Justification for Revising Commitment (Continued)

Item 43 - correct gaps found in NFPA 51B comparison
Item 45 - strengthen risk and defense in-depth administrative controls

Additionally, references to Shutdown Risk Management procedures to further control ignition sources will be added to address higher risk evolutions during non-power operations consistent with Item 26.

The existing controls and implementation of the LAR S-3 commitments will enhance the procedure to strengthen risk and defense in-depth controls of ignition sources. The revised procedure will focus on areas with risk significant scenarios involving hot work. A strengthened approach will remain in place after NFPA 805 modifications are implemented.

Additionally, periodic inspections, verifying ignition source controls are satisfied, are in place. These controls are permanent attributes of the Fire Protection Program, even after modifications are implemented.

Also, numerous compensatory actions associated with hot work remain in place in accordance with March 23, 2013 LAR cover letter and will remain throughout many existing plant fire areas until modifications are implemented. These include:

Item 07 - controls to protect RHR Pump 1A whenever hot work is performed in FAs 01-04, 04, 09
Item 08 - controls to protect Unit 1RHR Train A whenever hot work is performed in FA 08
Item 09 - controls to protect various equipment whenever hot work is performed in FA 02-02
Item 10 - controls to protect various equipment whenever hot work is performed in FA 02-03
Item 11 - controls to protect various equipment whenever hot work is performed in FA 02-04
Item 12 - controls to protect RHRSW Pump 2B whenever hot work is performed in FA 03-03
Item 17 - controls to protect various equipment whenever hot work is performed in FA 20

In summary, NPG-SPP-18.4.8, Control of Transient Combustibles is being revised to implement LAR Table S-3 items along with existing controls defined in Commitment 5 for fire areas where risk significant fire scenarios exist associated with hot work. This strengthened approach to control of ignition sources, along with the compensatory actions, and current Program requirements, will ensure that an appropriate level of fire protection is provided until modifications are implemented.

R20 160310 007

COMMITMENT CHANGE EVALUATION FORM (CCEF)

Commitment Tracking Database (CTD) Number: Original- 114569179 Transient Combustible Controls

Source Document: Original - J.W. Shea TVA to USNRC March 27, 2013 License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition) (Technical Specifications Change TS-480) (L44130327002) and modified by J.W. Shea TVA to USNRC August 26, 2014 Change in Commitment Related to INterim Compensatory Measures to Reduce Risk (no CTD number) Date: 08/26/2015

Existing Commitment Description: (*Attach additional page, as necessary*)

See Attached

Revised Commitment Description:

See Attached

Summarize Justification for Revising Commitment:

See Attached

See Attached FPPCRR

(*Refer to Attachments 3 (flow diagram) and 4 (description) for an explanation of the commitment change evaluation process.*)

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-09.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: The revised commitment has no impact on the ability to achieve safe shutdown in the event of a fire associated with transient combustibles. This commitment is an interim compensatory measure put in place to reduce the risk of a fire. See attached FPPCRR. Revised transient combustible controls consider transient fire scenarios.

(Attach additional page, as necessary)

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional page, as necessary)

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☒ Yes Go to 4.2.

☐ No Go to Part V.

4.2 Has the original commitment been implemented?

☒ Yes STOP. You have completed this evaluation. Revise commitment in CTD. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☐ No STOP. You have completed this evaluation. Revise commitment in CTD. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in CTD. Notify NRC of revised commitment in next scheduled summary report.

☐ No Revise commitment in CTD. No NRC notification required.

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Supervisor. List any documents (e.g., procedures, NRC submittals) affected by this change below: (*Attach additional page, as necessary*)

Description	EDMS #
TVA letter dated August 26, 2014	L44140826001
NPG-SPP-18.4.7	

Prepared by: R. K. Richter Jr Date. 09/17/2015

APPROVALS Walter Miller 3/10/16
Walter Miller

Sign/Print: L. T. Scaffidi Date: 2-22-16
Responsible Supervisor

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change

COMMITMENT CHANGE EVALUATION FORM (CCEF)

process has been used, and that this commitment change evaluation justifies the change.

Once approved by Licensing, this form shall be kept for the life of the plant in the EDMS vault and attached to CTD.
EDMS # R20 160310 007 DLR 3/10/16 CCEF attached to CTD and emailed to RS: Y / N

Sign/Print: TE Brandt Date: 3/10/16
Lic Eng, Lic Prog Mgr, Corporate or Site Lic Mgr

Attachment to Commitment Change for (original 114569179, as modified by August 26, 2014 letter)

Existing Commitment Description:

Commitment 4

TVA will implement stricter control of transient combustibles for the following BFN FAs [Fire Areas]: FA 01-03, FA 01-04, FA 02-02,FA 02-03, FA 02-04, FA 03-01, FA 03-02, FA 03-03, FA 16 (except for Main Control Room and Cable Spreading Room), FA 20, FA 21, and FA 25-01 (South Section of Elevation 550' only) until the modifications described for the listed FAs in Attachment S, Table S-2, "Plant Modifications Committed," of the enclosure to this letter are installed. Transient combustibles will be controlled as follows:

a. "In-Use" transient combustibles are acceptable without further evaluation; however, transient combustibles cannot be staged or stored in these higher risk areas unless:
1) a transient combustible evaluation is performed,
2) the transient combustibles are staged/stored in a closed metal container (e.g., cabinets, tool boxes, gang boxes, metal drums), or
3) the transient combustibles are continuously attended.

b. Transient combustible evaluations for staged/stored transient combustibles will be tracked to control quantities of transient combustibles allowed into these fire areas; and

c. Performing once-per-shift walk downs to ensure that transient combustible controls are satisfied (e.g., waste, debris, scraps, rags or other combustibles resulting from work activities are removed from the subject fire areas, and that no transient combustibles are placed in the 20-foot exclusion zone(s) within the listed FAs).

Additionally, TVA letter to NRC dated May 16, 2013, Commitment 19 states:

TVA will inform the NRC 30 days prior to discontinuing each of the interim compensatory measures identified in Commitments 3 through 18 above.

To satisfy this commitment, this commitment change must be submitted to the NRC. Commitment 19 is not being changed but is the basis for required NRC notification.

Revised Commitment Description:

Commitment 4

Withdrawn (upon issuance of NPG-SPP-18.4.7 Revision 8)

Summarize Justification for Revising Commitment:

Prior to implementation of the risk informed, performance based fire protection program, BFN will have implemented LAR Table S-3 items (04, 05, 11, 35, and 45) to strengthen BFN's controls of transient combustibles. Controls are based on industry expectations and will focus on plant areas where transient combustible-free areas is credited and risk significant scenarios involving transient combustibles are present. Descriptions of S-3 items are as follows:

Item 04 - allow only untreated lumber with a cross section dimension of 6" x 6" or larger to be used
Item 05 - delete allowance for materials meeting UL Standard 214
Item 11 - keep combustible fluids from coming in contact with hot pipes and surfaces, including insulated pipes and surfaces
Item 35 - establish limits on the types and quantities of materials in storage areas
Item 45 - strengthen risk and defense in-depth administrative controls

Based on item 45 of Table S-3, BFN will implement a graded approach to strengthen risk and defense in-depth controls of transient combustibles. This approach will have threshold values to determine the need for compensatory actions or engineering evaluations. Areas where combustible free areas are credited or there are risk significant transient combustible fire scenarios, will require greater controls allowing no, or limited quantity (lower threshold) of transient combustibles , consistent with item 35. If these threshold values of transient combustibles are exceeded, engineering evaluations and/or compensatory actions will be required. These controls will allow for mandatory work to be completed within the plant and implementation of compensatory actions should threshold values be exceeded. Similar controls with different threshold values and actions will be utilized for fire areas without risk significant transient fire scenarios.

The graded approach will continue to prohibit staging of transient combustibles in combustible free fire areas. In other areas, materials are permitted to be staged for short durations without evaluations or compensatory measures. This provision allows for the placement of material that would be consumed during a normal work shift, and the recognition that materials may pass through a fire area during staging or de-staging of a work activity.

Additionally, in areas where risk significant transient combustible scenarios exist and/or transient combustible-free areas are credited, periodic inspections, verifying transient combustible controls are satisfied, are in place. These controls are permanent attributes of the Fire Protection Program, even after modifications are implemented.

Also, compensatory action hourly and shiftily patrols remain in place until modifications are implemented.

In summary, NPG-SPP-18.4.7, Control of Transient Combustibles is being revised to implement LAR Table S-3 items which are consistent with the intent of Commitment 4. The graded

Summarize Justification for Revising Commitment (Continued)

approach to control of transient combustibles, along with current Program requirements and the compensatory actions, will ensure that an appropriate level of fire protection is provided until modifications are implemented.

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114569214

Source Document:

1. Letter from TVA to NRC, "License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition) (Technical Specification Change TS-480)," dated March 27, 2013 (ADAMS Accession No. ML13092A393)
2. Letter from TVA to NRC, "Response to NRC Request to Supplement License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants for the Browns Ferry Nuclear Plant, Units 1, 2, and 3 (TAC Nos. MF1185, MF1186, and MF1187)," dated May 16, 2013 (ML13141A291)
3. Letter from TVA to NRC, CNL-15-074, "Update to License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants for the Browns Ferry Nuclear Plant, Units 1, 2, and 3 (TAC Nos. MF1185, MF1186, and MF1187)," dated June 19, 2015 (ML15174A149)
4. Letter from TVA to NRC, CNL-14-140, "Change in Commitment Related to Interim Compensatory Measures to Reduce Fire Risk" dated August 14, 2014 (ML14231A961)
5. Letter from NRC to TVA, BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, and 3 – ISSUANCE OF AMENDMENTS REGARDING TRANSITION TO A RISK-INFORMED PERFORMANCE-BASED FIRE PROTECTION PROGRAM IN ACCORDANCE WITH 10 CFR 50.48© (CAC NOS. MF1185, MF1186, and MF1187," dated October 28, 2015 (ADAMS Accession No. ML15212A796)

Existing Commitment Description:

Commitment 6 in the referenced letters states: "TVA will implement controls to credit temporary diesel generators as an additional power source for a shutdown board, except during those periods when the temporary diesel generators are being used to support alternate decay heat removal or when moving a filled spent fuel storage cask past the temporary diesel generators, until the modifications described in Attachment S, Table S-2, "Plant Modifications Committed," of the enclose to this letter are installed."

Revised Commitment Description:

The revised Commtment 6 will state: "[Commitment withdrawn by TVA letter dated December 31, 2015]"

See CNL-15-253

Summarize Justification for Revising Commitment:

TVA voluntarily committed to the presence of the Temporary Diesel Gnerators (TDGs) as an interim compensatory measure based on conservative engineering judgment. See attached sheets for additional justification.

(Attach additional sheets, as necessary)

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

COMMITMENT CHANGE EVALUATION FORM

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Refer to rationale described in the attached continuation sheets

☐ Yes Go to 2.2

COMMITMENT CHANGE EVALUATION FORM

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______________________________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database. Notify NRC of revised commitment in summary report.

☐ No STOP Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

COMMITMENT CHANGE EVALUATION FORM

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

Although this evaluation indicates the commitment can be revised without NRC notification, TVA letter to NRC dated May 16, 2013 Commitment 19 states: "TVA will inform the NRC 30 days prior to discontinuing each of the interim compensatory measures identified in Commitments 3 through 18, above." Thus this commitment change must be submitted to NRC.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.

☐ No Revise commitment in commitment tracking database. No NRC notification required.

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	EDMS #
TVA letter to NRC dated May 16, 2013	L44130516001

Prepared by: [signature] Date: 12/17/15

APPROVALS

Print/Sign: JEFFREY S. BARKER / [signature] Date: 12/17/15
Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: Richard Eric Bates / Richard Eric Bates Date: 12/17/15
Nuclear Licensing

COMMITMENT CHANGE EVALUATION FORM

cc: Licensing File,
Responsible Organization
EDMS

Justification for removal of this commitment:

The rationale for this conclusion is as follows:

- The current safe shutdown instructions (SSIs) for fire under the Appendix R licensing basis provide a single success path to safe shutdown and involve intentionally disconnecting offsite power from the safety busses (Self-Induced Station Blackout, or SISBO), placing significant reliance on the availability of specific Emergency Diesel Generators (EDGs) depending on the fire area involved. The new fire safe shutdown procedures (FSSs) for BFN, which are currently scheduled to be implemented by May 23, 2016, eliminate SISBO and allow for the use of any available success path to achieve a Safe and Stable condition. For example, the FSSs will allow for the use of offsite power and EDG cross-tie capabilities between units if necessary.
- The inclusion of the temporary diesel generators (TDGs) as an interim compensatory measure was not based on risk quantification, but rather on conservative engineering judgment. The TDGs are not included in any BFN PRA models.
- Consistent with the NRC Safety Evaluation (Reference 5), appropriate compensatory measures will remain in place until the modifications described in Attachment S to the LAR are installed.
- The TDGs will continue to be available as needed during extended maintenance activities for the Emergency Diesel Generators (EDGs) in compliance with Technical Specifications Section 3.8.1.B.

Therefore, after reconsideration of the importance of the temporary diesel generators, TVA has concluded that the temporary diesel generators are not needed as an interim compensatory measure beyond the NFPA 805 Implementation date of May 23, 2016.

R20 160310 008

COMMITMENT CHANGE EVALUATION FORM (CCEF)

Commitment Tracking Database (CTD) Number: Original- 114569374 (Commitment #14 Transient Combustible Controls)

Source Document: Original - J.W. Shea TVA to USNRC March 27, 2013 License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition) (Technical Specifications Change TS-480) (L44130327002) and modified by J.W. Shea TVA to USNRC August 26, 2014 Change in Commitment Related to INterim Compensatory Measures to Reduce Risk (L44140826001) (no CTD number) Date: 08/26/2015

Existing Commitment Description: (*Attach additional page, as necessary*)

See Attached

Revised Commitment Description:

See Attached

Summarize Justification for Revising Commitment:

See Attached

See FPPCRR

(*Refer to Attachments 3 (flow diagram) and 4 (description) for an explanation of the commitment change evaluation process.*)

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-09.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: The revised commitment has no impact on the ability to achieve safe shutdown in the event of a fire associated with transient combustibles. This commitment is an interim compensatory measure put in place to reduce the risk of a fire. See attached FPPCRR. Revised transient combustible controls consider transient fire scenarios.

(Attach additional page, as necessary)

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ____________________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ____________________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ____________________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional page, as necessary)

PART III

3.1 Was the original commitment (e.g., response to NOV LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☒ Yes Go to 4.2.

☐ No Go to Part V.

4.2 Has the original commitment been implemented?

☒ Yes STOP. You have completed this evaluation. Revise commitment in CTD. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☐ No STOP. You have completed this evaluation. Revise commitment in CTD. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in CTD. Notify NRC of revised commitment in next scheduled summary report.

☐ No Revise commitment in CTD. No NRC notification required.

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Supervisor. List any documents (e.g., procedures, NRC submittals) affected by this change below: (*Attach additional page, as necessary*)

Description	EDMS #
TVA letter dated August 26, 2014	L44140826001
NPG-SPP-18.4.7	

Prepared by R. K. Richter Jr Date: 09/17/2015

APPROVALS WALTER MILLER 3/10/16 Walter Miller

Sign/Print: L. T. STAFFORD Date 2-22-16
Responsible Supervisor

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change

COMMITMENT CHANGE EVALUATION FORM (CCEF)

process has been used, and that this commitment change evaluation justifies the change.

Once approved by Licensing, this form shall be kept for the life of the plant in the EDMS vault and attached to CTD.
EDMS # L20 160310 008 DER CCEF attached to CTD and emailed to RS: Y / N
3/10/16

Sign/Print: T E Brandt Date: 3/10/16
Lic Eng, Lic Prog Mgr, Corporate or Site Lic Mgr

Attachment to Commitment Change for (original 114569374, as modified by August 26, 2014 letter)

Existing Commitment Description:

Commitment 14:

TVA will designate the FA 04, FA 08, FA 09, FA 16 (Cable Spreading Room only), FA 22, FA 23, FA 24, and FA 25-01 (North Section of Elevation 550' only) as transient combustible-free areas, perform once per- shift walk downs to ensure that transient combustible controls are satisfied, and establish the same controls as the current 20-foot exclusion zones, except that tags required by plant procedures may be present in these FAs, until the modifications for the listed FAs described in Attachment S, Table S-2. "Plant Modifications Committed," of the enclosure to this letter are installed.

Additionally, TVA letter to NRC dated May 16, 2013, Commitment 19 states:

TVA will inform the NRC 30 days prior to discontinuing each of the interim compensatory measures identified in Commitments 3 through 18 above.

To satisfy this commitment, this commitment change must be submitted to the NRC. Commitment 19 is not being changed but is the basis for required NRC notification.

Revised Commitment Description:

Commitment 14

Withdrawn (upon issuance of NPG-SPP-18.4.7 Rev 8)

Summarize Justification for Revising Commitment:

Prior to implementation of the risk informed, performance based fire protection program, BFN will have implemented LAR Table S-3 items (04, 05, 11, 35, and 45) to strengthen BFN's controls of transient combustibles. Controls are based on industry expectations and will focus on plant areas where transient combustible-free areas are credited or risk significant scenarios involving transient combustibles are present. Descriptions of S-3 items are as follows:

Item 04 - allow only untreated lumber with a cross section dimension of 6" x 6" or larger to be used
Item 05 - delete allowance for materials meeting UL Standard 214
Item 11 - keep combustible fluids from coming in contact with hot pipes and surfaces, including insulated pipes and surfaces
Item 35 - establish limits on the types and quantities of materials in storage areas
Item 45 - strengthen risk and defense in-depth administrative controls

Based on item 45 of Table S-3, BFN will implement a graded approach to strengthen risk and defense in-depth controls of transient combustibles. Areas where combustible free areas are credited or there are risk significant transient combustible fire scenarios, will require greater controls allowing no, or limited quantities of transient combustibles without evaluations or compensatory actions, consistent with item 35. If these threshold values of transient

Summarize Justification for Revising Commitment (continued)

combustibles are exceeded evaluations or compensatory actions will be required. These controls will allow for mandatory work to be completed within the plant and implementation of compensatory actions should threshold values be exceeded. Similar controls with different threshold values and actions will be utilized for fire areas without risk significant transient fire scenarios.

The graded approach will continue to prohibit staging of transient combustibles in combustible free fire areas. In other areas, materials are permitted to be staged for short durations without evaluations or compensatory measures. This provision allows for the placement of material that would be consumed during a normal work shift, and the recognition that materials may pass through a fire area during staging or de-staging of a work activity. Also, the revised transient combustible control procedure will allow the use of other non significant quantities of combustibles (e.g., signage, warning tape) that are required to support plant operations.

Additionally, in areas where risk significant transient combustible scenarios exist and/or transient combustible-free areas are credited, periodic inspections, verifying transient combustible controls are satisfied, are in place. These controls are attributes of the current Fire Protection Program even after modifications are implemented.

Also, compensatory action hourly and shiftily patrols remain in place until modifications are implemented.

In summary, NPG-SPP-18.4.7, Control of Transient Combustibles is being revised to implement LAR Table S-3 items which are consistent with the intent of Commitment 14. The graded approach to control of transient combustibles, along with current Program requirements and the compensatory actions, will ensure that an appropriate level of fire protection is provided until modifications are implemented.

C220 160311 016

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114574578

Source Document: NFPA 805 Transition LAR Date: 03/24/2013

Existing Commitment Description:

LAR - BFN Transition to 10CFR 50.48(c) -NFPA 805 This is a modification.

Install (8 total) interposing relays to isolate 4kV Shutdown Board breaker control circuits from spurious closure signals.

Revised Commitment Description:

LAR, Attachment S, Table S-2, Item 8 is revised to "Install (6 total) interposing relays to isolate 4kV Shutdown Board breaker control circuits from spurious closure due to fire damage to cables associated with lockout relays 86-1 and 86-2.

Summarize Justification for Revising Commitment:

The modification description is clarified to be more specific about the failure being addressed. There are 8 circuit breakers of concern that interface with lockout relays 86-1 and 86-2. However, two of the breakers are located in the same switchgear as the associated lockout relay with no external cables. Therefore two circuit breakers do not require installation of a lockout relay. The intent of the modification is not changed.

Variance from Deterministic Requirements (VFDR), Recovery Action, and LAR, Attachment A, "NEI 04-02 Table B-1 Transition of Fundamental Fire Protection Program & Design Elements," Changes:

This clarification requires no other changes to the LAR.

(Attach additional sheets, as necessary)

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity.

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

COMMITMENT CHANGE EVALUATION FORM

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.

☐ No Revise commitment in commitment tracking database. No NRC notification required.

COMMITMENT CHANGE EVALUATION FORM

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	EDMS #
NFPA 805 LAR	L44 130327 002

Prepared by: J. D. Wolcott Date: 6-17-15

APPROVALS

Print/Sign: Todd Stafford Date: 6-17-15
Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: Fred Mccall 8/4/16 Date: 07/15/2015
Nuclear Licensing

cc: Licensing File
Responsible Organization
EDMS

R20 160311 015

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114574790

Source Document: NFPA 805 Transition LAR   Date: 03/24/2013

Existing Commitment Description:

LAR - BFN Transition to 10CFR 50.48(c) -NFPA 805 This is a modification.

Install interposing relays (4 total) to isolate 4kV Shutdown Board crosstie breaker control circuits from spurious closure signals.

Revised Commitment Description:

LAR, Attachment S, Table S-2, Item 13 is revised to "Install interposing relays (4 total) to isolate 4kV Shutdown Board crosstie breaker control circuits from spurious closure due to fire damage to cables associated with lockout relays 86-SCA, 86-SCB, 86-SCC and 86-SCD.

Summarize Justification for Revising Commitment:

The modification description is clarified to be more specific about the failure being addressed. The modification is not changed.

Variance from Deterministic Requirements (VFDR), Recovery Action, and LAR, Attachment A, "NEI 04-02 Table B-1 Transition of Fundamental Fire Protection Program & Design Elements," Changes:

This clarification requires no other changes to the LAR.

(Attach additional sheets, as necessary)

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity.

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

COMMITMENT CHANGE EVALUATION FORM

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.

☐ No Revise commitment in commitment tracking database. No NRC notification required.

COMMITMENT CHANGE EVALUATION FORM

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	**EDMS #**
NFPA 805 LAR	L44 130327 002

Prepared by: J. D. Wolcott Date: 6-17-15

APPROVALS

Print/Sign: Todd Stafford Date: 6-17-15
Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: FRED MIZELL 9/14/16 Date: 07/15/2015
Nuclear Licensing

cc: Licensing File
Responsible Organization
EDMS

R20 160311 014

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114574882

Source Document: NFPA 805 Transition LAR   Date: 03/24/2013

Existing Commitment Description:

LAR - BFN Transition to 10CFR 50.48(c) -NFPA 805 This is a modification.

Install interposing relays (8 total) to isolate the 4kV Shutdown Board DG breaker control circuits from spurious closure signals.

Revised Commitment Description:

LAR, Attachment S, Table S-2, Item 18 is revised to "Install interposing relays (8 total) to isolate the 4kV Shutdown Board DG breaker control circuits from spurious closure due to fire damage to cables associated with lockout relays 86-GA, 86-GB, 86-GC, 86-GD, 86-G3A, 86-G3B, 86-G3C, and 86-G3D.

Summarize Justification for Revising Commitment:

The modification description is clarified to be more specific about the failure being addressed. The modification is not changed.

Variance from Deterministic Requirements (VFDR), Recovery Action, and LAR, Attachment A, "NEI 04-02 Table B-1 Transition of Fundamental Fire Protection Program & Design Elements," Changes:

This clarification requires no other changes to the LAR.

(Attach additional sheets, as necessary)

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity.

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

COMMITMENT CHANGE EVALUATION FORM

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.

☐ No Revise commitment in commitment tracking database. No NRC notification required.

COMMITMENT CHANGE EVALUATION FORM

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	EDMS #
NFPA 805 LAR	L44 130327 002

Prepared by: J. D. Wolcott Date: 6-17-15

APPROVALS

Print/Sign: Todd Stafford Date: 6-17-15

Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: FRED MIZELL 3/14/16 Date: 07/15/2015

Nuclear Licensing

cc: Licensing File
Responsible Organization
EDMS

E20 161025 079

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114690044

Source Document: NFPA 805 Transition LAR   Date: 03/24/2013

Existing Commitment Description:

LAR - BFN Transition to 10CFR 50.48(c) -NFPA 805 This is a modification.

Install additional fusing for trip circuits on all 4kV Shutdown Board load breakers and on nonsafety 4kV load breakers transitioning fire areas.

Revised Commitment Description: *

LAR, Attachment S, Table S-2, Item 24 is revised to "Install separate fuses for trip circuits extending outside the switchgear on all 4kV Shutdown Board load breakers and on non-safety 4kV load breakers transitioning fire areas." ← Verified actual verbiage in final Table S-2 JB 10/7/16

* This is a License Condition per the BFN NFPA 805 SE

Summarize Justification for Revising Commitment:

The modification description is clarified to better describe the modification.

Variance from Deterministic Requirements (VFDR), Recovery Action, and LAR, Attachment A, "NEI 04-02 Table B-1 Transition of Fundamental Fire Protection Program & Design Elements," Changes:

This clarification requires no other changes to the LAR.

(Attach additional sheets, as necessary)

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity.

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

COMMITMENT CHANGE EVALUATION FORM

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.

☐ No Revise commitment in commitment tracking database. No NRC notification required.

COMMITMENT CHANGE EVALUATION FORM

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	**EDMS #**
NFPA 805 LAR	L44 130327 002

Prepared by: J. D. Wolcott Date: 6-17-15

APPROVALS

Print/Sign: Todd Stafford ______ Date: 6-17-15
Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: FRED MIZELL ______ Date: 07/15/2015
Nuclear Licensing

TB 10/7/16

cc: *Licensing File*
Responsible Organization
EDMS

R20 161219 083

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114690158

Source Document: NFPA 805 Transition LAR   Date: 03/24/2013

Existing Commitment Description:

LAR - BFN Transition to 10CFR 50.48(c) -NFPA 805 This is a modification.

Install coordinated fuses to prevent loss of protective relaying for the eight safety related on-site Diesel Generators.

Revised Commitment Description:

LAR, Attachment S, Table S-2, Item 30 is revised to "Install separate fuses for circuits extending outside of the Diesel Generator compartments to prevent loss of protective relaying for the eight safety related on-site Diesel Generators for fires external to the Diesel Generator compartments.

This commitment is a License Condition per the NFPA 805 SE. JB 11/28/16

Summarize Justification for Revising Commitment:

The modification description is clarified to better describe the modification.

Variance from Deterministic Requirements (VFDR), Recovery Action, and LAR, Attachment A, "NEI 04-02 Table B-1 Transition of Fundamental Fire Protection Program & Design Elements," Changes:

This clarification requires no other changes to the LAR.

(Attach additional sheets, as necessary)

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity.

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

COMMITMENT CHANGE EVALUATION FORM

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.

☐ No Revise commitment in commitment tracking database. No NRC notification required.

COMMITMENT CHANGE EVALUATION FORM

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	EDMS #
NFPA 805 LAR	L44 130327 002

Prepared by: J. D. Wolcott Date: 6-17-15

APPROVALS

Print/Sign: Todd Stafford Date: 6-17-15
Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: FRED MIZELL Date: 07/15/2015
Nuclear Licensing

Verified 11/28/2016

cc: Licensing File
Responsible Organization
EDMS

1220 161220 090

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114698306

Source Document: NFPA 805 Transition LAR   Date: 03/24/2013

Existing Commitment Description:

LAR - BFN Transition to 10CFR 50.48(c) -NFPA 805 This is a modification.

Install adequate on-site Diesel Generator capacity to simultaneously run the three emergency high pressure makeup pumps (one per unit).

Revised Commitment Description:

LAR, Attachment S, Table S-2, Item 35a is revised to "Install adequate on-site Diesel Generator capacity to simultaneously run the three emergency high pressure makeup pumps (one per unit). Additionally, remove the hydrogen trailer port and the hydrogen system piping from this location.

In addition, the Problem Statement and Associated VFDRs discussion are revised for consistency with the proposed modification change. The Problem Statement is revised from "An additional power source for the emergency high pressure makeup pump is needed to improve baseline CDF," to "An additional power source for the emergency high pressure makeup pump is needed to improve baseline CDF. In addition, resolve the need for explosion proof fixtures at the hydrogen trailer port facility.

The Associated VFDRs description is revised from "Risk reduction modification not associated with a specific VFDR," to "Risk reduction modification not associated with a specific VFDR. Associated NFPA 805 Chapter 3 references: 3.3.1.2 (6), 3.3.7, 3.3.7.1, 3.3.7.2

This commitment is a License Condition per the NFPA 805 SE.
TB 11/28/16

Summarize Justification for Revising Commitment:

The hydrogen trailer port is the secondary source for hydrogen gas used in the Hydrogen Water Chemistry and Generator Hydrogen systems; however, this source is no longer used. A "hydrogen tank farm" located approximately 1/4 mile east of the plant is the primary source for hydrogen used in the plant, and no noncompliances with the tank farm were identified in the NFPA 50A Code Compliance Evaluation.

The Code Compliance Evaluation found that explosion-proof lighting is not installed at the hydrogen trailer port. Modification 35a in Table S-2 will preclude the need for explosion proof electrical fixtures by demolishing the hydrogen trailer port and removing the hydrogen system piping (i.e., the explosion hazard) from this location.

Variance from Deterministic Requirements (VFDR), Recovery Action, and LAR, Attachment A, "NEI 04-02 Table B-1 Transition of Fundamental Fire Protection Program & Design Elements," Changes:

(Attach additional sheets, as necessary)

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

COMMITMENT CHANGE EVALUATION FORM

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity.

☐ Yes Go to 2.2

COMMITMENT CHANGE EVALUATION FORM

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______________________________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

COMMITMENT CHANGE EVALUATION FORM

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.

☐ No Revise commitment in commitment tracking database. No NRC notification required.

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	EDMS #
NFPA 805 LAR	L44 130327 002
CNL-15-074	L44 150619 003

Prepared by: J. D. Wolcott Date: 07/15/2015

APPROVALS

Print/Sign: Todd Stafford [signature] Date: 7/15/2015
Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: Fred Mizelli [signature] Date: 07/15/2015
Nuclear Licensing

Verified [signature] 11/28/2016

cc: Licensing File
Responsible Organization
EDMS

1220 161219 081

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114698399

Source Document: NFPA 805 Transition LAR Date: 03/24/2013

Existing Commitment Description:

LAR - BFN Transition to 10CFR 50.48(c) -NFPA 805 This is a modification.

Modify the control circuits of valves FCV 73-34, FCV 73-35 and FCV 73-44 for all 3 Units to prevent spurious operation.

Revised Commitment Description:

LAR, Attachment S, Table S-2, Item 38 is revised to "Modify the control circuits of valves FCV 73-34, FCV 73-35 and FCV 73-44 for all 3 Units to reduce the probability of spurious operation.

In addition, the "Risk Informed Characterization," statement is revised from "Risk is reduced by preventing a possible unacceptable voltage drop on the batteries," to "Risk is reduced by reducing the probability of a possible unacceptable voltage drop on the batteries. This commitment is a License Condition per the NFPA 805 SE. 11/29/16

Summarize Justification for Revising Commitment:

The basis for Modification Item 38 is to address potential overload of station batteries due to spurious operation of large motor-operated valve (MOV) loads. The modification is designed to prevent spurious operation due to cable damage but will not deterministically prevent spurious operation of the valve for fires directly impacting the motor control center (MCC) compartment or the main control room (MCR) control switch for each valve. This method of addressing spurious valve operation is used with Table S-2 Modification Items 9a, 9b, 9c, 9d, 9e, 9f, and 54, and is described in the TVA response to NRC Request for Additional Information (RAI) for Safe Shutdown Analysis (SSA) RAI 12 in TVA letter dated December 20, 2013.

Variance from Deterministic Requirements (VFDR), Recovery Action, and LAR, Attachment A, "NEI 04-02 Table B-1 Transition of Fundamental Fire Protection Program & Design Elements," Changes:

This clarification requires no other changes to the LAR.

(Attach additional sheets, as necessary)

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity.

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: __________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: __________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: __________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

COMMITMENT CHANGE EVALUATION FORM

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.

☐ No Revise commitment in commitment tracking database. No NRC notification required.

COMMITMENT CHANGE EVALUATION FORM

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	EDMS #
NFPA 805 LAR	L44 130327 002

Prepared by: J. D. Wolcott   Date: 6-17-15

APPROVALS

Print/Sign: Todd Stafford ________ Date: 6-17-15
Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change:

Print/Sign: FRED MIZELL ________ Date: 07/15/2015
Nuclear Licensing

Verified ________ 11/28/2016

cc: Licensing File
Responsible Organization
EDMS

R20 161219 082

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114698542

Source Document: NFPA 805 Transition LAR Date: 03/24/2013

Existing Commitment Description:

LAR - BFN Transition to 10CFR 50.48(c) -NFPA 805 This is a modification.

Re-route cable 0PP285 associated with Shutdown Bus 2 normal feeder breaker 1722 away from Fire Area 02-01.

Revised Commitment Description:

LAR, Attachment S, Table S-2, Item 43 is revised to "Provide separation in Fire Area 02-01 for cable 0PP285 associated with Shutdown Bus 2 normal feeder breaker 1722.

This Commitment is a License Condition per the NFPA 805 SE. [illegible] 11/28/16

Summarize Justification for Revising Commitment:

The intent of the modification is to protect the cable from fire damage in Fire Area 02-01. The modification is not fully developed and therefore the description is being generalized to reflect the intent. Also the modification is revised to allow the option to use other approaches such as ERFBS.

Variance from Deterministic Requirements (VFDR), Recovery Action, and LAR, Attachment A, "NEI 04-02 Table B-1 Transition of Fundamental Fire Protection Program & Design Elements," Changes:

This clarification requires no other changes to the LAR.

(Attach additional sheets, as necessary)

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity.

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ____________________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ____________________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ____________________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

COMMITMENT CHANGE EVALUATION FORM

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.

☐ No Revise commitment in commitment tracking database. No NRC notification required.

COMMITMENT CHANGE EVALUATION FORM

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	EDMS #
NFPA 805 LAR	L44 130327 002

Prepared by: J. D. Wolcott Date: 6-17-15

APPROVALS

Print/Sign: Todd Stafford [signature] Date: 6-17-15
Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: FRED MIZELL [signature] Date: 07/15/2015
Nuclear Licensing

Verified. TE [signature] 11/28/2016

cc: Licensing File
Responsible Organization
EDMS

R20 161221 095

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114703120

Source Document: NFPA 805 Transition LAR   Date: 3/24/13

Existing Commitment Description:

LAR - BFN Transition to 10CFR 50.48(c) -NFPA 805 This is a modification.

Provide containment pressure and suppression pool level indication on the Backup Control Panel.

Revised Commitment Description:

LAR, Attachment S, Table S-2, Item 52b is revised to "Provide containment pressure indication on the Backup Control Panel."

In addition, the "Problem Statement" and "Risk Informed Characterization," statements are revised to conform to the proposed modification change. The "Problem Statement" is revised from "Containment pressure and suppression pool level indications are required for control room abandonment to support decay heat removal through the HWWV," to "Containment pressure indication is required for control room abandonment to support decay heat removal through the HWWV."

The "Risk Informed Characterization" is revised from "Reduces risk by allowing additional redundancy of drywell pressure and suppression pool level for indication to support decay heat removal via the HWWV," to "Reduces risk by allowing additional redundancy of drywell pressure for indication to support decay heat removal via the HWWV."

This commitment is a License Condition per the NFPA 805 SE.
[illegible] 11/28/16

Summarize Justification for Revising Commitment:

The basis for this modification is to ensure that adequate instrumentation is available to support containment venting with the hardened wet well vent (HWWV). TVA does not plan to require indication of suppression pool level in the fire safe shutdown procedures when utilizing the HWWV in fire safe shutdown events. Therefore, modification of the suppression pool level instrumentation is not required.

Variance from Deterministic Requirements (VFDR), Recovery Action, and LAR, Attachment A, "NEI 04-02 Table B-1 Transition of Fundamental Fire Protection Program & Design Elements," Changes:

This clarification requires no other changes to the LAR.

(Attach additional sheets, as necessary)

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity.

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

COMMITMENT CHANGE EVALUATION FORM

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.

☐ No Revise commitment in commitment tracking database. No NRC notification required.

COMMITMENT CHANGE EVALUATION FORM

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	**EDMS #**
NFPA 805 LAR	L44 130327 002

Prepared by: J. D. Wolcott Date: 6-17-15

APPROVALS

Print/Sign: Todd Stafford [signature] Date: 6-17-15
Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: FRED MIZELL [signature] Date: 07/15/2015
Nuclear Licensing
Verified [signature] 11/28/2016

cc: Licensing File
Responsible Organization
EDMS

T420 161221 096

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114703145

Source Document: NFPA 805 Transition LAR

Date: 3/24/13

Existing Commitment Description:

LAR - BFN Transition to 10CFR 50.48(c) -NFPA 805 This is a modification.

Re-route cables as required for Drywell wide range pressure instruments P-64-160A and B and Suppression Pool wide range level L-64-159A and B.

Revised Commitment Description:

LAR, Attachment S, Table S-2, Item 52c is revised to "Re-route cables as required to provide drywell pressure indication in the MCR."

In addition, the "Problem Statement" and "Risk Informed Characterization," statements are revised to conform to the proposed modification change. The "Problem Statement" is revised from "Containment pressure and suppression pool level indications are required in all fire areas to support decay heat removal through the HWWV. Cables and power supplies for redundant instrument loops may be damaged by the same fire scenario." to "Containment pressure indication is required in all fire areas to support decay heat removal through the HWWV. Cables and power supplies for redundant instrument loops may be damaged by the same fire scenario."

The "Risk Informed Characterization" is revised from "Reduces risk by allowing additional redundancy of drywell pressure and suppression pool level for indication to support decay heat removal via the HWWV." to "Reduces risk by allowing additional redundancy of drywell pressure for indication to support decay heat removal via the HWWV."

This commitment is a License Condition per the NFPA 805 SE. 11/28/16

Summarize Justification for Revising Commitment:

The basis for this modification is to ensure that adequate instrumentation is available to support containment venting with the hardened wet well vent (HWWV). TVA does not plan to require indication of suppression pool level in the fire safe shutdown procedures when utilizing the HWWV in fire safe shutdown events. Therefore, modification of the suppression pool level instrumentation is not required.

Other drywell pressure loops in addition to P-64-160A and B are available with sufficient range to support HWWV operation and therefore the modification solution is not limited to drywell pressure loops P-64-160A and B.

Variance from Deterministic Requirements (VFDR), Recovery Action, and LAR, Attachment A, "NEI 04-02 Table B-1 Transition of Fundamental Fire Protection Program & Design Elements," Changes:

This clarification requires no other changes to the LAR.

(Attach additional sheets, as necessary)

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity.

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: __________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: __________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: __________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

COMMITMENT CHANGE EVALUATION FORM

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.

☐ No Revise commitment in commitment tracking database. No NRC notification required.

COMMITMENT CHANGE EVALUATION FORM

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	EDMS #
NFPA 805 LAR	L44 130327 002

Prepared by: J. D. Wolcott Date: 6-17-15

APPROVALS

Print/Sign: Todd Stafford Date: 6-17-15
Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: FRED Mizell Date: 07/15/2015
Nuclear Licensing
Verified TE Brandt 11/28/2016

cc: Licensing File
Responsible Organization
EDMS

K20 161220 092

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114703194

Source Document: NFPA 805 Transition LAR Date: 03/24/2013

Existing Commitment Description:

LAR - BFN Transition to 10CFR 50.48(c) -NFPA 805 This is a modification.

For Drywell wide range pressure instruments P-64-160A and B and Suppression Pool wide range level instruments L-64-159A and B, provide isolation of associated circuits and make appropriate power supply available such that both division instruments are not lost in the same fire scenario.

Revised Commitment Description:

LAR, Attachment S, Table S-2, Item 52a is revised to "For Drywell wide range pressure instruments P-64-160A and B provide isolation of associated circuits and make appropriate power supply available such that both division instruments are not lost in the same fire scenario."

In addition, the "Problem Statement" and "Risk Informed Characterization" statements are revised to conform to the proposed modification change. The "Problem Statement" is revised from "Containment pressure and suppression pool level indications are required in all fire areas to support decay heat removal through the HWWV. Cables and power supplies for redundant instrument loops may be damaged by the same fire scenario." to "Containment pressure indication is required in all fire areas to support decay heat removal through the HWWV. Cables and power supplies for redundant instrument loops may be damaged by the same fire scenario."

The "Risk Informed Characterization" is revised from "Reduces risk by allowing additional redundancy of drywell pressure and suppression pool level for indication to support decay heat removal via the HWWV." to "Reduces risk by allowing additional redundancy of drywell pressure for indication to support decay heat removal via the HWWV."

This Commitment is a License Condition per NFPA 805 SE. JB 11/28/16

Summarize Justification for Revising Commitment:

The basis for this modification is to ensure that adequate instrumentation is available to support containment venting with the hardened wet well vent (HWWV). TVA does not plan to require indication of suppression pool level in the fire safe shutdown procedures when utilizing the hardened wet well vent (HWWV) in fire safe shutdown events. Therefore, modification of the suppression pool level instrumentation is not required.

Variance from Deterministic Requirements (VFDR), Recovery Action, and LAR, Attachment A, "NEI 04-02 Table B-1 Transition of Fundamental Fire Protection Program & Design Elements," Changes:

This clarification requires no other changes to the LAR.

(Attach additional sheets, as necessary)

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity.

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

COMMITMENT CHANGE EVALUATION FORM

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.

☐ No Revise commitment in commitment tracking database. No NRC notification required.

COMMITMENT CHANGE EVALUATION FORM

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	EDMS #
NFPA 805 LAR	L44 130327 002

Prepared by: J. D. Wolcott Date: 6-17-15

APPROVALS

Print/Sign: Todd Stafford ____________ Date: 6-17-15
Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: FRED MIZELL ____________ Date: 07/15/2015
Nuclear Licensing

cc: Licensing File
Responsible Organization
EDMS

R20 161220 091

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114703257

Source Document: NFPA 805 Transition LAR Date: 03/24/2013

Existing Commitment Description:

LAR - BFN Transition to 10CFR 50.48(c) -NFPA 805 This is a modification.

Enclose in ERFBS and re-route cable ES2691-II (RHR Pump 2D control cables) out of Fire Areas 02-03 and 02-04 to prevent spurious start.

Revised Commitment Description:

LAR, Attachment S, Table S-2, Item 57 is revised to "Provide separation in Fire Areas 02-03 and 02-04 for cable ES2691-II (RHR Pump 2D control cable) to prevent spurious start. This commitment is a License Condition per the NFPA 805 SE. TB 11/28/16

Summarize Justification for Revising Commitment:

The intent of the modification is to protect the cable from fire damage in Fire Areas 02-03 and 02-04. The modification is not fully developed and therefore the description is being generalized to reflect the intent.

Variance from Deterministic Requirements (VFDR), Recovery Action, and LAR, Attachment A, "NEI 04-02 Table B-1 Transition of Fundamental Fire Protection Program & Design Elements," Changes:

This clarification requires no other changes to the LAR.

(Attach additional sheets, as necessary)

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity.

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: __________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: __________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: __________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

COMMITMENT CHANGE EVALUATION FORM

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

- ☐ Yes Go to Question 3.2.
- ☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

- ☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.
- ☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

- ☐ Yes Go to 4.2.
- ☒ No Go to Part V.

4.2 Has the original commitment been implemented?

- ☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database. Notify NRC of revised commitment in summary report.
- ☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

- ☐ Yes Go to Question 5.2.
- ☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

- ☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.
- ☐ No Revise commitment in commitment tracking database. No NRC notification required.

COMMITMENT CHANGE EVALUATION FORM

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	EDMS #
NFPA 805 LAR	L44 130327 002

Prepared by: J. D. Wolcott Date: 6-17-15

APPROVALS

Print/Sign: Todd Stafford Date: 6-17-15
Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: Fred Mizell Date: 07/15/2015
Nuclear Licensing Verified /E Brandt 11/28/2016

cc: Licensing File
Responsible Organization
EDMS

R20 161220 093

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114703304

Source Document: NFPA 805 Transition LAR Date: 03/24/2013

Existing Commitment Description:

LAR - BFN Transition to 10CFR 50.48(c) -NFPA 805 This is a modification.

Modify the control circuit for Bus Tie breakers 1642 and 1742 to prevent spurious closure.

Revised Commitment Description:

LAR, Attachment S, Table S-2, Item 60 is revised to "Modify the control circuit for Bus Tie breakers 1642 and 1742 to prevent spurious closure from circuits extending outside of the switchgear.

Summarize Justification for Revising Commitment:

The modification description is clarified to better describe the modification.

Variance from Deterministic Requirements (VFDR), Recovery Action, and LAR, Attachment A, "NEI 04-02 Table B-1 Transition of Fundamental Fire Protection Program & Design Elements," Changes:

This clarification requires no other changes to the LAR. This commitment is a License Condition per NFPA 805 SE.

(Attach additional sheets, as necessary)

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity.

☐ Yes Go to 2.2

COMMITMENT CHANGE EVALUATION FORM

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ____

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ____

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ____

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

COMMITMENT CHANGE EVALUATION FORM

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.

☐ No Revise commitment in commitment tracking database. No NRC notification required.

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	EDMS #
NFPA 805 LAR	L44 130327 002

Prepared by: J. D. Wolcott Date: 6-17-15

APPROVALS

Print/Sign: Todd Stafford [signature] Date: 6-17-15
Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: Fred Mizell [signature] for Date: 07/15/2015
Nuclear Licensing
Verified [signature] 11/28/2016

cc: Licensing File
Responsible Organization
EDMS

R20 161219 084

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114712500

Source Document: NFPA 805 Transition LAR Date: 03/24/2013

Existing Commitment Description:

LAR - BFN Transition to 10CFR 50.48(c) -NFPA 805 This is a modification.

Re-route control cable 3ES1571-I for RHR Pump 3C away from Fire Area 03-02.

Revised Commitment Description:

LAR, Attachment S, Table S-2, Item 69 is revised to "Provide separation for cable 3ES1571-I (RHR Pump 3C control cable) in Fire Area 03-02 to prevent spurious start.

This commitment is a License Condition per NFPA-805 SE. [initials] 11/28/16

Summarize Justification for Revising Commitment:

The intent of the modification is to protect the cable from fire damage in Fire Area 03-02. The modification is not fully developed and therefore the description is being generalized to reflect the intent. Also the modification is revised to allow the option to use other approaches such as ERFBS.

Variance from Deterministic Requirements (VFDR), Recovery Action, and LAR, Attachment A, "NEI 04-02 Table B-1 Transition of Fundamental Fire Protection Program & Design Elements," Changes:

This clarification requires no other changes to the LAR.

(Attach additional sheets, as necessary)
Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity.

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______________________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

COMMITMENT CHANGE EVALUATION FORM

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.

☐ No Revise commitment in commitment tracking database. No NRC notification required.

COMMITMENT CHANGE EVALUATION FORM

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	EDMS #
NFPA 805 LAR	L44 130327 002

Prepared by: J. D. Wolcott Date: 6-17-15

APPROVALS

Print/Sign: Todd Stafford Date: 6-17-15
Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: Fred Mizell Date: 07/15/2015
Nuclear Licensing

Verified [signature] 11/28/2016

cc: Licensing File
Responsible Organization
EDMS

K20 161220 089

COMMITMENT CHANGE EVALUATION FORM (CCEF)

Commitment Tracking Database (CTD) Number: 114712572

Source Document: NFPA 805 Transition LAR Date: 05/02/2016

Existing Commitment Description: (*Attach additional page, as necessary*)

LAR Section: Attachment S, Table S-2 Plant Modifications Committed Modification 55: Reroute cables for the Main Generator CT circuits away from the auxiliary instrument rooms. Install separate fuses in main Relay Panel for USST coolers. The problem description is also changed to delete "for the EHC system" and add "Fire damage to cables can result in loss of USST cooling.

Revised Commitment Description:

LAR Section: Attachment S, Table S-2 Plant Modifications Committed Modification 75: Reroute cables for the Main Generator CT circuits away from the auxiliary instrument rooms. Install separate fuses in main Relay Panel for USST coolers. The problem description is also changed to delete "for the EHC system" and add "Fire damage to cables can result in loss of USST cooling.

This commitment is a License Condition per NFPA 805 SE TB 11/28/16

Summarize Justification for Revising Commitment:

Correct Typo in description. Previous CCEF inadvertently changed the comittment number from 75 to 55. Correct commitment is S2-75. Reference attached Enclosure 4 from CNL-14-025

(*Refer to Attachments 3 (flow diagram) and 4 (description) for an explanation of the commitment change evaluation process.*)

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-09.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: ____________________

(Attach additional page, as necessary)

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ____________________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ____________________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ____________________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional page, as necessary)

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in CTD. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in CTD. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in CTD. Notify NRC of revised commitment in next scheduled summary report.

☐ No Revise commitment in CTD. No NRC notification required.

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Supervisor. List any documents (e.g., procedures, NRC submittals) affected by this change below: (*Attach additional page, as necessary*)

Description	**EDMS #**

Prepared by: Anita Parker   Date: 05/02/2016

APPROVALS

Sign/Print: L.T. STAFFORD   Date: 5-4-16

Responsible Supervisor

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change

process has been used, and that this commitment change evaluation justifies the change.

Once approved by Licensing, this form shall be kept for the life of the plant in the EDMS vault and attached to CTD.
EDMS # ____________________ CCEF attached to CTD and emailed to RS: Y / N

Sign/Print: T E Barnett ______________________ Date: 11/29/16
Lic Eng, Lic Prog Mgr, Corporate or Site Lic Mgr

R20 161219 085

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114712681

Source Document: NFPA 805 Transition LAR Date: 03/24/2013

Existing Commitment Description:

LAR - BFN Transition to 10CFR 50.48(c) -NFPA 805 This is a modification.

Reduce the overcurrent settings for 4kV Shutdown Bus feeder breakers to protect Shutdown Bus from overload.

- Breaker 1126
- Breaker 1226
- Breaker 1132
- Breaker 1232

Revised Commitment Description:

LAR, Attachment S, Table S-2, Item 84 is revised to "Provide overload protection for 4kV Shutdown Bus feeder breakers.

- Breaker 1126
- Breaker 1226
- Breaker 1132
- Breaker 1232

This commitment is a License Condition per the NFPA 805 SE. JB 11/28/16

Summarize Justification for Revising Commitment:

The revised modification description continues to protect the subject 4kV Shutdown Buses from overload while allowing other methods of protection to be implemented.

Variance from Deterministic Requirements (VFDR), Recovery Action, and LAR, Attachment A, "NEI 04-02 Table B-1 Transition of Fundamental Fire Protection Program & Design Elements," Changes:

This clarification requires no other changes to the LAR.

(Attach additional sheets, as necessary)

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity.

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

COMMITMENT CHANGE EVALUATION FORM

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.

☐ No Revise commitment in commitment tracking database. No NRC notification required.

COMMITMENT CHANGE EVALUATION FORM

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	EDMS #
NFPA 805 LAR	L44 130327 002

Prepared by: J. D. Wolcott Date: 6-17-15

APPROVALS

Print/Sign: Todd Stafford [signature] Date: 6-17-15
Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: FRED MIZELL [signature] Date: 07/15/2015
Nuclear Licensing
Verified: [signature] 11/28/2016

cc: *Licensing File*
Responsible Organization
EDMS

L20 161025 078

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114712713

Source Document: NFPA 805 Transition LAR Date: 03/24/2013

Existing Commitment Description:

LAR - BFN Transition to 10CFR 50.48(c) -NFPA 805 This is a modification.

Install a shorting/disconnect switch in the MCR to prevent spurious opening of valves PC 69-15 for all 3 Units during power operation.

Revised Commitment Description:

LAR, Attachment S, Table S-2, Item 90 * is revised to "Install a shorting/disconnect switch in the MCR to reduce the probability of spurious opening of valves PCV 69-15 for all 3 Units during power operation." verified this matches words in final Tables. 10/7/16

* This is a License Condition per the BFN NFPA 805 SE.

Summarize Justification for Revising Commitment:

The modification description is revised to allow the use of a performance based approach which reduces the probability of failure but may not prevent failure.

Variance from Deterministic Requirements (VFDR), Recovery Action, and LAR, Attachment A, "NEI 04-02 Table B-1 Transition of Fundamental Fire Protection Program & Design Elements," Changes:

This clarification requires no other changes to the LAR.

(Attach additional sheets, as necessary)

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity.

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

COMMITMENT CHANGE EVALUATION FORM

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.

☐ No Revise commitment in commitment tracking database. No NRC notification required.

COMMITMENT CHANGE EVALUATION FORM

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	**EDMS #**
NFPA 805 LAR	L44 130327 002

Prepared by: J. D. Wolcott Date: 6-17-15

APPROVALS

Print/Sign: Todd Stafford [signature] Date: 6-17-15
Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: FRED Mizell [signature] Date: 07/15/2015
Nuclear Licensing

Verified [initials] 10/7/16

cc: Licensing File
Responsible Organization
EDMS

R20 160531 029

COMMITMENT CHANGE EVALUATION FORM (CCEF)

Commitment Tracking Database (CTD) Number: 114T12054

Source Document: NFPA 805 Transition LAR Date: 05/05/2016

Existing Commitment Description: *(Attach additional page, as necessary)*

LAR Section: 4.1 and Attachment A, 3.10.1(1)

There are corrective actions identified in MDQ099920100004 that will be completed prior to NFPA 805 implementation as follows:

-Install warning signs in conspicuous locations in and around the Lube Oil Purification Room.

-Replace the existing CO2 system safety signs with signs that comply with the three-panel format retroactively required by NFPA 12-2008.

- Adjust the CO2 discharge timer for the Control Building CO2 systems in the Computer Rooms 1,2 and 3, and Auxiliary Instrument Rooms 1,2 and 3 such that a CO2 design concentration of 50% in maintained for a minimum of 20 minutes.

Revised Commitment Description:

LAR Section: 4.1 and Attachment A, 3.10.1(1)

There are corrective actions identified in MDQ099920100004 that will be completed prior to NFPA 805 implementation as follows:

-Install warning signs in conspicuous locations in and around the Lube Oil Purification Room.

-Replace the existing CO2 system safety signs with signs that comply with the three-panel format retroactively required by NFPA 12-2008.

Summarize Justification for Revising Commitment:

TVA has determined that the manual carbon dioxide supression systems located in the Auxillary Instrumentation Rooms and Computer Rooms (all within FA 16) are not required. Therefore, these systems should not be identified as "Required" and upgrades are not needed to support the NFPA805 transition. CR 946868

(Refer to Attachments 3 (flow diagram) and 4 (description) for an explanation of the commitment change evaluation process.)

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-09.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: The upgrade is not needed for the system to accomplish it's NFPA 805 function

(Attach additional page, as necessary)

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional page, as necessary)

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in CTD. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in CTD. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in CTD. Notify NRC of revised commitment in next scheduled summary report.

☐ No Revise commitment in CTD. No NRC notification required.

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Supervisor. List any documents (e.g., procedures, NRC submittals) affected by this change below: (*Attach additional page: as necessary*)

Description	EDMS #
NFPA 805 LAR	L44130327002
MDN0009992012000100-Fire Rish Evaluation	R14141219108
CNL-14-208	L44141217001

Prepared by: J.D. Wolcott Date: 05/05/2016

APPROVALS

Sign/Print: L. T. Stafford [signature] Date: 5-6-16
Responsible Supervisor

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change

process has been used, and that this commitment change evaluation justifies the change.

Once approved by Licensing, this form shall be kept for the life of the plant in the EDMS vault and attached to CTD.
EDMS # R20 160531 029 CCEF attached to CTD and emailed to RS: Y / N

Sign/Print: [signature] Date: 5/15/16
Lic Eng, Lic Prog Mgr, Corporate or Site Lic Mgr

R20 160622 068

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114772326

Source Document: NFPA 805 Transition LAR   Date: 03/24/2013

Existing Commitment Description:

LAR Section: 4.1 and Attachment A, 3.3.1.2(6)

Implement WO 1132433146 to install explosion proof electrical fixtures in the Hydrogen Trailer Port Facility

Revised Commitment Description:

LAR, Attachment S, Table S-3, item 36 is deleted.

Summarize Justification for Revising Commitment:

Removal of hydrogen hazard will not be accomplished by WO 1132433146. The hydrogen hazard associated with the hydrogen trailer port will be removed as a result of Table S-2, Modification 35a that will install adequate diesel generator capacity to simultaneously run the three emergency high pressure makeup pumps.

(Attach additional sheets, as necessary)

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity.

☐ Yes Go to 2.2

COMMITMENT CHANGE EVALUATION FORM

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ____________________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ____________________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ____________________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

COMMITMENT CHANGE EVALUATION FORM

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.

☐ No Revise commitment in commitment tracking database. No NRC notification required.

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	EDMS #
NFPA 805 LAR	L44 130327 002

Prepared by: J. D. Wolcott Date: 6-17-15

APPROVALS

Print/Sign: Todd Stafford [signature] Date: 6-17-15
Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: [signature] Date: 6/21/16
Nuclear Licensing

cc: Licensing File
Responsible Organization
EDMS

R20 160620 040

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114774135

Source Document: NFPA 805 Transition LAR   Date: 03/24/2013

Existing Commitment Description:

LAR Section: 4.1 and Attachment A. 3.3.1.2(4)

Revise NPG-SPP-18.4.7 to establish limits on the types and quantities of materials in designated permanent storage areas.

Revised Commitment Description:

LAR, Attachment S, Table S-3, item 35 is revised to "Revise NPG-SPP-18.4.7 to establish limits on the types and quantities of materials in designated storage areas.

Summarize Justification for Revising Commitment:

The item for Implementation identifies a change to NPG-SPP-18.4.7, Control of Transient Combustibles, which does not address permanent storage. Permanent storage is addressed via the design change process/controls. The origin of the Implementation Item comes from the B-1 table section 3.3.1.2 Control of Combustible Materials (4) that discusses storage and staging of combustible materials and does not use the term "permanent." Limits on the types and quantities of transient combustible materials being stored and staged are being incorporated into NPG-SPP-18.4.7. The proposed change is consistent with the Compliance Basis for section 3.3.1.2 Control of Combustible Materials (4).

See Referenced document L44 150619 003 CNL-15-074

(Attach additional sheets, as necessary)

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II

COMMITMENT CHANGE EVALUATION FORM

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity.

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ________________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ________________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ________________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

COMMITMENT CHANGE EVALUATION FORM

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.

☐ No Revise commitment in commitment tracking database. No NRC notification required.

COMMITMENT CHANGE EVALUATION FORM

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	EDMS #
NFPA 805 LAR	L44 130327 002

Prepared by: J. D. Wolcott Date: 6-17-15

APPROVALS

Print/Sign: Todd Stafford Date: 6-17-15
Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: Date: 6/15/16
Nuclear Licensing

cc: Licensing File
Responsible Organization
EDMS

R20 160311 017

COMMITMENT CHANGE EVALUATION FORM (CCEF)

Commitment Tracking Database (CTD) Number: 114776636

Source Document: NFPA 805 Transition LAR & Safe Shutdown Analysis Request for Additional Information (RAI) 15) Date: 09/10/2015

Existing Commitment Description: (*Attach additional page, as necessary*)

LAR Section 4.2.1.1: Attachment S, Table S-3 Item 25: The additional considerations of NEI-00-01, Revision 2, Chapter 3 will be addressed by linking the EOIs to fire safe shutdown procedures consistent with the recommendations of BWROG document, BRWOG-TP-11-011, entitled, "BROWG Assessment of Generic Multiple Spurious Operations (MSOs) in Post-Fire Safe Shutdown Circuit Analysis for the Operating BWR Plants, dated June 2011."

See attached changes to RAI 15 identified in Enclosure 1

Revised Commitment Description:

LAR Section 4.2.1.1: Attachment S, Table S-3 Item 25: The additional considerations of NEI-00-01, Revision 2, Chapter 3 will be addressed by linking the EOIs to fire safe shutdown procedures consistent with the recommendations of BWROG document, BRWOG-TP-11-011, entitled, "BROWG Assessment of Generic Multiple Spurious Operations (MSOs) in Post-Fire Safe Shutdown Circuit Analysis for the Operating BWR Plants, dated June 2011.", with the exception of the MCR abandonment Fire Safe Shutdown procedures. The MCR abandoment Fire Safe Shutdown procedures will include procedure steps for fires impacting the ability to scram from the MCR.

See attached changes to RAI 15 identified in Enclosure 1. Changes to the response are denoted by deleted text struck through, inserted text underlined and in bold, and a revision bar in the right margin.

Summarize Justification for Revising Commitment:

Revised to reflect how BFN will inplement Fire Safe Shutdown procedures for Main Control Room abandoment

(*Refer to Attachments 3 (flow diagram) and 4 (description) for an explanation of the commitment change evaluation process.*)

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-09.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: The proposed alternate modification accomplishes the same function as the original committment

(Attach additional page, as necessary)

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☒ No

Basis: __________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☒ No

Basis: __________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☒ No

Basis: __________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional page, as necessary)

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in CTD. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in CTD. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in CTD. Notify NRC of revised commitment in next scheduled summary report.

☐ No Revise commitment in CTD. No NRC notification required.

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Supervisor. List any documents (e.g., procedures, NRC submittals) affected by this change below: (*Attach additional page, as necessary*)

Description	EDMS #
NFPA 805 LAR	L44130327002
CNL-15-191	L44150908001

Prepared by: Anita Parker Date: 09/24/2015

APPROVALS

Sign/Print: L. T. Stagorio [signature] Date: 1-6-16
Responsible Supervisor

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change

COMMITMENT CHANGE EVALUATION FORM (CCEF)

process has been used, and that this commitment change evaluation justifies the change.

Once approved by Licensing, this form shall be kept for the life of the plant in the EDMS vault and attached to CTD.
EDMS # ____________ CCEF attached to CTD and emailed to RS: Y / N

Sign/Print: TE Brandt TE Brandt Date: 3/4/16
Lic Eng, Lic Prog Mgr, Corporate or Site Lic Mgr

R20 160620 041

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114782716

Source Document: NFPA 805 Transition LAR   Date: 03/24/2013

Existing Commitment Description:

LAR Section: 4.1 and Attachment A. 3.8.1

1st bullet - Revise the fire alarm procedures to include retention of fire alarm signals received for at least one year

3rd bullet - Update Procedures 0-SI-4.11.A.1(1), 0-SI-4.11.A.1(2), 0-SI-4.11.A.1(3)a, 0-SI-4.11.A.1(3)b, 0-SI-4.11.A.1(4), 1-SI-4.11.A.1(1), 2-SI-4.11.A.1(1), 2-SI-4.11.A.1(2), 2-SI-4.11.A.1(3), 3-SI-4.11.A.1(1), 3-SI-4.11.A.1(2), 3-SI-4.11.A.1(3), and 3-SI-4.11.A.1(4) to exclude test magnets from being used during smoke detector testing and to ensure smoke detectors are tested and activated using chemical smoke designed solely for smoke detector testing.

Revised Commitment Description:

1st bullet - LAR, Attachment S, Table S-3, item 19 (1st bullet) is revised to "Revise the applicable procedure to include retention of fire alarm signals received for at least one year

3rd bullet - LAR, Attachment S, Table S-3, item 19, 3rd bullet, is revised to "Update applicable testing procedures to exclude test magnets from being used during smoke detector testing and to ensure smoke detectors are tested and activated using chemical smoke designed solely for smoke detector testing.

Summarize Justification for Revising Commitment:

1st bullet - This requirement will be located in a procedure that defines Fire Operations responsibilities and will not be located in a fire alarm procedure.

3rd bullet - The existing list contains procedures that do not require change. Significant changes in testing procedures are being made to implement NFPA 805. The proposed text allows flexibility to combine, rename, or create new testing procedures.

(Attach additional sheets, as necessary) See Referenced document L44 150619003 CNL-15-074

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2. Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II

COMMITMENT CHANGE EVALUATION FORM

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity.

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______

If any of the above questions are answered Yes, STOP: Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

COMMITMENT CHANGE EVALUATION FORM

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.

☐ No Revise commitment in commitment tracking database. No NRC notification required.

COMMITMENT CHANGE EVALUATION FORM

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	EDMS #
NFPA 805 LAR	L44 130327 002

Prepared by: J. D. Wolcott Date: 6-17-15

APPROVALS

Print/Sign: Todd Stafford ______________ Date: 6-17-15
Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: ______________ Date: 6/14/16
Nuclear Licensing

cc: Licensing File
Responsible Organization
EDMS

R20 160622 066

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114782752

Source Document: NFPA 805 Transition LAR   Date: 03/24/2013

Existing Commitment Description:

LAR Section: 4.1 and Attachment A. 3.5.3

Conduct test measurements of the fire pumps against the rated capacity (2,500 gpm at 130 psig). Based on the findings, evaluate the overall health of the pump.

Revised Commitment Description:

LAR, Attachment S. Table S-3, item 17. 2nd bullet is deleted.

See Referenced document L44 150619 003   Verified in Letter CNL-15-074 and in LAR Table S-3 Item 17 markup (see CCDEP)

Summarize Justification for Revising Commitment:

BFN currently tests fire pumps to meet system demand requirements of 2,250 gpm at 130 psig vs. the original fire pump rated capacity of 2,500 gpm at 130 psig, and trends data to support system health monitoring. The 2,250 gpm at 130 psig test criteria is supported by an engineering evaluation and, with testing, ensures that a single pump can supply the largest fire system flow demand plus margin. Pump testing is performed at a variety of flows and pressures so pump performance can be evaluated.

(Attach additional sheets, as necessary)

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity.

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

COMMITMENT CHANGE EVALUATION FORM

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

- ☐ Yes Go to Question 3.2.
- ☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

- ☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.
- ☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

- ☐ Yes Go to 4.2.
- ☒ No Go to Part V.

4.2 Has the original commitment been implemented?

- ☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database. Notify NRC of revised commitment in summary report.
- ☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

- ☐ Yes Go to Question 5.2.
- ☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

- ☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.
- ☐ No Revise commitment in commitment tracking database. No NRC notification required.

COMMITMENT CHANGE EVALUATION FORM

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	EDMS #
NFPA 805 LAR	L44 130327 002

Prepared by: J. D. Wolcott Date: 6-17-15

APPROVALS

Print/Sign: Todd Stafford Date: 6-17-15
Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: Date: 6/16/16
Nuclear Licensing

cc: Licensing File
Responsible Organization
EDMS

R20 160620 042

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114785981

Source Document: NFPA 805 Transition LAR   Date: 03/24/2013

Existing Commitment Description:

LAR Section: 4.1 and Attachment A. 3.11.3(1)

Revise appropriate procedures to inspect and ensure guides and bearings of sliding fire doors are maintained well lubricated.

Revised Commitment Description:

LAR. Attachment S. Table S-3. item 7 is revised to "Revise appropriate procedures to inspect and ensure guides and bearings of active NFPA 805 required sliding fire doors are maintained well lubricated.

Summarize Justification for Revising Commitment:

This text clarifies that changes will be made for NFPA 805 required doors only. There are other sliding doors at BFN that are not required per the NFPA 805 program. Additionally, some NFPA 805 required sliding doors are bolted in place and not active, so maintaining door bearings on non active doors is not practical.

Reference document L44 150619003 CNL-15-074

(Attach additional sheets, as necessary)

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity

☐ Yes Go to 2.2

COMMITMENT CHANGE EVALUATION FORM

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______________________________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in commitment tracking database Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date)

COMMITMENT CHANGE EVALUATION FORM

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP. You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report.

☐ No Revise commitment in commitment tracking database. No NRC notification required.

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization. List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	EDMS #
NFPA 805 LAR	L44 130327 002

Prepared by: J. D. Wolcott Date: 6-17-15

APPROVALS

Print/Sign: Todd Stafford [signature] Date: 6-17-15

Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: TE Brandt [signature] Date: 6/17/15

Nuclear Licensing

cc. Licensing File
Responsible Organization
EDMS

R20 160531 030

COMMITMENT CHANGE EVALUATION FORM

Commitment Tracking Number: 114790295

Source Document: NFPA 805 Transition LAR   Date: 03/24/2013

Existing Commitment Description:

LAR Section: 4.1 and Attachment A. 4.6.2 and Attachment A 3.2.3(3)

The monitoring program required by NFPA 805 Section 2.6 will be implemented after the LAR approval as part of the fire protection program transition to NFPA 805, in accordance with NFPA 805 FAQ 10-0059, and will include a process that reviews fire protection performance and trends in performance. Program specifics are provided in LAR Section 4.6.2.

Revised Commitment Description:

LAR, Attachment S, Table S-3, Item 3 is revised to "The monitoring program required by NFPA 805 Section 2.6 will be implemented as part of the fire protection program transition to NFPA 805, in accordance with NFPA 805 FAQ 10-0059, and will include a process that reviews fire protection performance and trends in performance. Program specifics are provided in LAR Section 4.6.2.

Summarize Justification for Revising Commitment:

The intent of this change is to eliminate issuance of the SE as a restraint to implementing NFPA 805 Monitoring procedures.

See CNL-15-074   L44 150619 003   Referenced document

(Attach additional sheets, as necessary)

Refer to Appendix C for a flow diagram that outlines the commitment change evaluation process.

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II

COMMITMENT CHANGE EVALUATION FORM

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-9.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Updating the remaining Implementation items and modifications of more clarity.

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ____________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ____________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ____________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional information, as necessary)

COMMITMENT CHANGE EVALUATION FORM

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP You have completed this evaluation. Revise commitment in commitment tracking database Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☒ No STOP You have completed this evaluation. Revise commitment in commitment tracking database. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in commitment tracking database. Notify NRC of revised commitment in next annual/RFO interval summary report

☐ No Revise commitment in commitment tracking database No NRC notification required

COMMITMENT CHANGE EVALUATION FORM

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Organization List any documents (e.g., procedures, NRC submittals) affected by this change below:

Description	EDMS #
NFPA 805 LAR	L44 130327 002

Prepared by: J. D Wolcott Date: 06/23/2015

APPROVALS

Print/Sign: Todd Stafford _______ Date: 6/23/2015
Responsible Organization

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Print/Sign: _______ Date: 5/15/16
Nuclear Licensing

cc: Licensing File
Responsible Organization
EDMS

R08 160412 127

COMMITMENT CHANGE EVALUATION FORM (CCEF)

Commitment Tracking Database (CTD) Number: 115295456

Source Document: CNL-14-068 [L44150129001] Supplement 1 to LAR TS-485 Date: 01/29/2015

Existing Commitment Description: (*Attach additional page, as necessary*)

To ensure the flow through the 0.1875-inch orifice is not obstructed, radiography inspections of the 0.1875-inch orifice will be performed during each refueling outage. These inspections will first be performed during the spring outage of 2014 for Unit 3, during the fall outage of 2014 for Unit 1, and during the spring outage of 2015 for Unit 2.

Revised Commitment Description:

Since the commitment source, LAR TS-485, was withdrawn, the commitment is also withdrawn/closed.

Summarize Justification for Revising Commitment:

CNL-15-070 [L44150529001] dated 05/29/2015 - Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Withdrawal of Proposed Technical Specification Change to Revise the Leakage Rate Through MSIVs - TS-485 and commitment to revise the Reply to NOV EA-11-252 regarding the ALT pathways by July 15, 2015.

NRC acknowledgement of Withdrawn LAR dated 06/16/2015 [NRC ADAMS Accession No. ML15161A344]
CNL-15-123 [L44150715002] Updated Reply to NOV EA-11-252

Attachments: Commitment Load Sheet, Maximo CTD screenshot, CNL-15-070 [L44150529001], NRC Letter [ML15161A344, and CNL-15-123 [L44150715002]

(*Refer to Attachments 3 (flow diagram) and 4 (description) for an explanation of the commitment change evaluation process.*)

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-09.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Commitment was associated with a LAR that has been withdrawn.

(*Attach additional page, as necessary*)

☐ Yes Go to 2.2

COMMITMENT CHANGE EVALUATION FORM (CCEF)

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______________________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional page, as necessary)

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☒ Yes Go to Question 3.2.

☐ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☒ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

After several rounds of RAIs and discussions, TVA withdrew the LAR and intends to resolve the NOV using plant modifications under 10 CFR 50.59. Subsequently, NRC acknowledged the Withdrawal of the LAR on 06/16/2015 [NRC ADAMS Accession No. ML15161A344]

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in CTD. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☒ Yes Go to Question 5.2.

☐ No STOP. You have completed this evaluation. Revise commitment in CTD. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☒ Yes Revise commitment in CTD. Notify NRC of revised commitment in next scheduled summary report.

NRC was notified of LAR withdrawal and NRC concurred with the withdrawal - See Justification.

☐ No Revise commitment in CTD. No NRC notification required.

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Supervisor. List any documents (e.g., procedures, NRC submittals) affected by this change below: (*Attach additional page, as necessary*)

Description	EDMS #
Impact Review not required due to commitment withdrawal/closure	

Prepared by: M. W. Oliver Date: 04/12/2016

APPROVALS

Sign/Print: J. L. Paul Jamie Paul Date: 4/12/16
Responsible Supervisor

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Once approved by Licensing, this form shall be kept for the life of the plant in the EDMS vault and attached to CTD.
EDMS # R08160412127 CCEF attached to CTD and emailed to RS Y / N

Sign/Print: M. W. Oliver MW Oliver Date: 04/12/2016
Lic Eng, Lic/Prog Mgr, Corporate or Site Lic Mgr

COMMITMENT CHANGE EVALUATION FORM (CCEF)

Commitment Tracking Database (CTD) Number: 116201617

Source Document: CNL-14-068 [L44150129001] Supplement 1 to LAR TS-485 Date: 01/29/2015

Existing Commitment Description: (*Attach additional page, as necessary*)

TVA will perform a radiography examination, during each Unit's refueling outage, starting with the Unit 2 outage in the spring of 2015, on the 1 inch globe valve (BOV-1-525) to ensure that it is not plugged.

Revised Commitment Description:

Since the commitment source, LAR TS-485, was withdrawn, the commitment is also withdrawn/closed.

Summarize Justification for Revising Commitment:

CNL-15-070 [L44150529001] dated 05/29/2015 - Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Withdrawal of Proposed Technical Specification Change to Revise the Leakage Rate Through MSIVs - TS-485 and commitment to revise the Reply to NOV EA-11-252 regarding the ALT pathways by July 15, 2015.

NRC acknowledgement of Withdrawn LAR dated 06/16/2015 [NRC ADAMS Accession No. ML15161A344] CNL-15-123 [L44150715002] Updated Reply to NOV EA-11-252

Attachments: Commitment Load Sheet, Maximo CTD screenshot, CNL-15-070 [L44150529001], NRC Letter [ML15161A344, and CNL-15-123 [L44150715002]

(*Refer to Attachments 3 (flow diagram) and 4 (description) for an explanation of the commitment change evaluation process.*)

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-09.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Commitment was associated with a LAR that has been withdrawn.

(*Attach additional page, as necessary*)

☐ Yes Go to 2.2

COMMITMENT CHANGE EVALUATION FORM (CCEF)

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______________________________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional page, as necessary)

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☒ Yes Go to Question 3.2.

☐ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☒ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

After several rounds of RAIs and discussions, TVA withdrew the LAR and intends to resolve the NOV using plant modifications under 10 CFR 50.59. Subsequently, NRC acknowledged the Withdrawal of the LAR on 06/16/2015 [NRC ADAMS Accession No. ML15161A344]

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in CTD. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☒ Yes Go to Question 5.2.

☐ No STOP. You have completed this evaluation. Revise commitment in CTD. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☒ Yes Revise commitment in CTD. Notify NRC of revised commitment in next scheduled summary report.

NRC was notified of LAR withdrawal and NRC concurred with the withdrawal - See Justification.

☐ No Revise commitment in CTD. No NRC notification required.

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Supervisor. List any documents (e.g., procedures, NRC submittals) affected by this change below: (*Attach additional page, as necessary*)

Description	**EDMS #**
Impact Review not required due to commitment withdrawal/closure	

Prepared by: M. W. Oliver Date: 04/12/2016

APPROVALS

Sign/Print: J. L. Paul Jamie Paul Date: 4/12/16
Responsible Supervisor

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Once approved by Licensing, this form shall be kept for the life of the plant in the EDMS vault and attached to CTD.
EDMS # R08 160412 129 CCEF attached to CTD and emailed to RS Y / N

Sign/Print: M. W. Oliver M W Oliver Date: 04/12/2016
Lic Eng, Lic Prog Mgr, Corporate or Site Lic Mgr

R08 160412 128

COMMITMENT CHANGE EVALUATION FORM (CCEF)

Commitment Tracking Database (CTD) Number: 116201648

Source Document: CNL-14-068 [L44150129001] Supplement 1 to LAR TS-485 Date: 01/29/2015

Existing Commitment Description: *(Attach additional page, as necessary)*

TVA will perform a radiography examination, during each Unit's refueling outage, starting with the Unit 2 outage in the spring of 2015, on the 0.25-inch orifice around valves FCV-1-168, FCV-1-169, FCV-1-170, and FCV-1-171 to ensure that the flow path through these orifices is not plugged.

Revised Commitment Description:

Since the commitment source, LAR TS-485, was withdrawn, the commitment is also withdrawn/closed.

Summarize Justification for Revising Commitment:

CNL-15-070 [L44150529001] dated 05/29/2015 - Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Withdrawal of Proposed Technical Specification Change to Revise the Leakage Rate Through MSIVs - TS-485 and commitment to revise the Reply to NOV EA-11-252 regarding the ALT pathways by July 15, 2015.

NRC acknowledgement of Withdrawn LAR dated 06/16/2015 [NRC ADAMS Accession No. ML15161A344]
CNL-15-123 [L44150715002] Updated Reply to NOV EA-11-252

Attachments: Commitment Load Sheet, Maximo CTD screenshot, CNL-15-070 [L44150529001], NRC Letter [ML15161A344, and CNL-15-123 [L44150715002]

(Refer to Attachments 3 (flow diagram) and 4 (description) for an explanation of the commitment change evaluation process.)

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-09.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Commitment was associated with a LAR that has been withdrawn.

(Attach additional page, as necessary)

☐ Yes Go to 2.2

COMMITMENT CHANGE EVALUATION FORM (CCEF)

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______________________________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional page, as necessary)

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☒ Yes Go to Question 3.2.

☐ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☒ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

<u>After several rounds of RAIs and discussions, TVA withdrew the LAR and intends to resolve the NOV using plant modifications under 10 CFR 50.59. Subsequently, NRC acknowledged the Withdrawal of the LAR on 06/16/2015 [NRC ADAMS Accession No. ML15161A344]</u>

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in CTD. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☒ Yes Go to Question 5.2.

☐ No STOP. You have completed this evaluation. Revise commitment in CTD. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☒ Yes Revise commitment in CTD. Notify NRC of revised commitment in next scheduled summary report.

NRC was notified of LAR withdrawal and NRC concurred with the withdrawal - See Justification.

☐ No Revise commitment in CTD. No NRC notification required.

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Supervisor. List any documents (e.g., procedures, NRC submittals) affected by this change below: (*Attach additional page, as necessary*)

Description	EDMS #
Impact Review not required due to commitment withdrawal/closure	

Prepared by: M. W. Oliver Date: 04/12/2016

APPROVALS

Sign/Print: J. L. Paul Jamie Paul Date: 4/12/16
Responsible Supervisor

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Once approved by Licensing, this form shall be kept for the life of the plant in the EDMS vault and attached to CTD.
EDMS # R08160412128 CCEF attached to CTD and emailed to RS: Y / N

Sign/Print: M. W. Oliver MWOliver Date: 04/12/2016
Lic Eng, Lic Prog Mgr, Corporate or Site Lic Mgr

ROB 160412 126

COMMITMENT CHANGE EVALUATION FORM (CCEF)

Commitment Tracking Database (CTD) Number: 116201658

Source Document: CNL-14-068 [L44150129001] Supplement 1 to LAR TS-485 Date: 01/29/2015

Existing Commitment Description: (*Attach additional page, as necessary*)

TVA will verify FCV-1-57 is open and the motive power is removed prior to entry into Mode 3 from Mode 4. Furthermore, TVA will ensure, through administrative means that FCV-1-57 is open and that the motive power is removed when the Unit is in Modes 1, 2, and 3.

Revised Commitment Description:

Since the commitment source, LAR TS-485, was withdrawn, the commitment is also withdrawn/closed.

Summarize Justification for Revising Commitment:

CNL-15-070 [L44150529001] dated 05/29/2015 - Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Withdrawal of Proposed Technical Specification Change to Revise the Leakage Rate Through MSIVs - TS-485 and commitment to revise the Reply to NOV EA-11-252 regarding the ALT pathways by July 15, 2015.

NRC acknowledgement of Withdrawn LAR dated 06/16/2015 [NRC ADAMS Accession No. ML15161A344]
CNL-15-123 [L44150715002] Updated Reply to NOV EA-11-252

Attachments: Commitment Load Sheet, Maximo CTD screenshot, CNL-15-070 [L44150529001], NRC Letter [ML15161A344, and CNL-15-123 [L44150715002]

(*Refer to Attachments 3 (flow diagram) and 4 (description) for an explanation of the commitment change evaluation process.*)

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-09.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Commitment was associated with a LAR that has been withdrawn.

(*Attach additional page, as necessary*)

☐ Yes Go to 2.2

COMMITMENT CHANGE EVALUATION FORM (CCEF)

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______________________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional page, as necessary)

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☒ Yes Go to Question 3.2.

☐ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☒ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

After several rounds of RAIs and discussions, TVA withdrew the LAR and intends to resolve the NOV using plant modifications under 10 CFR 50.59. Subsequently, NRC acknowledged the Withdrawal of the LAR on 06/16/2015 [NRC ADAMS Accession No. ML15161A344]

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in CTD. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☒ Yes Go to Question 5.2.

☐ No STOP. You have completed this evaluation. Revise commitment in CTD. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☒ Yes Revise commitment in CTD. Notify NRC of revised commitment in next scheduled summary report.

NRC was notified of LAR withdrawal and NRC concurred with the withdrawal - See Justification.

☐ No Revise commitment in CTD. No NRC notification required.

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Supervisor. List any documents (e.g., procedures, NRC submittals) affected by this change below: (*Attach additional page, as necessary*)

Description	EDMS #
Impact Review not required due to commitment withdrawal/closure	

Prepared by: M. W. Oliver Date: 04/12/2016

APPROVALS

Sign/Print: J. L. Paul Jamie Paul Date: 4/12/16
Responsible Supervisor

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Once approved by Licensing, this form shall be kept for the life of the plant in the EDMS vault and attached to CTD.
EDMS # R08160412126 CCEF attached to CTD and emailed to RS: Y N

Sign/Print: M. W. Oliver MWOliver Date: 04/12/2016
Lic Eng, Lic Prog Mgr, Corporate or Site Lic Mgr

R08 160412 132

COMMITMENT CHANGE EVALUATION FORM (CCEF)

Commitment Tracking Database (CTD) Number: 116201665

Source Document: CNL-14-068 [L44150129001] Supplement 1 to LAR TS-485 Date: 01/29/2015

Existing Commitment Description: (*Attach additional page, as necessary*)

TVA will revise procedures to require removing motive power from valves FCV-1-58 and FCV-1-59 once they have been opened in response to receiving a main steam line radiation monitor alarm or a drywell radiation monitor alarm.

Revised Commitment Description:

Since the commitment source, LAR TS-485, was withdrawn, the commitment is also withdrawn/closed.

Summarize Justification for Revising Commitment:

CNL-15-070 [L44150529001] dated 05/29/2015 - Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Withdrawal of Proposed Technical Specification Change to Revise the Leakage Rate Through MSIVs - TS-485 and commitment to revise the Reply to NOV EA-11-252 regarding the ALT pathways by July 15, 2015.

NRC acknowledgement of Withdrawn LAR dated 06/16/2015 [NRC ADAMS Accession No. ML15161A344]
CNL-15-123 [L44150715002] Updated Reply to NOV EA-11-252

Attachments: Commitment Load Sheet, Maximo CTD screenshot, CNL-15-070 [L44150529001], NRC Letter [ML15161A344, and CNL-15-123 [L44150715002]

(*Refer to Attachments 3 (flow diagram) and 4 (description) for an explanation of the commitment change evaluation process.*)

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-09.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Commitment was associated with a LAR that has been withdrawn.

(*Attach additional page, as necessary*)

☐ Yes Go to 2.2

COMMITMENT CHANGE EVALUATION FORM (CCEF)

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______________________________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional page, as necessary)

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☒ Yes Go to Question 3.2.

☐ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☒ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

After several rounds of RAIs and discussions, TVA withdrew the LAR and intends to resolve the NOV using plant modifications under 10 CFR 50.59. Subsequently, NRC acknowledged the Withdrawal of the LAR on 06/16/2015 [NRC ADAMS Accession No. ML15161A344]

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in CTD. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☒ Yes Go to Question 5.2.

☐ No STOP. You have completed this evaluation. Revise commitment in CTD. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☒ Yes Revise commitment in CTD. Notify NRC of revised commitment in next scheduled summary report.

NRC was notified of LAR withdrawal and NRC concurred with the withdrawal - See Justification.

☐ No Revise commitment in CTD. No NRC notification required.

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Supervisor. List any documents (e.g., procedures, NRC submittals) affected by this change below: (*Attach additional page, as necessary*)

Description	EDMS #
Impact Review not required due to commitment withdrawal/closure	

Prepared by: M. W. Oliver Date: 04/12/2016

APPROVALS

Sign/Print: J. L. Paul Jamie Paul Date: 4/12/16
Responsible Supervisor

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Once approved by Licensing, this form shall be kept for the life of the plant in the EDMS vault and attached to CTD, EDMS # R08 160412 132 CCEF attached to CTD and emailed to RS: Y / N

Sign/Print: M. W. Oliver M W Oliver Date: 04/12/2016
Lic Eng, Lic Prog Mgr, Corporate or Site Lic Mgr

R08 160412 131

COMMITMENT CHANGE EVALUATION FORM (CCEF)

Commitment Tracking Database (CTD) Number: 116201669

Source Document: CNL-14-068 [L44150129001] Supplement 1 to LAR TS-485 Date: 01/29/2015

Existing Commitment Description: (*Attach additional page, as necessary*)

TVA will revise procedures to require manually restoring motive power to valves FCV-1-58 and FCV-1-59 on RMOV Board 3C in the event of a loss of offsite power in combination with receipt of a main steam line radiation monitor alarm or a drywell radiation monitor alarm.

Revised Commitment Description:

Since the commitment source, LAR TS-485, was withdrawn, the commitment is also withdrawn/closed.

Summarize Justification for Revising Commitment:

CNL-15-070 [L44150529001] dated 05/29/2015 - Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Withdrawal of Proposed Technical Specification Change to Revise the Leakage Rate Through MSIVs - TS-485 and commitment to revise the Reply to NOV EA-11-252 regarding the ALT pathways by July 15, 2015.

NRC acknowledgement of Withdrawn LAR dated 06/16/2015 [NRC ADAMS Accession No. ML15161A344]
CNL-15-123 [L44150715002] Updated Reply to NOV EA-11-252

Attachments: Commitment Load Sheet, Maximo CTD screenshot, CNL-15-070 [L44150529001], NRC Letter [ML15161A344, and CNL-15-123 [L44150715002]

(*Refer to Attachments 3 (flow diagram) and 4 (description) for an explanation of the commitment change evaluation process.*)

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-09.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Commitment was associated with a LAR that has been withdrawn.

(*Attach additional page, as necessary*)

☐ Yes Go to 2.2

COMMITMENT CHANGE EVALUATION FORM (CCEF)

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______________________________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional page, as necessary)

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☒ Yes Go to Question 3.2.

☐ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☒ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

After several rounds of RAIs and discussions, TVA withdrew the LAR and intends to resolve the NOV using plant modifications under 10 CFR 50.59. Subsequently, NRC acknowledged the Withdrawal of the LAR on 06/16/2015 [NRC ADAMS Accession No. ML15161A344]

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in CTD. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☒ Yes Go to Question 5.2.

☐ No STOP. You have completed this evaluation. Revise commitment in CTD. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☒ Yes Revise commitment in CTD. Notify NRC of revised commitment in next scheduled summary report.

NRC was notified of LAR withdrawal and NRC concurred with the withdrawal - See Justification.

☐ No Revise commitment in CTD. No NRC notification required.

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Supervisor. List any documents (e.g., procedures, NRC submittals) affected by this change below: *(Attach additional page, as necessary)*

Description	EDMS #
Impact Review not required due to commitment withdrawal/closure	

Prepared by: M. W. Oliver Date: 04/12/2016

APPROVALS

Sign/Print: J. L. Paul Jamie Paul Date: ~~4/16~~ JLP 4/12/16 4/12/16
Responsible Supervisor

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Once approved by Licensing, this form shall be kept for the life of the plant in the EDMS vault and attached to CTD.
EDMS # R08160412 131 CCEF attached to CTD and emailed to RS: Y/N

Sign/Print: M. W. Oliver MW Oliver Date: 04/12/2016
Lic Eng, Lic Prog Mgr, Corporate or Site Lic Mgr

R0816041Z 130

COMMITMENT CHANGE EVALUATION FORM (CCEF)

Commitment Tracking Database (CTD) Number: 116201672

Source Document: CNL-14-068 [L44150129001] Supplement 1 to LAR TS-485 Date: 01/29/2015

Existing Commitment Description: (*Attach additional page, as necessary*)

TVA will revise the main steam line radiation alarm response procedures to require valves FCV-1-58 and FCV-1-59 to be opened when a main steam line high radiation alarm is received and the main steam line radiation monitor indicates that main steam radiation is increasing.

Revised Commitment Description:

Since the commitment source, LAR TS-485, was withdrawn, the commitment is also withdrawn/closed.

Summarize Justification for Revising Commitment:

CNL-15-070 [L44150529001] dated 05/29/2015 - Browns Ferry Nuclear Plant, Units 1, 2, and 3 - Withdrawal of Proposed Technical Specification Change to Revise the Leakage Rate Through MSIVs - TS-485 and commitment to revise the Reply to NOV EA-11-252 regarding the ALT pathways by July 15, 2015.

NRC acknowledgement of Withdrawn LAR dated 06/16/2015 [NRC ADAMS Accession No. ML15161A344]
CNL-15-123 [L44150715002] Updated Reply to NOV EA-11-252

Attachments: Commitment Load Sheet, Maximo CTD screenshot, CNL-15-070 [L44150529001], NRC Letter [ML15161A344, and CNL-15-123 [L44150715002]

(*Refer to Attachments 3 (flow diagram) and 4 (description) for an explanation of the commitment change evaluation process.*)

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-09.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Commitment was associated with a LAR that has been withdrawn.

(Attach additional page, as necessary)

☐ Yes Go to 2.2

COMMITMENT CHANGE EVALUATION FORM (CCEF)

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________________

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______________________________

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______________________________

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional page, as necessary)

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☒ Yes Go to Question 3.2.

☐ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☒ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

After several rounds of RAIs and discussions, TVA withdrew the LAR and intends to resolve the NOV using plant modifications under 10 CFR 50.59. Subsequently, NRC acknowledged the Withdrawal of the LAR on 06/16/2015 [NRC ADAMS Accession No. ML15161A344]

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☐ Yes Go to 4.2.

☒ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in CTD. Notify NRC of revised commitment in summary report.

☐ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☒ Yes Go to Question 5.2.

☐ No STOP. You have completed this evaluation. Revise commitment in CTD. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☒ Yes Revise commitment in CTD. Notify NRC of revised commitment in next scheduled summary report.

NRC was notified of LAR withdrawal and NRC concurred with the withdrawal - See Justification.

☐ No Revise commitment in CTD. No NRC notification required.

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Supervisor. List any documents (e.g., procedures, NRC submittals) affected by this change below: (*Attach additional page, as necessary*)

Description	EDMS #
Impact Review not required due to commitment withdrawal/closure	

Prepared by: M. W. Oliver Date: 04/12/2016

APPROVALS

Sign/Print: J. L. Paul Jamie Paul Date: 4/12/16
Responsible Supervisor

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change process has been used, and that this commitment change evaluation justifies the change.

Once approved by Licensing, this form shall be kept for the life of the plant in the EDMS vault and attached to CTD.
EDMS # R08160412 130 CCEF attached to CTD and emailed to RS: Y / N

Sign/Print: M. W. Oliver M W Oliver Date: 04/12/2016
Lic Eng, Lic Prog Mgr, Corporate or Site Lic Mgr

COMMITMENT CHANGE EVALUATION FORM (CCEF)

Commitment Tracking Database (CTD) Number: 116633768

Source Document: L44 150312 004 - BFN Flooding HRR Date: 03/12/2015

Existing Commitment Description: (*Attach additional page, as necessary*)

TVA will determine a resolution to the potential backflow through the diesel generator interior flooding drain lines into the Diesel Generator Buildings during the Local Intense Precipitation (LIP) event by September 28, 2015.

Revised Commitment Description:

If required to protect safety-related function, TVA will revise existing procedures or create new procedures to isolate the diesel generator interior flooding drain lines for the LIP event based on the results of a 2-dimensional hydrologic model and the NRC endorsed NEI 15-05 "Warning Time for Maximum Precipitation Events" by August 31, 2016

Summarize Justification for Revising Commitment:

TVA reviewed multiple options to resolve the potential backflow through the DG interior flooding drain lines, including physical modifications. The review concluded that the resolution of the potential impact to safety-related equipment function due to backflow through the DG interior flooding drain lines would be BFN procedurally defined manual actions to close the exterior building drain isolation valves based on a LIP warning time.

A key factor in the warning time determination is the critical location-specific LIP flood depth and a bounding but realistic site LIP flood hydrograph. The current LIP flood depth and hydrograph is based on a simplistic 1-dimensional hydrological analysis. To ensure the warning time is maximized, the confirmed impact to BFN safety related SSCs is prevented and the number of "false positives" is reduced, a more refined 2-dimensional analysis of the LIP event is needed. This 2-dimensional analysis and, if required, the final warning time analysis and warning procedural development takes more time than originally planned. Therefore the commitment requires revision to establish the selected resolutions and the completion time of the resolution.

(*Refer to Attachments 3 (flow diagram) and 4 (description) for an explanation of the commitment change evaluation process.*)

PART I

1.1 Is the existing commitment located in the Updated Final Safety Analysis Report, Emergency Plan, Quality Assurance Plan, Fire Protection Program, or Security Plan?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the appropriate codified process (e.g., 10 CFR 50.59, 10 CFR 50.71(e), 10 CFR 50.54) to evaluate commitment.

☒ No Go to Part 1.2.

1.2 Does the change to the existing commitment affect compliance with the License Conditions associated with B.5.b Mitigation Strategies?

☐ Yes STOP. Do not proceed with this evaluation. Instead, use the 10 CFR 50.90 process to obtain the necessary NRC approval prior to implementation of the proposed change.

☒ No Go to Part II.

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART II

2.1 Could the change negatively affect the ability of a System, Structure or Component (SSC) to perform its safety function or negatively affect the ability of licensee personnel to ensure the SSC is capable of performing its intended safety function or could the change negatively affect the ability to comply with existing commitments associated with B.5.b Mitigation Strategies?

Refer to NPG-SPP-09.3, "Plant Modifications and Engineering Change Control," which provides topics for evaluating if the change affects the ability of an SSC to perform its safety function.

☒ No Continue with Part III.

Briefly describe rationale: Commitment to mitigate the effects of a Beyond Design Basis Event.No potential impact to design basis SSCs.

(Attach additional page, as necessary)

☐ Yes Go to 2.2

2.2 Perform a safety evaluation using the following 10 CFR 50.92 criteria to determine if a significant hazards consideration exists:

Does the revised commitment involve a significant increase in the probability or consequences of an accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment create the possibility of a new or different kind of accident from any accident previously evaluated?

☐ Yes ☐ No

Basis: ______

Does the revised commitment involve a significant reduction in a margin of safety?

☐ Yes ☐ No

Basis: ______

If any of the above questions are answered Yes, STOP. Do not proceed with the revision, OR discuss change with NRC and obtain any necessary approvals prior to implementation of the proposed change.

If all three questions are answered No, go to Part III.

(Attach additional page, as necessary)

PART III

3.1 Was the original commitment (e.g., response to NOV, LER) to restore an OBLIGATION (e.g., rule, regulation, order, or license conditions)?

☐ Yes Go to Question 3.2.

☒ No Go to Part IV.

3.2 Is the proposed revised commitment date necessary and justified?

☐ Yes Briefly describe rationale (attach additional sheets as necessary) and notify NRC of revised commitment date prior to the original commitment date.

☐ No STOP. Do not proceed with revision, OR apply for appropriate regulatory relief.

COMMITMENT CHANGE EVALUATION FORM (CCEF)

PART IV

4.1 Was the original commitment: (1) explicitly credited as the basis for a safety decision in an NRC SER, (2) made in response to an NRC Bulletin or Generic Letter, or (3) made in response to a request for information under 10 CFR 50.54(f) or 10 CFR 2.204?

☒ Yes Go to 4.2.

☐ No Go to Part V.

4.2 Has the original commitment been implemented?

☐ Yes STOP. You have completed this evaluation. Revise commitment in CTD. Notify NRC of revised commitment in summary report.

☒ No STOP. Timely notify NRC of the change (as soon as practicable after the change is approved by management, and before any committed completion date).

PART V

5.1 Was the original commitment made to minimize recurrence of a condition adverse to quality (e.g., a long-term corrective action stated in an LER)?

☐ Yes Go to Question 5.2.

☐ No STOP. You have completed this evaluation. Revise commitment in CTD. No NRC notification required.

5.2 Is the revised commitment necessary to minimize recurrence of the condition adverse to quality?

☐ Yes Revise commitment in CTD. Notify NRC of revised commitment in next scheduled summary report.

☐ No Revise commitment in CTD. No NRC notification required.

IMPACT REVIEW

Conduct a commitment change "impact review" in conjunction with the Responsible Supervisor. List any documents (e.g., procedures, NRC submittals) affected by this change below: (*Attach additional page, as necessary*)

Description	EDMS #
BFN Flooding Hazard Reevaluation Report, R1	B41 150312 003

Prepared by: Kyle Bianco Date: 09/08/2015

APPROVALS

Sign/Print: Kevin D Dutton
Responsible Supervisor Date: 9/18/2015

Licensing signature indicates that the relevant licensing basis has been considered, the appropriate licensing change

COMMITMENT CHANGE EVALUATION FORM (CCEF)

process has been used, and that this commitment change evaluation justifies the change.

Once approved by Licensing, this form shall be kept for the life of the plant in the EDMS vault and attached to CTD.
EDMS # L44 150918 001   CCEF attached to CTD and emailed to RS: Y / N

Sign/Print: Russell R Thompson Russell R Thompson Date: 9/18/15
Lic Eng, Lic Prog Mgr, Corporate or Site Lic Mgr