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Kevin Bronson
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JAFP-11-0046
April 6, 2011

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

Subject: 2011 Updated Final Safety Analysis Report, Technical Specification Bases and Technical Requirements Manual Changes Transmittal
James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
License No. DPR-059

Dear Sir or Madam:

The changes to the Final Safety Analysis Report (FSAR) for the James A. FitzPatrick (JAF) Nuclear Power Plant are being submitted as required by 10 CFR 50.71(e).

This submittal also includes the changes made to the JAF Technical Specifications Bases and the Technical Requirements Manual, which are controlled under 10 CFR 50.59, and submitted to the NRC biennially with the changes to the UFSAR.

The changes and their bases are summarized in Attachment 1, 2, and 3. The changed pages are included in Enclosures 1, 2, and 3 respectively.

There are no commitments contained in this letter. If you have any questions, please contact Mr. Joseph Pechacek, Licensing Manager, at 315-349-6766.

I declare under penalty of perjury that the foregoing is true and correct to the best of my knowledge. Executed on this 6th day of April 2011.

Very truly yours,

Kevin Bronson

Kevin Bronson
Site Vice President

A053
NRR

KB/JP/mh

Attachments	1:	Table of Final Safety Analysis Report (FSAR) 2011 Changes
	2:	Table of Technical Specification (TS) Bases 2011 Changes
	3:	Table of Technical Requirements Manual (TRM) 2011 Changes
Enclosures	1:	Final Safety Analysis Report (FSAR) 2011 Change Pages
	2:	Technical Specification (TS) Bases 2011 Change Pages
	3:	Technical Requirements Manual (TRM) 2011 Change Pages

cc: Regional Administrator, Region I
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ATTACHMENT 2

Table of Technical Specification (TS) Bases 2011 Changes
(1 Page)

Table of Technical Specification (TS) Bases 2011 Changes

TS Bases Page	Description of Change	Justification for Change
B3.7.2-2, B3.7.2-3, B3.7.2-6	Revise Bases Section 3.7.2, Ultimate Heat Sink, for the de-icing heaters to reflect vendor tolerance on heater output value.	Calculation JAF-CALC-09-00005 Rev. 0
B 3.7.4-5	Revise Surveillance Requirement 3.7.4.1 Bases to reflect different methods of meeting the surveillance.	50.59 Screen
B 3.5.1-11	Revise Surveillance Requirement 3.5.1.5 Bases to be consistent with the NUREG 1433 Bases.	50.59 Screen
B 3.5.1-16, B 3.5.1-17	Revise Automatic Depressurization System (ADS) Bases to implement amendment.	Technical Specification Amendment 297 Safety Evaluation Report
B 3.3.6.1-21, B 3.3.6.1-22	Revise Shutdown Cooling Instrumentation Bases to implement amendment.	Technical Specification Amendment 298 Safety Evaluation Report
B 3.8.1-20, B 3.8.1-21, B 3.8.1-22, B 3.8.1-23, B 3.8.1-24, B 3.8.1-25, B 3.8.1-26	Clarify bases for the brake horse power rating for CSP Motor in SR 3.8.1.3 Bases. Note: Actual change was on page B 3.8.1-20 the remaining pages were revised due to text roll.	50.59 Screen
B 3.8.4-2	Revise Bases Section 3.8.4, DC Sources – Operating to revise equipment description based on the EC which replaced the A LPCI Inverter.	50.59 Screen
B 3.4.3-1	Revise Bases Section 3.4.3, Safety / Relief Valves to describe both 2-Stage and 3-Stage pilot assemblies.	Technical Specification Amendment 297 Safety Evaluation Report

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ATTACHMENT 3

Table of Technical Requirements Manual (TRM) 2011 Changes
(2 Pages)

Table of Technical Requirements Manual (TRM) 2011 Changes

TRM Page / Table	Description of Change	Justification for Change
3.3.C-2	Revise TRO 3.3.C, PAM Instrumentation, Condition D Completion Time.	Administrative Change
B 3.7.F-1	Revise Bases for TRO 3.7.F, Explosive Gas Monitoring, to correctly describe recombiner operation.	50.59 Screen
A-7, A-13, A-14, A-21, A-23, A-45, and A-51	Revise Appendix A, Primary Containment Isolation Valves, to 1) reflect changes associated with the Drywell Cams replacement; 2) be consistent with designations on plant drawing FM-94A (Penetration 16X-45 is isolated by 16-1LRA-401 not 27CAD-401); and 3) add 20RDW-939.	Modification JD-01-129; Drawing FM-49A; Engineering Change EC-4586
5.0-17	Revise TRO 5.5.M, Configuration Risk Management Program, to reflect current work management procedure.	Change reflects replacement of site procedure with a fleet procedure.
3.7.P-1, E-2	Revise TRO 3.7.P, EDG Fuel Oil Storage, Sampling Frequency and Remove redundant sampling requirements in Appendix E.	Chemistry Technical Information Document CTID-08-001
5.0-11	Revise TRO 5.5.G, Inservice Testing (IST) Program, to reflect the current IST Implementing document.	Engineering Program Document SEP-IST-007

Table of Technical Requirements Manual (TRM) 2011 Changes

TRM Page / Table	Description of Change	Justification for Change
B3.7.C-2	Revise Bases for TRO 3.7.C, Crescent Area Ventilation System, to include information regarding performance criteria for establishing operability of Crescent area coolers.	50.59 Screen
B 3.3.L-1	Revise Bases for TRO 3.3.L, EDG Instrumentation, to provide background information.	50.59 Screen
Appendix H	Replace Appendix H, Off-Site Dose Calculation Manual with latest revision.	50.59 Screen
3.3.A-2, B 3.3.A-1	Revise TRO 3.3.A, RHR Shutdown Cooling and Bases to reflect changes in instrumentation approved by amendment.	Technical Specification Amendment 298 Safety Evaluation Report
Appendix G	Replace Appendix G with COLR Rev 25.	50.59 Screen and Technical Specification Amendment 299 Safety Evaluation Report Note: The Amendment addressed the changes in the Safety Limit Minimum Critical Power Ratio, the balance of the changes were evaluated under 10 CFR 50.59.

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Enclosure 2

Technical Specification (TS) Bases 2011 Change Pages

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JAMES A. FITZPATRICK NUCLEAR POWER PLANT
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B 3.6.1.2-7	0	B 3.6.2.1-1	0	B 3.7.1-1	0
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B 3.8.1-15	0	B 3.8.5-3	14	B 3.9.7-3	0
B 3.8.1-16	0	B 3.8.5-4	14	B 3.9.7-4	0
B 3.8.1-17	0	B 3.8.6-1	0	B 3.9.8-1	0
B 3.8.1-18	0	B 3.8.6-2	0	B 3.9.8-2	0
B 3.8.1-19	0	B 3.8.6-3	0	B 3.9.8-3	0
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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5.e. Reactor Vessel Water Level — Low (Level 3) (continued)

352.56 inches above the lowest point in the inside bottom of the RPV and also corresponds to the top of a 144 inch fuel column (Ref. 13).

This Function isolates both RWCU suction valves and the RWCU return valve.

5.f. Drywell Pressure — High

High drywell pressure can indicate a break in the RCPB inside the primary containment. The isolation of some of the primary containment isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR 100 are not exceeded. The Drywell Pressure — High Function, associated with isolation of the primary containment, is implicitly assumed in the UFSAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Four channels of Drywell Pressure — High 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 as low as possible without inducing spurious trips. The Allowable Value is chosen to be the same as the RPS Drywell Pressure — High Allowable Value (LCO 3.3.1.1), since this may be indicative of a LOCA inside primary containment.

This Function isolates both RWCU suction valves and one RWCU return valve.

6.a. Reactor Pressure — High

The Reactor Pressure — High Function is provided to isolate the shutdown cooling portion of the Residual Heat Removal (RHR) System. This interlock Function is provided only for equipment protection to prevent an intersystem LOCA scenario, and credit for the interlock is not assumed in the accident or transient analysis in the UFSAR.

The Reactor Pressure — High signals are initiated from two transmitters that are connected to different condensing chambers.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6.a. Reactor Pressure — High (continued)

Each transmitter senses reactor pressure and provides input to each trip system. However, only one channel input is required to be OPERABLE for a trip system to be considered OPERABLE. Two channels of Reactor Pressure — High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. The Function is only required to be OPERABLE in MODES 1, 2, and 3, since these are the only MODES in which the reactor can be pressurized; thus, equipment protection is needed.

The Allowable Value was chosen to be low enough to protect the system equipment from overpressurization.

This Function isolates both RHR shutdown cooling pump suction valves.

6.b. Reactor Vessel Water Level — Low (Level 3)

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 reactor vessel interfaces occurs to begin isolating the potential sources of a break. The Reactor Vessel Water Level — Low (Level 3) Function associated with RHR Shutdown Cooling System isolation is not directly assumed in safety analyses because a break of the RHR Shutdown Cooling System is bounded by breaks of the reactor water recirculation system and MSL. The RHR Shutdown Cooling System isolation on Level 3 supports actions to ensure that the RPV water level does not drop below the top of the active fuel during a vessel draindown event caused by a leak (e.g., pipe break or inadvertent valve opening) in the RHR Shutdown Cooling System.

Reactor Vessel Water Level — Low (Level 3) signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels (two channels per trip system) of the Reactor Vessel Water Level — Low (Level 3) Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. As noted (footnote (e) to Table 3.3.6.1-1), only one trip system of the Reactor Vessel Water Level — Low (Level 3) Function are required to

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 Safety/Relief Valves (S/RVs)

BASES

BACKGROUND

The ASME Boiler and Pressure Vessel Code (Ref. 1) requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of S/RVs are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB).

The S/RVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. Each S/RV discharges steam through a discharge line to a point below the minimum water level in the suppression pool.

The S/RVs can actuate by either of two modes: the safety mode or the relief mode. However, for the purposes of this LCO, only the safety mode is required. Actuation of the S/RV in the safety mode occurs as follows:

2-Stage S/RV: The spring loaded pilot valve opens when steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. Opening the pilot valve allows a pressure differential to develop across the main valve piston and opens the main valve.

3-Stage S/RV: As pressure increases, the expanding bellows pulls the connecting rod attached to the pilot disk, eliminating an abutment gap between the rod and disk. At the valve set pressure, the disk lifts off the seat and pressure is ported to the second stage piston chamber. The force on the piston pushes the second stage disk off its seat. Opening the second stage valve allows a pressure differential to develop across the main valve piston and opens the main valve.

Both of these actuation types satisfy the Code requirement.

Each S/RV can be opened manually in the relief mode from the control room by its associated two-position switch. If one of these switches is placed in the open position the logic output will energize the associated S/RV solenoid control valve directing the pneumatic

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.3 (continued)

supply. The 31 day Frequency takes into consideration administrative controls over operation of the pneumatic system and alarms for low pneumatic pressure.

SR 3.5.1.4

Verification every 31 days that the RHR System cross tie valves are closed and power to the motor operated valve is disconnected ensures that each LPCI subsystem remains independent and a failure of the flow path in one subsystem will not affect the flow path of the other LPCI subsystem. Acceptable methods of removing power to the operator include de-energizing breaker control power or racking out or removing the breaker. If one or more of the RHR System cross tie valves are open or power has not been removed from the motor operated valve, both LPCI subsystems must be considered inoperable. In addition, plant procedures require the motor operated cross tie valve to be chain-locked closed and the manual cross tie valve to be locked closed. The 31 day Frequency has been found acceptable, considering that these valves are under strict administrative controls that will ensure the valves continue to remain closed with either control or motive power removed.

SR 3.5.1.5

Verification every 31 days that each LPCI inverter output has a voltage of ≥ 576 V and ≤ 624 V while supplying its respective bus demonstrates that the AC electrical power is available to ensure proper operation of the associated LPCI injection and heat exchanger bypass valves and the recirculation pump discharge valve. Each inverter must be OPERABLE for the associated LPCI subsystem to be OPERABLE. The 31 day Frequency has been found acceptable based on operating experience.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.1.12 (continued)

The Frequency of 24 months is acceptable, given plant conditions required to perform the test and the other requirements existing to ensure adequate LPCI inverter performance during the 24 month interval. In addition, the Frequency is intended to be consistent with expected fuel cycle lengths.

SR 3.5.1.13

Valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME Code requirements, prior to valve installation. Actuation of each required ADS valve is performed to verify that mechanically the valve is functioning properly. For both two-stage and three-stage S/RVs, tests are required to demonstrate:

- That each ADS S/RV solenoid valve ports pneumatic pressure to the associated S/RV actuator when energized;
- That each ADS S/RV pilot stage actuates to open the associated main stage when the pneumatic actuator is pressurized; and
- That each ADS S/RV main stage opens and passes steam when the associated pilot stage actuates.

The solenoid valves are functionally tested once per cycle as part of the Inservice Testing Program. The actuators and main stages are bench tested, together or separately, as part of the certification process, at intervals determined in accordance with the Inservice Testing Program. Maintenance procedures ensure that the S/RV actuators and main stages are correctly installed in the plant, and that the S/RV and associated piping remain clear of foreign material that might obstruct valve operation or full steam flow. This approach provides adequate assurance that the required ADS valves will operate when actuated, while minimizing the challenges to the valves and the likelihood of leakage or spurious operation. While two-stage actuator assemblies are not tested in-situ due to a high probability of causing unseating or leakage of the pilot stage which can lead to spurious actuation or failure to reclose, installed three-stage actuator

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.13 (continued)

assemblies are dry lift tested after installation. SR 3.5.1.11 and the LOGIC SYSTEM FUNCTIONAL Test performed in LCO 3.3.5.1 overlap this Surveillance to provide complete testing of the assumed safety function.

REFERENCES

1. UFSAR, Section 6.4.3.
2. UFSAR, Section 6.4.4.
3. UFSAR, Section 6.4.1.
4. UFSAR, Section 6.4.2.
5. NEDC-31317P, Revision 2, James A. FitzPatrick Nuclear Power Plant SAFER/GESTR-LOCA, Loss-of-Coolant Accident Analysis, April 1993.
6. UFSAR, Section 14.6.1.5.
7. UFSAR, Section 14.6.1.3.
8. 10 CFR 50, Appendix K.
9. UFSAR, Section 6.5.
10. 10 CFR 50.46.
11. 10 CFR 50.36(c)(2)(ii).
12. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), Recommended Interim Revisions to LCOs for ECCS Components, December 1, 1975.
13. UFSAR, Section 4.4.5.

BASES

BACKGROUND (continued)

side openings that are protected with bar racks spaced at 1 ft centers to block the entrance of large debris. This results in water being taken in at lower levels and prevents the formation of vortices at the surface, thus minimizing the possibility of floating ice being drawn down from the surface. The side intake area of approximately 8 ft by 70 ft, less bar rack area, provides a net clear area of 552 ft^2. During normal operation, with a maximum nominal operating flow of 388,600 gpm from three circulating water pumps and two normal service water pumps, the average intake velocity is approximately 1.6 ft per second across the intake bar racks. However, during safe shutdown conditions with only two Residual Heat Removal Service Water (RHRSW) pumps and one ESW pump in operation, the maximum nominal flow is reduced to 10,000 gpm, corresponding to an average intake velocity of 0.04 ft per second.

The formation of frazil ice on the steel bar racks at the intake structure openings is common in northern climates. This kind of ice is formed when meteorological conditions are such that the water is supercooled below its freezing point due to radiational cooling. Under these conditions, frazil ice can form on intake bar racks or spongy masses of this ice, formed in other parts of the lake and carried past an intake by wind-driven currents, can adhere to the bar racks. Sufficient transport velocity exists to move buoyant frazil ice from the lake surface to the intake structure during normal operation, but not under safe shutdown conditions. If ice formation does occur on the bar racks during normal operation, sufficient local erosion velocities will develop to limit total ice accumulation such that the remaining net clear intake area would be sufficient to meet required safe shutdown flows. In an effort to suppress the formation of frazil ice on the bar racks, each of the 88 rack bars is heated by a deicing heater. Each deicing heater is rated at 1670 ± 10% watts and is normally energized. Forty four heaters are powered by one division while the remaining 44 heaters are powered by the other division. The deicing heaters are not adequately sized to prevent frazil ice formation under extreme supercooling conditions, although they will minimize frazil ice formation most of the time.

APPLICABLE SAFETY ANALYSES

Since Lake Ontario is the UHS, sufficient water inventory is available for all ESW System post LOCA cooling requirements for a 30 day period. The OPERABILITY of the ESW System is assumed in evaluations of the equipment required for safe reactor shutdown

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

presented in the UFSAR, Chapters 5 and 14 (Refs. 2 and 3, respectively). These analyses include the evaluation of the long term primary containment response after a design basis LOCA.

The ability of the ESW System to provide adequate cooling to the identified safety equipment is an implicit assumption for the safety analyses evaluated in References 2 and 3. The ability to provide onsite emergency AC power is dependent on the ability of the ESW System to cool the EDGs. The long term cooling capability of RHR and core spray pumps is dependent on the capability of the ESW System to provide cooling to the EDGs as well as the crescent area coolers.

The ESW System, together with the UHS, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).

LCO

The ESW subsystems are independent of each other to the degree that each has separate controls, power supplies, and the operation of one does not depend on the other. In the event of a DBA, one subsystem of ESW is required to provide the minimum heat removal capability assumed in the safety analysis for the system to which it supplies cooling water. To ensure this requirement is met, two subsystems of ESW must be OPERABLE. At least one subsystem will operate, if the worst single active failure occurs coincident with the loss of offsite power.

A subsystem is considered OPERABLE when it has an OPERABLE UHS, one OPERABLE pump, and an OPERABLE flow path capable of taking suction from the intake structure and transferring the water to the appropriate equipment. OPERABILITY of equipment cooled by the ESW System is based on heat transfer, not flow rates; OPERABILITY of the ESW pumps is based on measured performance remaining within allowable IST Program acceptance criteria.

The OPERABILITY of the UHS is based on having a minimum water level in the screenwell of 236.5 ft mean sea level and a maximum water temperature of 85 °F. With UHS temperature ≤ 37 °F, conditions become increasingly favorable for the formation of frazil ice on the intake structure bar racks during normal operation. Therefore, in an effort to suppress the formation of frazil ice on the intake structure bar racks, at least 18 out of the 44 deicing heaters (each heater producing 1670 ± 10% watts) in each electrical division are maintained OPERABLE whenever UHS temperature is ≤ 37 °F.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.2.3, SR 3.7.2.5, and SR 3.7.2.6

These SRs are modified by a NOTE indicating that these SRs are not required to be met if UHS temperature is > 37°F. Industry experience has shown that frazil ice will not adhere to the bar racks that are above freezing temperatures. Therefore at these elevated temperatures, blockage of the intake is unlikely and the deicing heaters are not required to be OPERABLE.

Verification of the required deicing feeder current in SR 3.7.2.3 and the required deicing heater power in SR 3.7.2.5 will help ensure that adequate heat is being provided at the bar racks to help ensure that frazil ice does not adhere to them. Verification of the required deicing heater resistance to ground in SR 3.7.2.6 is performed to monitor long term degradation of the cable and heater insulations. SR 3.7.2.3 can be performed by measuring the current in all three phases of the feeder cables to each division and ensuring the total current is within limits to confirm that at least 18 deicing heaters are OPERABLE in each division. SR 3.7.2.5 is performed to verify that at least 18 deicing heaters in each division are each dissipating at least 1503 watts (from Vendor Specification 1670 ±10% watts). The 7 day Frequency of SR 3.7.2.3 and the 6 month Frequency of SR 3.7.2.5 is based on operating experience that shows the heaters are reliable. The 12 month Frequency of SR 3.7.2.6 has shown that the components usually pass the SR when performed at the 12 month Frequency. Therefore, this Frequency is considered to be acceptable from a reliability standpoint.

SR 3.7.2.4

Verifying the correct alignment for each manual, power operated, and automatic valve in each ESW subsystem flow path provides assurance that the proper flow paths will exist for ESW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position, and yet considered in the correct position, provided it can be automatically realigned to its accident position within the required time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

If applicable, handling of recently irradiated fuel in the secondary containment must be suspended immediately. Suspension of this activity shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the heat removal capability of the system is sufficient to remove the control room heat load assumed in the safety analyses with ESW providing water to the cooling coils of the air handling units. Heat transfer testing is not performed on the Control Room (CR) and Relay Room (RR) Air Handlin Units (AHUs) as these coolers are closed loop, glycol based systems which are not prone to fouling. To verify the system has the capability to remove the assumed heat, the ESW supply function (safety related) is required to be operable and the following surveillance requirements met: 1) the manual valves needed to initiate ESW flow to these coolers are cycled to verify operability; 2) the ESW supply piping to the AHUs is flushed during check valve testing; and 3) flow rates are measured against target flow rates. Therefore, any degradation would be detected and corrected through the corrective action program. The 24-month frequency is appropriate since significant degradation of the Control Room AC System is not expected over this time period.

JAF calculations verify maximum Allowable Tube Plugging Limit for CR and RR AHUs if maintenance is required on the AHUs. The level of allowed plugging provides a margin in the CR and RR equipment heat load and still maintains the CR and RR below 104°F under accident conditions using ESW at 85°F. In addition, JAF calculations state the potential for plugged tubes is low crediting use of a closed loop cooling water system using glycol/demineralized water (not service water) as the cooling medium.

REFERENCES

1. UFSAR, Section 9.9.3.11.
2. 10 CFR 50.36(c)(2)(ii).
3. SEP-SW-001 Rev.0, NRC Generic Letter 89-13 Service Water Program

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.8

Each EDG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the EDG subsystem capability to reject the largest single load without exceeding a predetermined frequency and while maintaining a specified margin to the overspeed trip. The largest single load for each EDG subsystem is a core spray pump (1250 bhp motor rating actual load will depend on accident progression). This Surveillance may be accomplished by:

a. Tripping the EDG output breakers with the EDG subsystem carrying greater than or equal to its associated single largest post-accident load while paralleled with normal, reserve, or backfeed power, or while solely supplying the bus; or

b. Tripping its associated single largest post-accident load with the EDG subsystem solely supplying the bus.

Consistent with Safety Guide 9 (Ref. 3), the load rejection test is acceptable if the diesel speed does not exceed the nominal (synchronous) speed plus 75% of the difference between nominal speed and the overspeed trip setpoint, or 115% of nominal speed, whichever is lower.

The Frequency of 24 months, takes into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. In order to ensure that the EDG subsystem is tested under load conditions that are as close to design basis conditions as possible, the Note requires that, if paralleled with normal, reserve or backfeed power, testing must be performed using a power factor ≤ 0.9. This power factor is chosen to be representative of the actual design basis inductive loading that the EDG subsystem would experience. However, if the grid conditions do not permit, the power factor limit is not required to be met. In this condition the test is performed with a power factor as close to the design rating of the machine as practicable. This is permitted since, with a high grid voltage it may not be possible to raise the EDG

(continued)

SURVEILLANCE REQUIREMENTS (continued)

subsystem output voltage sufficiently to obtain the required power factor without creating an overvoltage condition on the emergency bus.

SR 3.8.1.9

Consistent with Regulatory Guide 1.108 (Ref. 9), paragraph 2.a.(1), this SR demonstrates the as designed operation of the onsite power sources due to an emergency bus loss of power (LOP) signal. This test verifies all actions required following receipt of the LOP signal, including shedding of the nonessential loads and energization of the emergency buses and respective loads from the EDG subsystem. It further demonstrates the capability of the EDG subsystem to automatically achieve the required voltage and frequency within the specified time.

The EDG auto-start time of 11 seconds is derived from requirements of the accident analysis for responding to a design basis large break LOCA. The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate that all starting transients have decayed and stability has been achieved.

The requirement to verify the connection and power supply of permanent and auto-connected loads is intended to satisfactorily show the relationship of these loads to the EDG subsystem loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, Emergency Core Cooling Systems (ECCS) injection valves are not desired to be stroked open, or systems are not capable of being operated at full flow, or RHR systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of the connection and loading of these loads, testing that adequately shows the capability of the EDG subsystem to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 24 months, takes into consideration plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.1.9 (continued)

This SR is modified by a Note. The reason for the Note is to minimize the wear and tear on the EDGs during testing. For the purpose of this testing, the EDGs must be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations.

SR 3.8.1.10

This SR demonstrates that the EDG subsystem automatically starts, force parallels and achieves the required voltage and frequency within the specified time (10 seconds) from the design basis actuation signal (LOCA signal) and operates for ≥ 5 minutes. The 5 minute period provides sufficient time to demonstrate stability. SR 3.8.1.10.d and SR 3.8.1.10.e ensure that permanently connected loads and emergency loads are energized from the offsite electrical power system on a LOCA signal without a LOP signal.

The requirement to verify the connection and power supply of permanent and auto-connected loads is intended to satisfactorily show the relationship of these loads to the loading logic for loading onto offsite power. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, ECCS injection valves are not desired to be stroked open, systems are not capable of being operated at full flow, or RHR systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of the connection and loading of these loads, testing that adequately shows the capability of the EDG subsystem to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

In addition to the SR requirements, the time for the EDG subsystem to reach steady state operation is periodically monitored and the trend evaluated to identify degradation of governor and voltage regulator performance.

The Frequency of 24 months takes into consideration plant

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.1.10 (continued)

conditions required to perform the Surveillance and is intended to be consistent with the expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

This SR is modified by a Note. The reason for the Note is to minimize the wear and tear on the EDGs during testing. For the purpose of this testing, the EDGs must be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations.

SR 3.8.1.11

Consistent with IEEE-387 (Ref. 13), Section 7.5.9 and Table 3, this SR requires demonstration that the EDGs can run continuously at full load capability for an interval of not less than 8 hours — 6 hours of which is at a load equivalent to 90-100% of the continuous rating of the EDG, and 2 hours of which is at a load equivalent to 105% to 110% of the continuous duty rating of the EDG. The EDG starts for this Surveillance can be performed either from standby or hot conditions. The provisions for gradual loading, discussed in SR 3.8.1.3, are applicable to this SR.

In order to ensure that the EDG subsystem is tested under load conditions that are as close to design conditions as possible, testing must be performed using a power factor ≤ 0.9. This power factor is chosen to be representative of the actual design basis inductive loading that the EDG subsystem could experience. A load band is provided to avoid routine overloading of the EDG subsystem. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain EDG OPERABILITY.

The 24 month Frequency is consistent with the recommendations of IEEE-387 (Ref. 13), Section 7.5.9 and Table 3 which takes into consideration plant conditions required to perform the Surveillance; and is intended to be consistent with expected fuel cycle lengths.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.8.1.11 (continued)

This Surveillance is modified by two Notes. Note 1 states that momentary transients due to changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the limit do not invalidate the test. Note 2 is provided in recognition that when grid conditions do not permit, the power factor limit is not required to be met. In this condition, the test is performed with a power factor as close to the design rating of the machine as practicable. This is permitted since, with a high grid voltage it may not be possible to raise the EDG output voltage sufficiently to obtain the required power factor without creating an overvoltage condition on the emergency bus.

SR 3.8.1.12

In the event of a DBA coincident with an emergency bus loss of power signal, the EDGs are required to supply the necessary power to Engineered Safeguards systems so that the fuel, RCS, and containment design limits are not exceeded.

This SR demonstrates EDG subsystem operation, as discussed in the Bases for SR 3.8.1.9, during an emergency bus LOP signal in conjunction with an ECCS initiation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the EDG subsystem to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 24 months takes into consideration plant conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of 24 months.

This SR is modified by a Note. The reason for the Note is to minimize the wear and tear on the EDGs during testing. For the purpose of this testing, the EDGs must be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.13

Under accident conditions loads are sequentially connected to the bus by the individual time delay relays. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the EDGs due to high motor starting currents. The minimum load sequence time interval tolerance ensures that sufficient time exists for the EDG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding engineered safeguards equipment time delays are not violated. There is no upper limit for the load sequence time interval since, for a single load interval (i.e., the time between two load blocks), the capability of the EDG to restore frequency and voltage prior to applying the second load is not negatively affected by a longer than designed load interval, and if there are additional load blocks (i.e., the design includes multiple load intervals), then the lower limit requirements will ensure that sufficient time exists for the EDG to restore frequency and voltage prior to applying the remaining load blocks (i.e., all load intervals must be greater than or equal to the minimum design interval).

The Frequency of 24 months takes into consideration plant conditions required to perform the Surveillance and is intended to be consistent with expected fuel cycle lengths.

REFERENCES

1. UFSAR, Section 16.6.
2. UFSAR, Chapter 8.
3. Safety Guide 9, Selection Of Diesel Generator Set Capacity For Standby Power Supplies, March 1971.
4. UFSAR, Chapter 6.
5. UFSAR, Chapter 14.
6. 10 CFR 50.36(c)(2)(ii).
7. Generic Letter 84-15, Proposed Staff Actions To Improve And Maintain Diesel Generator Reliability, July 1984.

(continued)

REFERENCES (continued)

8. Regulatory Guide 1.93, Availability Of Electric Power Sources, December 1974.

9. Regulatory Guide 1.108, Revision 1, Periodic Testing of Diesel Generator Units Used As Onsite Electric Power Systems At Nuclear Power Plants, August 1977.

10. Regulatory Guide 1.137, Revision 1, Fuel-Oil Systems for Standby Diesel Generators, October 1979.

11. ANSI C84.1, Voltage Ratings for Electric Power Systems and Equipment, 1982.

12. UFSAR, Section 6.5.

13. IEEE-387, IEEE Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations, 1995.

BASES

BACKGROUND (continued)

Each 419 VDC LPCI MOV independent power supply subsystem is energized by the associated 419 VDC battery or the associated 419 VDC rectifier/charger. Each battery and rectifier/charger is exclusively associated with a 419 VDC LPCI MOV independent power supply subsystem and cannot be interconnected with the other 419 VDC LPCI MOV independent power supply subsystem.

During normal operation, the DC loads are powered from the battery chargers with the batteries floating on the system. In cases where momentary loads are greater than the charger capability, or battery charger output voltage is low, or on loss of normal power to the battery charger, the DC loads are automatically powered from the batteries. Also, on a LPCI automatic actuation signal, the 419 VDC rectifier/charger AC input isolation devices will open and the 600 VAC LPCI independent power supply inverters will be powered from the 419 VDC LPCI MOV independent power supply batteries.

The DC power distribution system is described in more detail in Bases for LCO 3.8.7, "Distribution System - Operating," and LCO 3.8.8, "Distribution System - Shutdown."

Each 125 VDC and 419 VDC battery is separately housed in a ventilated room apart from its charger and distribution centers. Each subsystem is located in an area separated physically and electrically from its redundant subsystem to ensure that a single failure in one subsystem does not cause a failure in the redundant subsystem. There is no sharing between redundant subsystems such as batteries, battery chargers, or distribution panels.

Each 125 VDC battery has adequate storage capacity to meet the duty cycle(s) discussed in the UFSAR, Chapter 8 (Ref. 4). The battery is designed with additional capacity above that required by the design duty cycle to allow for temperature variations and other factors. Each 419 VDC LPCI MOV independent power supply battery has adequate storage capacity for one repositioning of the LPCI subsystem motor operated valves (MOVs) on its respective MOV bus.

The 125 VDC batteries are sized to supply associated DC loads required for safe shutdown of the plant, following abnormal operational transients and postulated accidents, until AC power sources are restored (Ref. 4). The 419 VDC batteries are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. The minimum design voltage limit for

(continued)

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Enclosure 3

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SURVEILLANCE REQUIREMENTS

-----------------------------------NOTE-----------------------------------
When a channel is placed in an inoperable status solely for performance of the required Surveillance, entry into the associated Conditions and Required Actions may be delayed for up to 6 hours.

SURVEILLANCE		FREQUENCY
TRS 3.3.A.1	Perform CHANNEL CALIBRATION. The allowable value is $\geq$ [41] psig and $\leq$ [75] psig.	24 months
TRS 3.3.A.2	Perform CHANNEL CHECK	12 hours
TRS 3.3.A.3	Perform CHANNEL FUNCTIONAL TEST	92 days
TRS 3.3.A.2	CALIBRATE the Trip Units	184 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2.1	Restore required channel to OPERABLE status.	7 days
	OR		
	B.2.2	Initiate a Condition Report.	7 days
C. ---------NOTE---------- Only applicable to Function 11. ----------------------- One or more Functions with two required channels inoperable.	C.1	Monitor and log the primary containment H_2 or O_2 parameter(s) using 27PCX-101A or 27PCX-101B. OR Obtain and analyze a grab sample of the primary containment H_2 or O_2.	24 hours AND Once per 24 hours thereafter
	AND		
	C.2.1	Restore one required channel to OPERABLE status.	7 days
	OR		
	C.2.2	Initiate a Condition Report.	7 days
D. For Function 8.b, associated thermocouple inoperable.	D.1	Monitor SRV performance with the associated in service acoustical detector.	24 hours AND Once per 24 hours thereafter

(continued)

3.7 PLANT SYSTEMS

3.7.P Emergency Diesel Generator (EDG) Fuel Oil Storage

TRO 3.7.P EDG fuel oil storage shall be FUNCTIONAL

APPLICABILITY: When the EDGs are required to be OPERABLE

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TRS 3.7.P-1	Manually measure and compare to the reading of the local level indicators, the quantity of EDG fuel oil available in each storage tank.	31 days
TRS 3.7.P-2	Check for quality per Appendix E, Diesel Fuel Oil Properties, a sample of EDG fuel oil in each storage tank.	92 days

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs

5.5.G Inservice Testing Program

Technical Specification 5.5.7, "Inservice Testing Program," requires controls be established for inservice testing of certain ASME Code Class 1, 2, and 3 components. This program is implemented by:

- SEP-IST-007, James A. FitzPatrick Inservice Testing for Pumps and Valves, Fourth Ten-Year Interval Program Section

Programs and Components Engineering Department is responsible for this program.

5.5 ADMINISTRATIVE CONTROLS

5.5 Programs

5.5.M Configuration Risk Management Program (CRMP)

Technical Specification 5.5.13, "Configuration Risk Management Program (CRMP)" requires means be provided for a procedural risk-informed assessment to manage the risk associated with equipment inoperability. This program is implemented by:

- AP-10.09, "Outage Risk Assessment,"
- AP-10.10, "On-Line Risk Assessment," and
- EN-WM-104, "On-Line Risk Assessment."

Planning, Scheduling, and Outage Management Department is responsible for this program.

Table A-1 (page 6 of 42)
Primary Containment Isolation Valves

CONTAINMENT PENETRATION	PENETRATION FUNCTION	VALVE NUMBER	ISOLATION SIGNAL	CLOSE TIME (sec)[a]	NORMAL STATUS[b]	REMARKS
18	Drywell Floor Drain Sump Discharge	20MOV-82	A, F, R	30	Open	
		20AOV-83	A, F, R	NA	Open	
		20RDW-933	NA	NA	Closed	LLRT
19	Drywell Equipment Drain Sump Discharge	20MOV-94	A, F, R	30	Open	
		20RDW-938	NA	NA	Closed	LLRT
		20RDW-939	NA	NA	Closed	LLRT
		20AOV-95	A, F, R	NA	Closed	(h)
20	Service Water	46SWS-53D	Reverse Flow	NA	Open	
22	Instrument Air or Nitrogen	39IAS-22	Reverse Flow	NA	Open	
		27SOV-141	R	NA	Open	(i)
		39IAS-757	NA	NA	Closed	LLRT
23	RBCLCW	15AOV-130A	R	NA	Open	To drywell cooler assembly A and equipment sump cooler
		15RBC-404A 15RBC-710A 15RBC-711A 15RBC-713 15RBC-203A	NA	NA	Closed	LLRT, vents, and drains

(continued)

(a) The standard minimum closing rate for automatic motor operated isolation valves is based on a nominal line size of 12 inch. Using the standard closing rate, a 12 inch line is isolated in 60 seconds.

(b) Normal status position of valve (open or closed) is the position during normal power operation of the reactor and is provided in this table for information only.

(h) Valve 20AOV-95 opens during pump out of the drywell equipment sump. Automatic isolation signals A and F override an open signal that might be present for sump pump out.

(i) Fail in open position to ensure adequate pneumatic supply.

Table A-1 (Page 12 of 42)
Primary Containment Isolation Valves

CONTAINMENT PENETRATION	PENETRATION FUNCTION	VALVE NUMBER	ISOLATION SIGNAL	CLOSE TIME (sec)[a]	NORMAL STATUS[b]	REMARKS
31Ad	Drywell atmosphere sample (suction)	27SOV-135C	A, F, R, Z	NA	Open	From elev. 276′ to Radiation Monitors (k) (m)
		27CAD-794	NA	NA	Closed	LLRT
		27SOV-135A	A, F, R, Z	NA	Open	From elev. 276′ to Radiation Monitors (k) (m)
		27CAD-793	NA	NA	Closed	LLRT
31Ba	Instrumentation	02-2EFV-PS-128A	Excess Flow	NA	Open	EFCV
31Bb	Instrumentation	02-2EFV1-FT-110A	Excess Flow	NA	Open	EFCV

(continued)

(a) The standard minimum closing rate for automatic motor operated isolation valves is based on a nominal line size of 12 inch. Using the standard closing rate, a 12 inch line is isolated in 60 seconds.

(b) Normal status position of valve (open or closed) is the position during normal power operation of the reactor and is provided in this table for information only.

(k) These valves are not considered to be redundant isolation valves. These instrument lines meet the criteria for containment isolation because they satisfy the AEC Safety Guide 11 requirements for instrument lines. Both valves on the penetration are powered by the same power supply train to ensure post-accident sampling capability is maintained.

(m) Radiation Monitors used for sampling particulate and gaseous are as follows:

Radiation Monitors	Sample
17RM-123A 17RM-123B	Particulate
17RM-123A 17RM-123B	Gaseous

Table A-1 (Page 13 of 42)
Primary Containment Isolation Valves

CONTAINMENT PENETRATION	PENETRATION FUNCTION	VALVE NUMBER	ISOLATION SIGNAL	CLOSE TIME (sec)[a]	NORMAL STATUS[b]	REMARKS
31Bd	Drywell atmosphere sample (suction)	27SOV-135D	A, F, R, Z	NA	Open	From elev. 296′ to Radiation Monitors (k) (m)
		27CAD-779	NA	NA	Closed	LLRT
		27SOV-135B	A, F, R, Z	NA	Open	From elev. 296′ to Radiation Monitors (k) (m)
		27CAD-778	NA	NA	Closed	LLRT

(continued)

(a) The standard minimum closing rate for automatic motor operated isolation valves is based on a nominal line size of 12 inch. Using the standard closing rate, a 12 inch line is isolated in 60 seconds.

(b) Normal status position of valve (open or closed) is the position during normal power operation of the reactor and is provided in this table for information only.

(k) These valves are not considered to be redundant isolation valves. These instrument lines meet the criteria for containment isolation because they satisfy the AEC Safety Guide 11 requirements for instrument lines. Both valves on the penetration are powered by the same power supply train to ensure post-accident sampling capability is maintained.

(m) Radiation Monitors used for sampling particulate and gaseous are as follows:

Radiation Monitors	Sample
17RM-123A 17RM-123B	Particulate
17RM-123A 17RM-123B	Gaseous

Table A-1 (Page 20 of 42)
Primary Containment Isolation Valves

CONTAINMENT PENETRATION	PENETRATION FUNCTION	VALVE NUMBER	ISOLATION SIGNAL	CLOSE TIME (sec)[a]	NORMAL STATUS[b]	REMARKS
52a	Drywell atmosphere sample (Return)	27SOV-125C 27SOV-125A	A, F, R, Z	NA	Open	To elev. 282′ from Radiation Monitors (k) (m)
		27CAD-781 27CAD-782	NA	NA	Closed	LLRT
52c	Instrumentation	13EFV-01A	Excess Flow	NA	Open	EFCV
52d	Instrumentation	13EFV-01B	Excess Flow	NA	Open	EFCV
53a	Instrumentation	29EFV-53A	Excess Flow	NA	Open	EFCV

(continued)

(a) The standard minimum closing rate for automatic motor operated isolation valves is based on a nominal line size of 12 inch. Using the standard closing rate, a 12 inch line is isolated in 60 seconds.

(b) Normal status position of valve (open or closed) is the position during normal power operation of the reactor and is provided in this table for information only.

(k) These valves are not considered to be redundant isolation valves. These instrument lines meet the criteria for containment isolation because they satisfy the AEC Safety Guide 11 requirements for instrument lines. Both valves on the penetration are powered by the same power supply train to ensure post-accident sampling capability is maintained.

(m) Radiation Monitors used for sampling particulate and gaseous are as follows:

Radiation Monitors	Sample
17RM-123A 17RM-123B	Particulate
17RM-123A 17RM-123B	Gaseous

Table A-1 (Page 22 of 42)
Primary Containment Isolation Valves

CONTAINMENT PENETRATION	PENETRATION FUNCTION	VALVE NUMBER	ISOLATION SIGNAL	CLOSE TIME (sec)[a]	NORMAL STATUS[b]	REMARKS
55a	Instrumentation	10RHR-60A 10RHR-985 10RHR-986	NA	NA	Closed	LLRT, vents, and drains
55b	Drywell atmosphere sample (Return)	27SOV-125D 27SOV-125B	A, F, R, Z	NA	Open	To elev. 296' from Radiation Monitors (k) (m)
		27CAD-791 27CAD-790	NA	NA	Closed	LLRT
55c	Instrumentation	10RHR-60C 10RHR-983 10RHR-984	NA	NA	Closed	LLRT, vents, and drains
55d	Instrumentation	13EFV-02B	Excess Flow	NA	NA	EFCV
56a	Instrumentation	02-2RWR-795A	NA	NA	Closed	
		02-2EFV-PT-25A	Excess Flow	NA	NA	EFCV

(continued)

(a) The standard minimum closing rate for automatic motor operated isolation valves is based on a nominal line size of 12 inch. Using the standard closing rate, a 12 inch line is isolated in 60 seconds.

(b) Normal status position of valve (open or closed) is the position during normal power operation of the reactor and is provided in this table for information only.

(k) These valves are not considered to be redundant isolation valves. These instrument lines meet the criteria for containment isolation because they satisfy the AEC Safety Guide 11 requirements for instrument lines. Both valves on the penetration are powered by the same power supply train to ensure post-accident sampling capability is maintained.

(m) Radiation Monitors used for sampling particulate and gaseous are as follows:

Radiation Monitors	Sample
17RM-123A 17RM-123B	Particulate
17RM-123A 17RM-123B	Gaseous

Table A-2 (Page 2 of 18)
Primary Containment Valves, Blind Flanges, and Other Bolted or Threaded Connections Located Outside Primary Containment that are Required to be Closed During Accident Conditions

Penetration	Description
16X-6	Manway with bolted flanges (FM-107A, FM-110A, FV-1G, DBD-16A)
16X-7A	29AOV-86A, 29MST-754A (FM-29A)
16X-7B	29AOV-86B, 29MST-753A (FM-29A)
16X-7C	29AOV-86C, 29MST-756A (FM-29A)
16X-7D	29AOV-86D, 29MST-757A (FM-29A)
16X-8	29MOV-77, 29MST-750A (FM-29A)
16X-9A	34FWS-12A, 13RCIC-715, 13RCIC-761, 13MOV-21, 34NRV-111A, 12MOV-69 (FM-34A)
16X-9B	34FWS-12B, 34NRV-111B, 23HPI-708, 23MOV-19, 23HPI-29 (FM-34A, FM-25A)
16X-10	13MOV-16 RCIC-420, 13RCIC-730 (FM-22A)
16X-11	23MOV-16, 23MOV-60, 23HPI-27A (FM-25A)
16X-12	10MOV-17, 10RHR-84 (FM-20A, FP-24A, FP-24J)
16X-13A	10MOV-25A, 10MOV-27A, 10RHR-78A, 10RHR-783A, 10RHR-784A, 10RHR-787A, 10RHR-788A, 10RHR-789A, 10RO-131A (FM-20A)
16X-13B	10MOV-25B, 10MOV-27B, 10RHR-78B, 10RHR-783B, 10RHR-784B, 10RHR-787B, 10RHR-788B, 10RHR-789B, 10RO-131B (FM-20A)
16X-14	12MOV-18, 12RWC-16, 12RWC-18-2 (FM-24A, FM-102A)
16X-16A	14MOV-11A, 14MOV-12A, 14CSP-24A (FM-23A)
16X-16B	14MOV-11B, 14MOV-12B, 14CSP-24B (FM-23A)
16X-17	10RHR-708 (FM-20A, modification F1-92-091, line spec. and pipe spec.)
16X-18	20RDW-933, 20AOV-83 (FM-17A)
16X-19	20AOV-95, 20RDW-939 (FM-17A)

(continued)

Table A-2 (Page 8 of 18)
Primary Containment Valves, Blind Flanges, and Other Bolted or Threaded Connections Located Outside Primary Containment that are Required to be Closed During Accident Conditions

Penetration	Description
16X-45	16-1AOV-101A, 16-1AOV-101B, 16-1LRA-401 (FM-49A, modification F1-84-072, line spec. and pipe spec.)
16X-46	None (DBD-16A, 3.14-32, 3.11-35)
16X-47	None (DBD-16A, 3.14-32, 3.11-35)
16X-48	None (DBD-16A, 3.14.10, 3.11-35)
16X-49a	None - penetration closed by a welded cap inside containment that is an extension of the containment liner and does not require any other isolation barriers (FK-1A, FM-26A)
16X-49b	None - penetration closed by a welded cap inside containment that is an extension of the containment liner and does not require any other isolation barriers (FK-1A, FM-26A)
16X-49c	None - spare penetration (FV-1A)
16X-49d	13EFV-02A (FM-22A)
16X-50a	None - penetration closed by a welded cap inside containment that is an extension of the containment liner and does not require any other isolation barriers (FK-1A, FM-26A)
16X-50b	None - penetration closed by a welded cap inside containment that is an extension of the containment liner and does not require any other isolation barriers (FK-1A, FM-26A)
16X-50c	27CAD-618, 27CAD-619 (FM-18B)
16X-50d	02-3EFV-33 (FM-47A)
16X-51a	None - penetration closed by a welded cap inside containment that is an extension of the containment liner and does not require any other isolation barriers (FK-1A, FM-26A)
16X-51b	39IAS-765 (FM-39C, line spec. and pipe spec.)

(continued)

Table E-1 (page 1 of 1)
Diesel Fuel Oil Properties

	FUNCTION	SETPOINT
1.	Pour Point - °F	10°F maximum
2.	Water and Sediment	0.05% maximum
3.	Distillation 90% Point	540 minimum
4.	Sulfur	1% maximum
5.	Copper Strip Corrosion	No. 3 maximum
6.	Cetane Number	40 minimum

APPENDIX G
CORE OPERATING LIMITS REPORT

Entergy

ENTERGY NUCLEAR OPERATIONS, INC.
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
REPORT

CORE OPERATING LIMITS REPORT
REVISION 25

APPROVED BY: William Drews DATE: 10/6/10
REACTOR ENGINEERING SUPERVISOR

APPROVED BY: DATE: 10/6/2010
GENERAL MANAGER – PLANT OPERATIONS

REVISION RECORD

Revision	Cycle	Date	Description
25	20	Sept. 2010	Cycle 20 Revision

Summary of Changes		
Rev. 25	Effective upon final approval	Applicable for use during Cycle 20 Operation. Revision issued to update this document for FitzPatrick Reload 19 Cycle 20 cycle dependent data.
		Changed MCPR Limits reporting format for $\tau = 0$ and $\tau \neq 0$ to Tables 8.1, 8.2, 8.3, and 8.4. Redundant information contained in Figures was removed.
		APLHGR Limits reporting format no longer uses figure format. Pertinent information is tabulated in Table 8. 5
		LHGR Limits reporting is limited to tabulated format in Table 8.6. GNF2 Pellet exposure extended to 63.5 GWD/ST per Ref. 3.18. Redundant information contained in the exposure dependent LHGR limit figures was removed. Fig. 8.4 3rd decimal changed to 0.58 from 0.581.
		Two new references added: ODYSY application to Licensing Stability calculation LTR (Ref. 3.10), and SER for Amendment 33 to GESTAR (Ref. 3.18) approving PRIME application for GNF2 thermal-mechanical limits. Update to cycle specific references.
		Page re-numbering, re-formatting, tables and figures order and re-numbering triggered editorial changes all throughout this document.
		§7.4.1.2 note removed as no longer applicable because ARTS-MEOD fully implemented.
		This revision record and summary added.

TABLE OF CONTENTS

1.0 PURPOSE

This report provides the cycle-specific operating limits for Cycle 20 of the James A. FitzPatrick Nuclear Power Plant. The following limits are addressed:

- Operating Limit Minimum Critical Power Ratio (MCPR)
- Flow Dependent MCPR Limits
- Average Planar Linear Heat Generation Rate (APLHGR)
- Linear Heat Generation Rate (LHGR)
- Flow-Biased Average Power Range Monitor (APRM) and Rod Block Monitor (RBM) Allowable Values
- Stability Option ID Exclusion Region

2.0 APPLICABILITY

The plant shall be operated within the limits specified in this report. If any of these limits are exceeded, the corrective actions specified in the Technical Specifications shall be taken.

3.0 REFERENCES

3.1 EN-LI-113, Licensing Basis Document Change process

3.2 JAFNPP Technical Specifications.

3.3 EC18500, Cycle 20 Core Reload

3.4 EN-DC-503, 3D Monicore New Cycle Update and Databank Maintenance.

3.5 Plant Operation Up To 100% Power With One Steam Line Isolated, JAF-SE-96-035.

3.6 GE Report, J.A. FitzPatrick Nuclear Power Plant APRM/RBM/Technical Specifications / Maximum Extended Operating Domain (ARTS/MEOD), NEDC-33087P, Revision 1, September 2005

3.7 General Electric Standard Application for Reload Fuel, NEDE-24011-P-A-16

3.8 GNF Report, Supplemental Reload Licensing Report for James A. FitzPatrick Reload 19 Cycle 20, 0000-0108-3718-SRLR, Revision 0, Class I, June, 2010. [EC23541, ECH-NE-10-00060 R0]

3.9 "GNF2 Fuel Design Cycle-Independent Analyses For Entergy FitzPatrick", GE Report, , GEH- - 0000-0074-2662-R1, June 2010. [EC23634, JAF-RPT-08-00013 R1]

3.10 Licensing Topical Report, ODYSY Application for Stability Licensing Calculations Including Option I-D and II Long Term Solutions, NEDE-33213P-A, April 2009

3.11 GE Letter, R. Kingston to P. Lemberg, Scram Time Versus Notch Positions for Option B, REK-E: 02-009, May 28, 2002

3.12 GE Report, James A. FitzPatrick Nuclear Power Plant Final Feedwater Temperature Reduction NEDC-33077, September 2002.

3.13 JD-02-122, Final Feedwater Temperature Reduction Implementation.

3.14 GE Report, GE14 Fuel Design Cycle-Independent Analyses for J. A. Fitzpatrick Nuclear Power Plant, GE-NE-0000-0002-1752-01P, Rev. 0, DRF 0000-0002-1752, September 2002.

3.15 GNF Report, Fuel Bundle Information Report for James A. FitzPatrick Reload 19 Cycle 20, 0000-0108-3718-FBIR, Revision 0, June 2010. [EC23547, ECH-NE-10-00061 R0]

3.16 JF-03-00402, ARTS/MEOD Phase 1 Implementation

3.17 JAF-RPT-MISC-04489, Rev.7, Power-Flow Map Report

3.18 "Final Safety Evaluation For Amendment 33 To Global Nuclear Fuel Topical Report NEDE-24011-P, "General Electric Standard Application For Reactor Fuel (GESTAR II)" (TAC NO. ME3525), Aug. 30, 2010

4.0 DEFINITIONS

4.1 Average Planar Linear Heat Generation Rate (APLHGR):

The APLHGR shall be applicable to a specific planar height and is equal to the sum of the heat generation rate per unit length of fuel rod for all the fuel rods in the specified assembly at the specified height divided by the number of fuel rods in the fuel assembly at the height.

4.2 Linear Heat Generation Rate (LHGR):

The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

4.3 Minimum critical power ratio (MCPR):

The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each type of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

4.4 Rated Recirculation Flow :

That drive flow which produces a core flow of 77.0 x 10^6 lb/hr.

5.0 RESPONSIBILITIES

NOTE: See EN-LI-113 (Reference 3.1)

5.1 **Shift Manager:**

Assure that the reactor is operated within the limits described herein.

5.2 **Reactor Engineering Supervisor:**

Assure that the limits described herein are properly installed in the 3D-Monicore databank used for thermal limit surveillance (Reference 3.4)

6.0 SPECIAL INSTRUCTIONS/REQUIREMENTS

Not Applicable

7.0 PROCEDURE

7.1 Operating Limit MCPR

During operation, with thermal power $\geq$ 25% of rated thermal power (RTP), the Operating Limit MCPR shall be equal to or greater than the limits given below.

7.1.1 Technical Specification LCO 3.2.2, Minimum Critical Power Ratio (MCPR)

7.1.2 The Operating Limit MCPR shall be determined based on the following requirement:

7.1.2.1 The average scram time to notch position 36 shall be:

$$\tau_{AVE} \leq \tau_B$$

7.1.2.2 The average scram time to notch position 36 is determined as follows:

$$\tau_{AVE} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i}$$

WHERE:

n = Number of surveillance tests performed to date in the cycle,

N_i = Number of active rods measured in the surveillance i

τ_i = Average scram time to notch position 36 of all rods measured in surveillance test i.

7.1.2.3 The adjusted analysis mean scram time is calculated as follows:

$$\tau_B(\sec) = \mu + 1.65\sigma \left[\frac{N_1}{\sum_{i=1}^{n} N_i} \right]^{1/2}$$

WHERE:

μ = Mean of the distribution for the average scram insertion time to the dropout of notch position 36 = 0.830 sec.

σ = Standard deviation of the distribution for average scram insertion time to the dropout of notch position 36 = 0.019 sec.

N_1 = The total number of active rods measured in Technical Specification SR 3.1.4.4.

The number of rods to be scram tested and the test intervals are given in Technical Specification LCO 3.1.4, Control Rod Scram Times

7.1.3 When requirement of 7.1.2.1 is met, the Operating Limit MCPR shall not be less than that specified in Table8.1, Table8.2, Table8.2, or Table8.4 as applicable for $\tau = 0$.

7.1.4 **WHEN** the requirement 7.1.2.1 is not met (i.e. $\tau_{AVE} > \tau_B$), **THEN** the Operating Limit MCPR values (as a function of τ) are given in Tables 8.1, 8.2, 8.2, or 8.4 as applicable.

$$\tau = \frac{(\tau_{AVE} - \tau_B)}{(\tau_A - \tau_B)}$$

WHERE:

τ_{AVE} = The average scram time to notch position 36 as defined in 7.1.2.2.

τ_B = The adjusted analysis mean scram time as defined in 7.1.2.3.

τ_A = the scram time to notch position 36 as defined in Technical Specification Table 3.1.4-1.

7.1.5 During single-loop operation, the Operating Limit MCPR shall be increased by 0.03.

7.1.6 The Operating Limit MCPR is the greater of the flow and power dependent MCPR operating limits, MCPR(F) and MCPR(P).

Operating Limit MCPR = MAX (MCPR(P), MCPR(F))

The flow dependent MCPR operating limit, MCPR(F), is provided in Figure 8.1.

For core thermal powers equal to or greater than 25%, MCPR (P) is the product of the rated Operating Limit MCPR presented in Tables 8.1, 8.2, 8.2, or 8.4 and the K (P) factor presented in Figure 8.2.

7.2 **Average Planar Linear Heat Generation Rate (APLHGR)**

7.2.1 Technical Specification LCO 3.2.1, Average Planar Linear Heat Generation Rate (APLHGR)

7.2.2 During operation, with thermal power $\geq$ 25% rated thermal power (RTP), the APLHGR shall be within the limits given in Table 8. 5 for the appropriate fuel type.

7.2.3 During single loop operation, the APLHGR for each fuel type shall not exceed the values given in 7.2.2 above multiplied by the appropriate value (0.78 for GE14 fuel and 0.85 for GNF2 fuel, per Ref. 3.8).

7.3 **Linear Heat Generation Rate (LHGR)**

7.3.1 Technical Specification LCO 3.2.3, Linear Heat Generation Rate (LHGR)

7.3.2 During operation, with thermal power ≥ 25% rated thermal power (RTP), the applicable limiting LHGR values for each fuel rod as a function of axial location and exposure shall be the smaller of the power and flow dependent LHGR limits multiplied by the applicable power and flow adjustment or the LHGR limit multiplied by 0.78 (for GE14) or 0.85 (for GNF2) when in single loop operation.

LHGR limit = MIN (LHGR (P), LHGR (F)).

Power-dependent LHGR limit, LHGR (P), is the product of the LHGR power dependent LHGR limit adjustment factor, LHGRFAC (P), shown in Figure 8.4 and the $LHGR_{std}$ in Table 8.6.

$LHGR (P) = LHGRFAC(P) \times LHGR_{std}$

The flow-dependent LHGR limit, LHGR (F), is the product of the LHGR flow dependent LHGR limit adjustment factor, LHGRFAC (F), shown in Figure 8.3 and the $LHGR_{std}$ in Table 8.6.

$LHGR (F) = LHGRFAC(F) \times LHGR_{std}$

7.4 **APRM Allowable Values (Digital Flow Cards)**

7.4.1 **APRM Flow Referenced Flux Scram Allowable Value (Run Mode)**

7.4.1.1 Technical Specifications:
LCO 3.3.1.1, Reactor Protection System (RPS) Instrumentation

7.4.1.2 When operating in Mode 1, the APRM Neutron Flux-High (Flow Biased) Allowable Value shall be

for two loop operation:

S≤ (% RTP) = 0.38*W+61.0%	0< W ≤ 24.7%
S≤ (% RTP) = 1.15*W+42.0%	24.7< W ≤ 47.0%
S≤ (% RTP) = 0.63*W+73.7%	47.0< W ≤ 68.7%
S≤ (% RTP) = 117.00% (Clamp)	W > 68.7%

for single loop operation:

S≤ (% RTP) = 0.38*W+57.9%	0< W ≤ 32.7%
S≤ (% RTP) = 1.15*W+32.8%	32.7< W ≤ 50.1%
S≤ (% RTP) = 0.58*W+61.3%	50.1< W ≤ 95.9%
S≤ (% RTP) = 117.00% (Clamp)	W > 95.9%

WHERE:

S = Allowable value in percent of rated thermal power;

W = Recirculation flow in percent of rated;

7.4.2 APRM Neutron Flux-High (Flow Biased) Rod Block Allowable Value (Relocated to the Technical Requirements Manual)

7.5 **RBM Upscale Rod Block Allowable Value**

7.5.1 Technical Specification LCO 3.3.2.1, Control Rod Block Instrumentation

7.5.2 The RBM upscale rod block allowable value shall be:

$S \leq 0.66W + K$ for two loop operation;

$S \leq 0.66W + K - 0.66\,\Delta W$ for single loop operation;

WHERE:

S	=	rod block allowable value in percent of initial;
W	=	Loop flow in percent of rated
K	=	Any intercept value may be used because the RBM intercept value <u>does not</u> effect the MCPR Operating Limit and the RBM is not assumed to function to protect the Safety Limit MCPR.
ΔW	=	Difference between two loop and single loop effective drive flow at the same core flow.

NOTE: If K can be any value, then $K - 0.66\Delta W$ can also be any value, and the allowable value adjustment for single loop operation is not necessary.

7.6 **Stability Option 1-D Exclusion Region and Buffer Zone.**

7.6.1 Technical Specification LCO 3.4.1, Recirculation Loops Operating

7.6.2 The reactor shall not be intentionally operated within the Exclusion Region given in <u>Figure 8.5</u> when the SOLOMON Code is operable.

7.6.3 The reactor shall not be intentionally operated within the Buffer Zone given in <u>Figure 8.5</u> when the SOLOMON Code is inoperable.

8.0 TABLES AND FIGURES

8.1 Following pages present Tables 8.1 through 8.6, and Figures 8.1 through 8.6. Exact tables and figures names are listed in the Table of Content on page 3.

TABLE 8.1
MCPR Operating Limit For Incremental Cycle Core Average Exposure

		GNF2 (Reload 19)		GNF2 (Reload 18)		GE14	
τ		BOC to MOC	MOC to EOC	BOC to MOC	MOC to EOC	BOC to MOC	MOC to EOC
= 0		1.42	1.48	1.43	1.48	1.39	1.43
>0.0	≤0.1	1.43	1.49	1.43	1.49	1.40	1.45
>0.1	≤0.2	1.44	1.50	1.44	1.50	1.41	1.46
>0.2	≤0.3	1.45	1.51	1.45	1.51	1.42	1.48
>0.3	≤0.4	1.46	1.52	1.46	1.52	1.43	1.50
>0.4	≤0.5	1.47	1.53	1.47	1.53	1.45	1.52
>0.5	≤0.6	1.48	1.54	1.48	1.54	1.46	1.53
>0.6	≤0.7	1.49	1.55	1.49	1.55	1.47	1.55
>0.7	≤0.8	1.50	1.56	1.50	1.56	1.48	1.57
>0.8	≤0.9	1.51	1.57	1.51	1.57	1.49	1.58
>0.9	≤1	1.52	1.58	1.52	1.58	1.50	1.60

Technical Specification LCO 3.2.2, Minimum Critical Power Ratio (MCPR)

For single loop operation, these limits shall be increased as given in Section 7.1.5.

The MCPR limits in this Table are subject to Power and Flow dependent adjustment per Section 7.1.6

NOTE: 1. When entering a new Exposure Range, check the current value of τ to assure adjustment per Step 7.1.4

2. Applicable for any value of K, see Step 7.5.2

TABLE 8.2
MCPR Operating Limit for Incremental Cycle Core Average Exposure for Operation above 75% of Rated Thermal Power with Three Steam Lines in Service

		GNF2		GE14	
τ		BOC to MOC	MOC to EOC	BOC to MOC	MOC to EOC
= 0		1.44	1.50	1.41	1.45
>0.0	≤0.1	1.45	1.51	1.42	1.47
>0.1	≤0.2	1.46	1.52	1.43	1.48
>0.2	≤0.3	1.47	1.53	1.44	1.50
>0.3	≤0.4	1.48	1.54	1.45	1.52
>0.4	≤0.5	1.49	1.55	1.47	1.54
>0.5	≤0.6	1.5	1.56	1.48	1.55
>0.6	≤0.7	1.51	1.57	1.49	1.57
>0.7	≤0.8	1.52	1.58	1.50	1.59
>0.8	≤0.9	1.53	1.59	1.51	1.60
>0.9	≤1	1.54	1.60	1.52	1.62

Technical Specification LCO 3.2.2, Minimum Critical Power Ratio (MCPR)

For single loop operation, these limits shall be increased as given in Section 7.1.5.

The MCPR limits in this Table are subject to Power and Flow dependent adjustment per Section 7.1.6

NOTE: **1.** When entering a new Exposure Range, check the current value of τ to assure adjustment per Step 7.1.4

2. Applicable for any value of K, see Step 7.5.2

TABLE 8.3
MCPR Operating Limit for Operation with Turbine Bypass Valves Out of Service

τ		GNF2 BOC to EOC	GE14 BOC to EOC
= 0		1.51	1.47
>0.0	≤0.1	1.52	1.49
>0.1	≤0.2	1.53	1.50
>0.2	≤0.3	1.54	1.52
>0.3	≤0.4	1.55	1.54
>0.4	≤0.5	1.56	1.56
>0.5	≤0.6	1.57	1.57
>0.6	≤0.7	1.58	1.59
>0.7	≤0.8	1.59	1.61
>0.8	≤0.9	1.60	1.62
>0.9	≤1	1.61	1.64

Technical Specification LCO 3.2.2, Minimum Critical Power Ratio (MCPR)

Technical Specification LCO 3.7.6, Main Turbine Bypass System

For single loop operation, these limits shall be increased as given in Section 7.1.5.

The MCPR limits in this Table are subject to Power and Flow dependent adjustment per Section 7.1.6

NOTE: 1. When entering a new Exposure Range, check the current value of τ to assure adjustment per Step 7.1.4

2. Applicable for any value of K, see Step 7.5.2

TABLE 8.4
MCPR Operating Limit for Operation with Final Feedwater Temperature Reduction

τ		GNF2	GE14
		EOC	EOC
= 0		1.48	1.43
>0.0	≤0.1	1.49	1.45
>0.1	≤0.2	1.50	1.46
>0.2	≤0.3	1.51	1.48
>0.3	≤0.4	1.52	1.50
>0.4	≤0.5	1.53	1.52
>0.5	≤0.6	1.54	1.53
>0.6	≤0.7	1.55	1.55
>0.7	≤0.8	1.56	1.57
>0.8	≤0.9	1.57	1.58
>0.9	≤1	1.58	1.60

Technical Specification LCO 3.2.2, Minimum Critical Power Ratio (MCPR)

For single loop operation, these limits shall be increased as given in Section 7.1.5.

The MCPR limits in this Table are subject to Power and Flow dependent adjustment per Section 7.1.6

NOTE: 1. When entering a new Exposure Range, check the current value of τ to assure adjustment per Step 7.1.4

2. Applicable for any value of K, see Step 7.5.2

MCPR Operating Limits in this table apply when at reduced feedwater temperature near end-of-cycle, see JD-02-122 (Reference 3.13) for further information.

TABLE 8.5
Exposure Dependent APLHGR Limits

GE14 Fuel Types

Average Planar Exposure	APLHGR Limit
GWd/ST	kW/ft
0.00	12.82
14.51	12.82
19.13	12.82
57.61	8.00
63.50	5.00

GNF2 Fuel Types

Average Planar Exposure	APLHGR Limit
GWd/ST	kW/ft
0.00	13.78
13.24	13.78
17.52	13.78
60.78	7.50
63.50	6.69

Technical Specification LCO 3.2.1, Average Planar Linear Heat Generation Rate (APLHGR)

For single loop operation these APLHGR values shall be multiplied by 0.85 for GNF2 fuel or 0.78 for GE14 fuel.

Linearly interpolate for APLHGR at intermediate exposure.

TABLE 8.6
Maximum LHGR

Maximum LHGR – GE14

Peak Pellet Exposure, GWD/ST	UO_2 LHGR Limit, kW/ft
0.00	13.40
14.51	13.40
57.61	8.00
63.50	5.00

Peak Pellet Exposure, GWd/ST	Most Limiting Gadolinia LHGR Limit, kW/ft
0.00	12.26
12.28	12.26
55.00	7.32
60.84	4.57

Maximum LHGR – GNF2

Peak Pellet Exposure, GWD/ST	UO_2 LHGR Limit, kW/ft
0.00	14.40
13.24	14.40
60.78	6.88
63.50	5.50

Peak Pellet Exposure, GWd/ST	Most Limiting Gadolinia LHGR Limit, kW/ft
0.00	12.45
10.47	12.45
55.65	6.48
63.5	4.36

Technical Specification LCO 3.2.3, Linear Heat Generation Rate (LHGR)

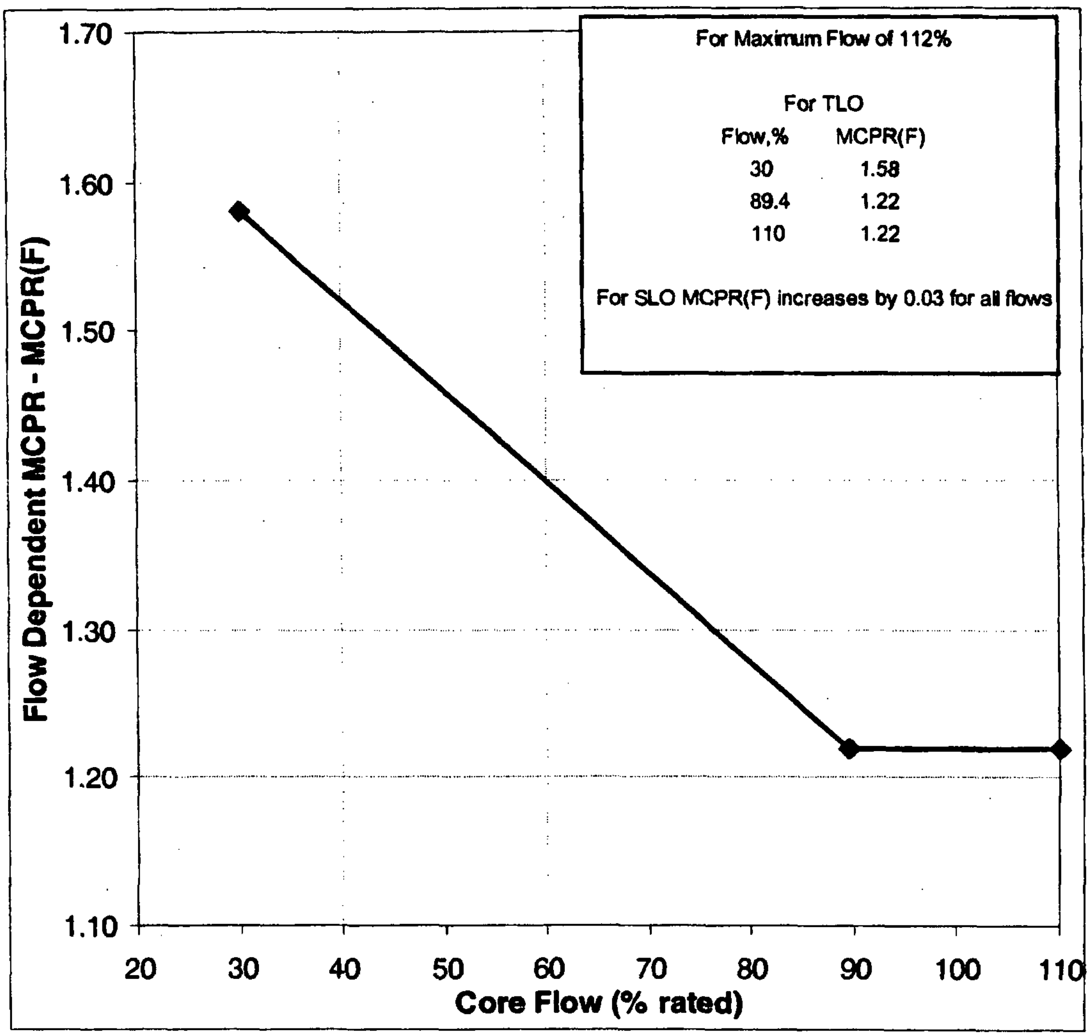
Design features of the fuel assemblies in the Cycle 20 core are provided in References 3.3, 3.15.

$LHGR_{std}$ values in the above Table 8.6 are subject to Power and Flow dependent adjustments per Section 7.3

For single loop operation these LHGR values shall be multiplied by 0.85 (for GNF2 fuel) or 0.78 (for GE14)

Linearly interpolate for LHGR at intermediate exposure

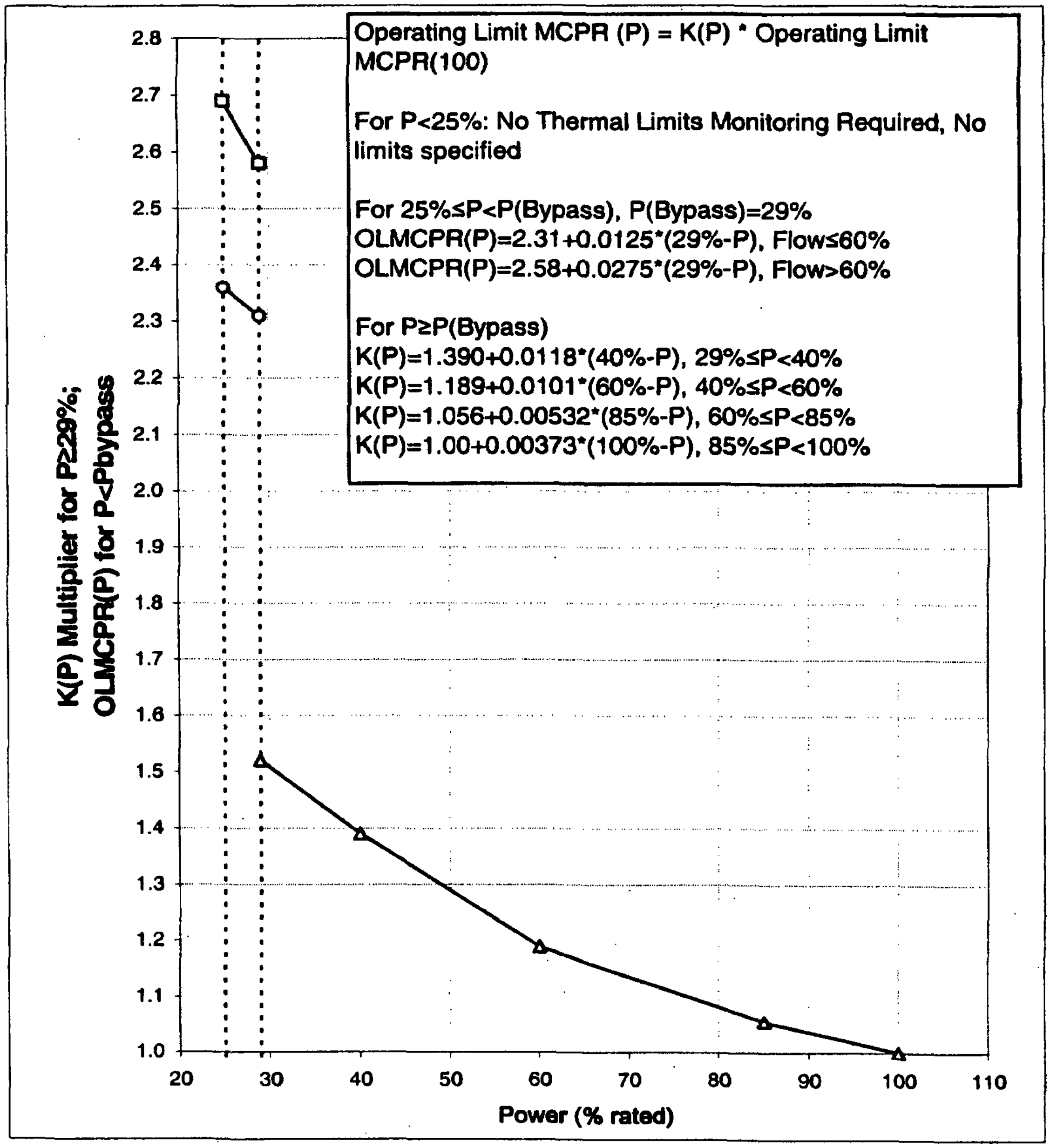
Figure 8.1
MCPR(F) Factor



Technical Specification LCO 3.2.2, Minimum Critical Power Ratio (MCPR)

Reference 3.8

Figure 8.2
K(P), OLMCPR(P) Factor

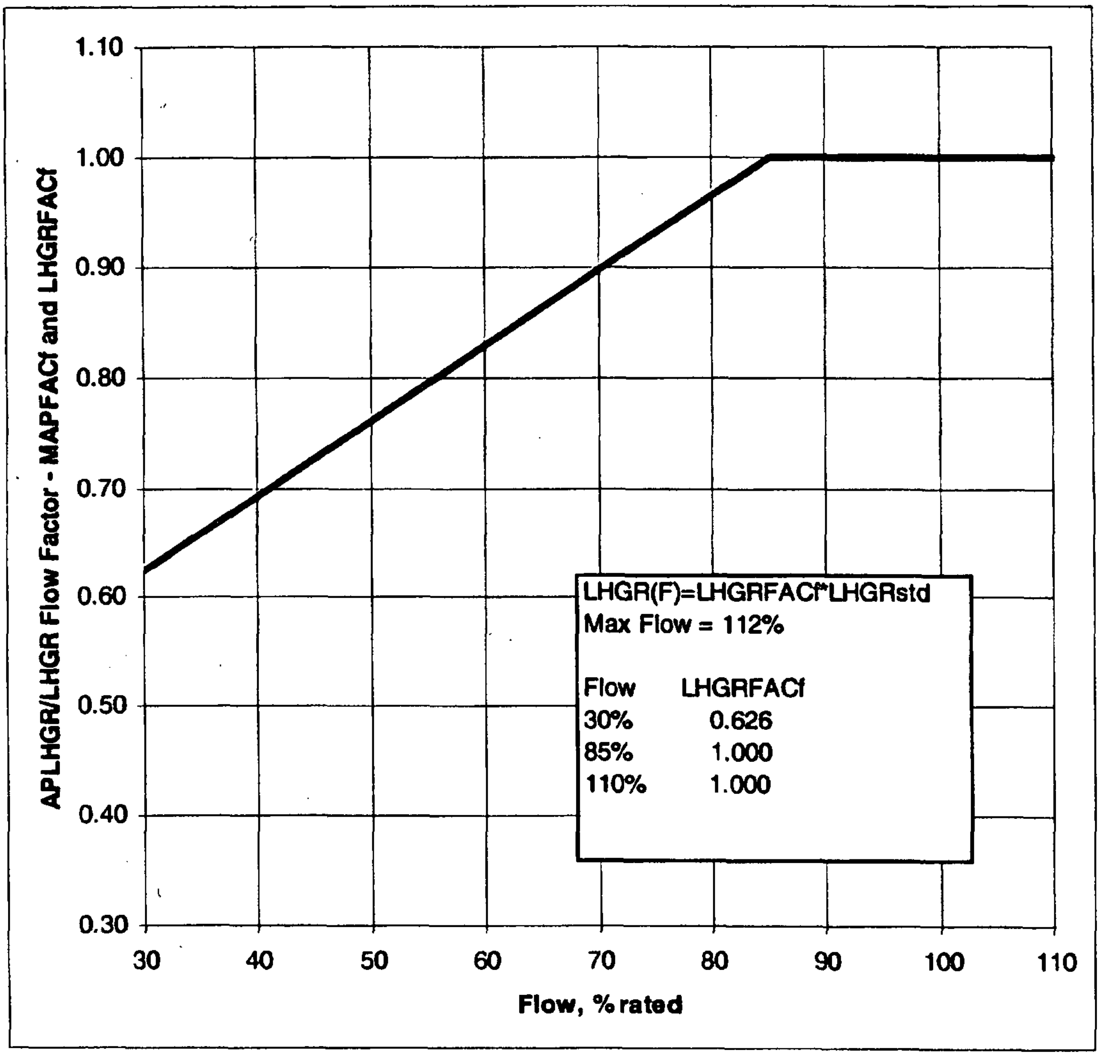


See Table 8.1, 8.2, 8.3,and Table 8.4 for Operating Limit MCPR(100)

Technical Specification LCO 3.2.2, Minimum Critical Power Ratio (MCPR)

Reference 3.8

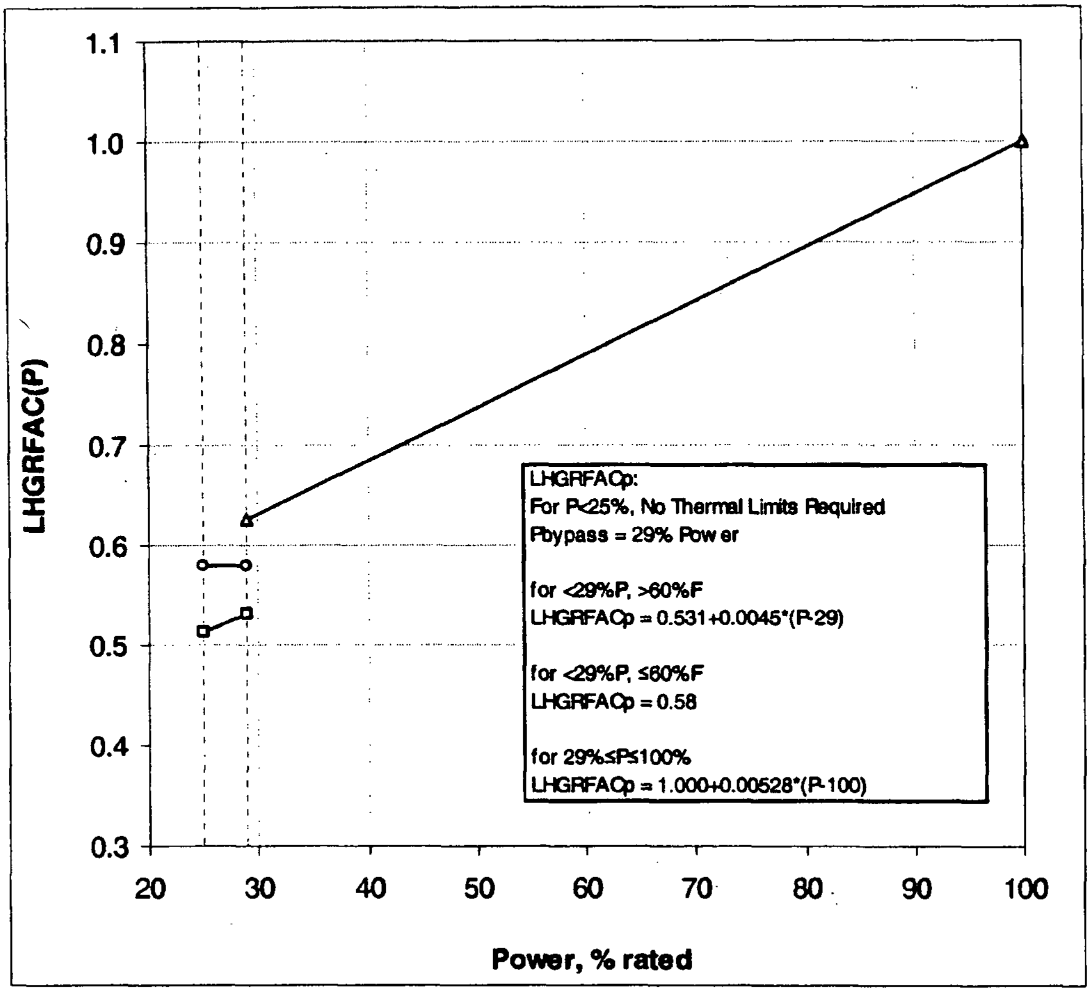
Figure 8.3
Flow-Dependent LHGR Multiplier, LHGRFAC(F)



See Table 8.6 for $LHGR_{STD}$ value

Reference 3.9

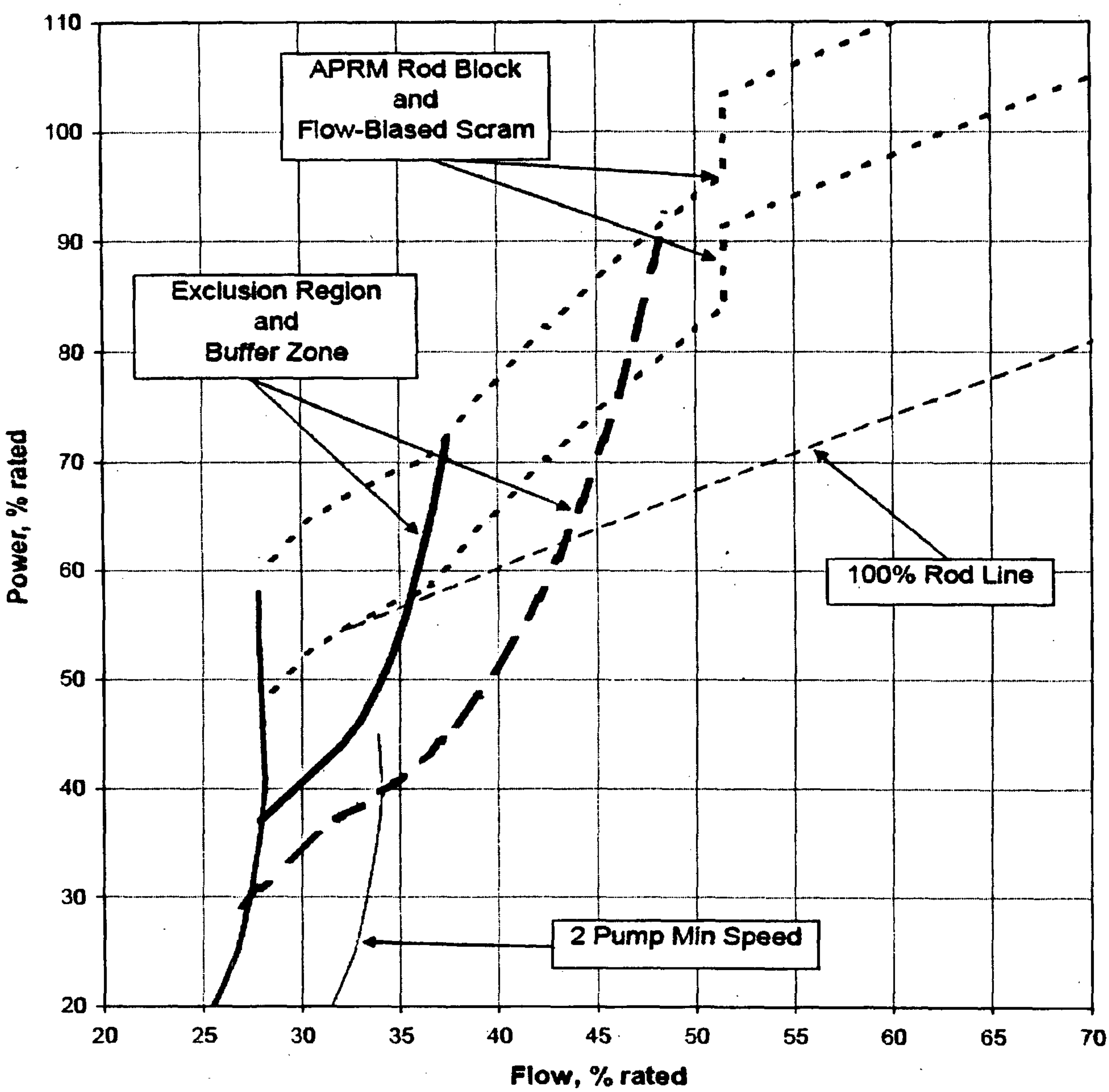
Figure 8.4
Power-Dependent LHGR Multiplier, LHGRFAC(P)



See Table 8.6 for $LHGR_{STD}$ values

Reference 3.9

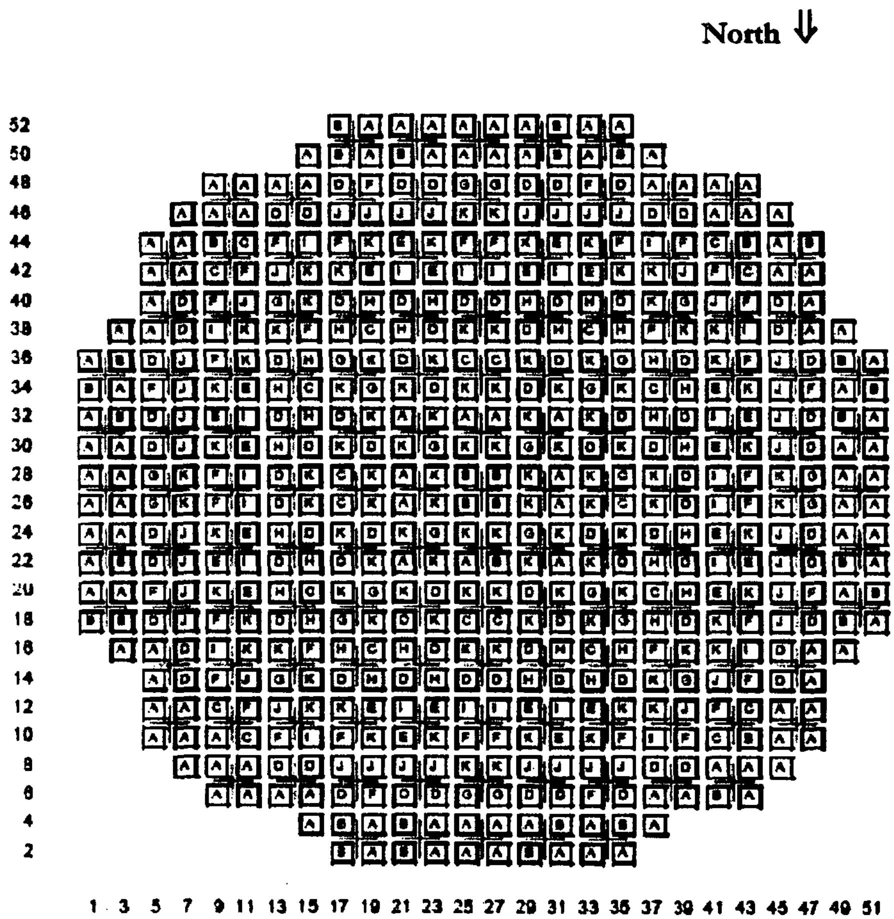
Figure 8.5
Stability Option 1-D Exclusion Region



See References 3.17 and 3.8 for details

The Cycle 20 Exclusion Region boundary is applicable for Cycle 20 exposure up to 15613 MWD/ST

Figure 8.6
Cycle 20 Loading Pattern by Bundle Design



Fuel Type	
A=GE14-P10DNAB402-10G6.0/4G5.0/1G2.0-100T-150-T6-2905 (Cycle 18)	G=GNF2-P10DG2B394-13GZ-100T2-150-T6-3077 (Cycle 19)
B=GE14-P10DNAB405-16GZ-100T-150-T6-2906 (Cycle 18)	H=GNF2-P10DG2B378-16GZ-100T2-150-T6-3299 (Cycle 20)
C=GNF2-P10DG2B377-13GZ-100T2-150-T6-3073 (Cycle 19)	I=GNF2-P10DG2B380-16GZ-100T2-150-T6-3298 (Cycle 20)
D=GNF2-P10DG2B379-14GZ-100T2-150-T6-3074 (Cycle 19)	J=GNF2-P10DG2B404-12GZ-100T2-150-T6-3297 (Cycle 20)
E=GNF2-P10DG2B396-15GZ-100T2-150-T6-3075 (Cycle 19)	K=GNF2-P10DG2B390-14GZ-100T2-150-T6-3300 (Cycle 20)
F=GNF2-P10DG2B407-6G6.0/6G5.0-100T2-150-T6-3076 (Cycle 19)	

9.0 USERS GUIDE

The COLR defines thermal limits for the various operating conditions expected during the cycle. At the start of the cycle the 3D-Monicore databank limits are set for;

- Cycle exposure range of BOC to MOC20
- $\tau = 0$
- Dual recirculation pump operation
- Four steam line operation, and
- Normal Feedwater Temperature

The following is a table that offers a check to assure the correct limits are applied when operating states or conditions change.

Change in Operating State	Change in Limits	Procedure Reference
Cycle Exposure = EOC20 – 2.946 GWD/ST OLMCPR changes to EOC values at cycle exposure of 11.495 GWD/ST. Databank will use 11.000 GWD/ST to account for uncertainties.	See Table 8.1 for $\tau \neq 0$ for change in MCPR.	EN-DC-503 transition to EOC limits will occur automatically
Scram Time Test Results such that $\tau \neq 0$ Option B limits for OLMCPR must be interpolated with Option A limits	Use new τ and see Table 8.1, 8.2, 8.3,and Table 8.4.	RAP-7.4.1
Single Loop Operation The SLMCPR increases by 0.03 and, therefore OLMCPR limits increase by 0.03. LHGR and MAPLHGR are reduced by a multiplier in SLO.	Increase MCPR Limits by 0.03, or change acceptance criterion in ST-40D (Step 8.2.19) to 0.97. Verify that 3D-Monicore has recognized the idle recirculation loop and is applying the SLO LHGR and MAPLHGR multipliers of 0.78 for GE14 and 0.85 for GNF2.	ST-40D
Three Steam Line Operation (3SL)	Increase OLMCPR according to Table 8.2.	None
Operation with Turbine Bypass Valves Out-of-Service OLMCPR values increase, no LHGR change required	Increase OLMCPR according to Table 8.3.	None
Operation under Final Feedwater Temperature Reduction	Apply OLMCPR according to Table 8.4.	None

TECHNICAL REQUIREMENTS MANUAL
BASES

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3.3 INSTRUMENTATION

B 3.3.A Residual Hear Removal (RHR) Shutdown Cooling

Bases

The RHR Shutdown Cooling Reactor Pressure - High reset function arms the logic circuit that initiates closure of the LPCI injection valves, 10MOV-25A and 10MOV-25B, on a reactor vessel low water level signal or a high drywell pressure signal when both shutdown cooling suction valves (10MOV-17 and 10MOV-18) are not fully closed.

The RHR Shutdown Cooling Reactor High Pressure reset function utilizes the instruments which provide Technical Specifications 3.3.6.1, Primary Containment Isolation Instrumentation, Function 6.a, Shutdown Cooling System Isolation Reactor Pressure - High. The signals are initiated from two reactor steam dome pressure transmitters that are connected to different condensing chambers through trip units. Each trip unit provides input to each trip system. However, only one channel input is required to be OPERABLE for a trip system to be considered OPERABLE. Both trip systems, with one channel per trip system are required to be OPERABLE.

The function required to be declared inoperable for Required Actions B.1 and C.1 is the PCIS isolation of the LPCI injection valves on a reactor vessel low level signal or a high drywell pressure signal when both shutdown cooling suction valves are not fully closed.

B 3.3 INSTRUMENTATION

B 3.3.L Emergency Diesel Generator (EDG) Instrumentation

Bases

Each EDG is provided with the following local and remote indications:

- Engine speed at the Engine Control Panel (ECP)
- Engine speed, generator voltage, and frequency at the Emergency Generator Panel (EGP) and at Control Room Panel 09-8
- Associated Emergency Bus voltage at the EGP

These instruments support operator control of the EDGs, but provide no automatic functions. These instruments are not credited in the Regulatory Guide (RG) 1.97 analysis in Updated Final Safety Analysis Report (UFSAR) Section 7.19. They are used to control the EDGs during the performance of required surveillances, including the assessment of TS voltage and frequency requirements.

Emergency Diesel Generator Speed Switches provide a signal to three Emergency Diesel Generator Tachometers and also provide a signal to the 40 RPM, 200 RPM and 400 RPM speed relays. The signal to the three tachometers located at the 09-8 panel, ECP, and EGP is indication only and is not a RG 1.97 credited indication. The speed indication is not considered a safety related or an augmented quality function. However, the speed relays are essential for the EDG start sequence.

Actual engine speed is not a critical parameter. The condition of the EDG can be monitored using the frequency meters locally and on Control Room Panel 09-8 regardless of the status of the EDG speed indication. If EDG frequency is nominally 60 Hz then the EDG speed must be at rated speed.

Bases (continued)

Engineering analyses indicate that the temperature rise in safeguards compartments without adequate ventilation flow or cooling is such that continued operation of the safeguards equipment or associated auxiliary equipment cannot be assured.

A summary of the Emergency Service Water flow rates is shown in Table 9.7-1 of the UFSAR. These design flow rates provide the bases for the required heat removal of 82°F ESW supply temperature. However, the required heat removal can also be met at other than design flow rates based on thermal performance testing and the number of coolers in service (i.e., crediting the heat removal capability of five unit coolers in a subsystem is acceptable). These conditions establish the maximum allowable lake temperature for the actual flow rate supplied to the individual crescent area unit coolers. Acceptance criteria of the thermal performance test require that at least four out of five coolers on each subsystem be available, no cooler is less than 50% effective, and the total heat removal capability is at least 621,600 BTU/hr at modified design conditions of 110°F Crescent Area temperature with 82°F lake water. For lake temperatures above 82°F, the same holds true, but with correspondingly higher heat loads. Therefore, OPERABILITY of equipment is verified based on heat transfer capability and not flow rates.

When less than five unit coolers in a Crescent Area are OPERABLE, the remaining four unit coolers require additional assessment to establish OPERABILITY. This is established in OP-4 Section G.20.

Screenwell intake lake temperature is monitored via EPIC or via local portable instrumentation and corrected for instrument inaccuracies. The adjusted measured lake temperature is compared to the maximum lake temperature for each unit cooler to be considered OPERABLE. The unit cooler maximum lake temperature is the maximum lake temperature at which the unit cooler can remove accident heat loads. The unit cooler maximum lake temperature is derived from the most recently conducted ST-8Q in which unit cooler thermal performance was determined. If the adjusted Screenwell intake temperature exceeds the maximum lake temperature the unit cooler is declared INOPERABLE. Four of five OPERABLE Crescent Area Unit coolers are required to maintain OPERABILITY of Crescent Area Ventilation. The OPERABILITY of the subject Crescent Area Ventilation is then limited to the maximum lake temperature of the lowest rated unit cooler within that Crescent Area.

With one Crescent Area Ventilation and Cooling subsystem inoperable, the Crescent Area Ventilation and Cooling subsystem must be restored to OPERABLE status within 7 days. This is consistent with Technical Specification LCO 3.7.2 Condition A, which results in the loss of cooling water to all coolers supplied by the ESW subsystem.

The 7 day Completion Time is based on the redundant capabilities afforded by the OPERABLE subsystem, the low probability of an accident occurring during this time period, and is consistent with Technical Specification LCO 3.7.2 Condition A, which results in the loss of cooling water to all coolers supplied by the ESW subsystem.

B 3.7 PLANT SYSTEMS

B 3.7.F Explosive Gas Monitoring

Bases

When TRO 3.7.F.b is not met, the following instrumentation shall be OPERABLE, as stated in Required Action A.2, and capable of providing automatic isolation of the offgas treatment system under the following conditions:

1. The offgas dilution steam flow instrumentation shall alarm and automatically isolate the offgas recombiner system at a low flow setpoint ≥ 4800 pounds per hour and at a high flow setpoint ≤ 7900 pounds per hour;

2. The offgas recombiner inlet temperature sensor shall alarm and automatically isolate the offgas recombiner system at a temperature setpoint of ≥ 125°C; and

3. The offgas recombiner outlet temperature sensor shall alarm and automatically isolate the offgas treatment system at a temperature setpoint of ≥ 150°C.

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in portions of the offgas treatment system not designed to withstand a hydrogen explosion (i.e., piping downstream of the recombiner) is maintained below the lower explosive limit of hydrogen. Operation of the offgas recombiner system ensures that the concentration of hydrogen in the offgas charcoal filters remain below combustion levels.

Thus it provides assurance that the release of radioactive materials will be controlled in conformance with the requirements of 10 CFR 50 Appendix A, General Design Criterion 60. The design steam dilution flow rate is that which results in a recombiner influent that is 4% hydrogen by volume. The low steam flow trip point is based on 70% of the design steam flow in order to ensure that the hydrogen concentration in the recombiner influent remains below the 7.3% flammability limit for hydrogen in a steam environment, and reroutes the offgas to prevent overheating or ignition of the recombiner catalyst. The hydrogen content for recombiner effluent must be maintained below the flammability limit of 4% for hydrogen in air. The high steam flow trip isolates the recombiner on excess steam flow that may be associated with a system malfunction.