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October 23, 2001

U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Technical Specification Bases Update to the NRC for Period Dated  
October 23, 2001  
Grand Gulf Nuclear Station  
Docket No. 50-416  
License No. NPF-29

GNRO-2001/00081

Ladies and Gentlemen:

Pursuant to Grand Gulf Nuclear Station (GGNS) Technical Specification 5.5.11, Entergy Operations, Inc. hereby submits an update of all changes made to GGNS Technical Specification Bases since the last submittal (GNRO-2001/00061 letter dated August 14, 2001 to the NRC from GGNS). This update is consistent with update frequency listed in 10CFR50.71(e).

This letter does not contain any commitments.

Should you have any questions, please contact Mike Larson at (601) 437-6685.

Yours truly,

A handwritten signature in cursive script, appearing to read "C. Bottemiller".

for  
Charles A. Bottemiller  
Manager, Plant Licensing

MJL/mjl  
attachment: GGNS Technical Specification Bases Revised Pages  
cc: (See Next Page)

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**ATTACHMENT TO GNRO-2001/00081**

**GGNS Gulf Technical Specification Bases Revised Pages  
Dated  
October 23, 2001**

<b>LDC#</b>	<b>BASES PAGES AFFECTED</b>	<b>TOPIC of CHANGE</b>
01083	B 3.6-87, 3.6-87a	Clarification added to clarify words and testing method for SR 3.6.4.1.4.
01050	B 2.0-5, 2.0-6, 2.0-7, 2.0-9, 2.0-10, 3.1-38, 3.1-38a, 3.1-45, 3.1-49, 3.2-9, 3.3-141, 3.3-142, 3.3-145, 3.3-147, 3.3-148, 3.3-161, 3.3-172, 3.4-38, 3.4-39, 3.4-40, 3.4-41, 3.6-15, 3.6-23a, 3.7-25, 3.7-27, 3.9-19, 3.9-21, 3.9-22, 3.9-24	Changes as a result of Alternate Source Term Technical Specification Changes

BASES (continued)

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SAFETY LIMIT  
VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 50.67 limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. (The required actions for a violation of the reactor water level SL include manually initiating ECCS to restore water level and depressurizing the reactor vessel, if necessary, for ECCS operation.) 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.

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(continued)

BASES (continued)

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
  2. ANF-524(P)(A), Revision 2, Supplements 1 and 2, November 1990.
  3. EMF-2209(P)(A), Revision 1, July 2000.
  4. 10 CFR 50.67, "Accident Source Term." |
  5. Letter: CEX0-2000-00293, J. B. Lee (EOI) to K.V. Walters (SPC), "Grand Gulf Nuclear Station Unit 1 and Riverbend Station Unit 1, Reload Transition Data - GE11 Additive Constants", July 25, 2000.
  6. NEDE-24011-P-A, GESTAR-II.
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B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

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BACKGROUND

The SL on reactor steam dome pressure protects the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and anticipated operational occurrences (AOOs).

During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." Following inception of unit operation, RCS components shall be pressure tested in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB, reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in 10 CFR 50.67 (Ref. 4). If this occurred in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere.

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APPLICABLE  
SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Vessel Steam Dome Pressure-High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

(continued)

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BASES

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SAFETY LIMIT  
VIOLATIONS  
(continued)

excess of 10 CFR 50.67 limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL 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.

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(continued)

BASES (continued)

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- REFERENCES
1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
  2. ASME, Boiler and Pressure Vessel Code, Section III.
  3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWA-5000.
  4. 10 CFR 50.67, "Accident Source Term." |
  5. ASME, Boiler and Pressure Vessel Code, 1971 Edition, Addenda, winter of 1972.
  6. ASME, Boiler and Pressure Vessel Code, 1974 Edition.
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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected.

The SLC System is also credited in the LOCA radiological analysis. The sodium pentaborate solution has been shown to sufficiently buffer the post-accident suppression pool that iodine re-evaluation can be precluded.

The SLC System satisfies the requirements of the NRC Policy Statement because operating experience and probabilistic risk assessment have generally shown it to be important to public health and safety.

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LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control, independent of normal reactivity control provisions provided by the control rods. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE, each containing an OPERABLE pump, an explosive valve and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

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APPLICABILITY

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE during these conditions, when only a single control rod can be withdrawn.

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(continued)

BASES

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ACTIONS

A.1 and A.2 (continued)

When the boron concentration is in the Limited Operation region (between 15.2 weight percent and 28.5 weight percent), the SBLC System contains sufficient boron to perform its design basis functions. But the associated solution temperatures required to prevent precipitation of the boron from solution is potentially greater than the primary containment's ambient temperature. As a result, the non safety tank heaters may be required to maintain the tank

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(continued)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

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**BACKGROUND** The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV consists of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two headers and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. The two instrument volumes are connected to a common drain line with two valves in series. Each header is connected to a common vent line with two valves in series. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

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**APPLICABLE SAFETY ANALYSES** The Design Basis Accident and transient analyses assume all the control rods are capable of scrambling. The primary function of the SDV is to limit the amount of reactor coolant discharged during a scram. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

- a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR 50.67 (Ref. 2); and
- b. Open on scram reset to maintain the SDV vent and drain path open so there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of 10 CFR 50.67 (Ref. 2) and adequate core cooling is maintained (Ref. 3). The SDV vent and drain valves also

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.1.8.3 (continued)

reset signal, the opening of the SDV vent and drain valves is verified. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1 and the scram time testing of control rods in LCO 3.1.3, "Control Rod OPERABILITY," overlap this Surveillance to provide complete testing of the assumed safety function. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

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REFERENCES

1. UFSAR, Section 4.6.1.1.2.4.2.6.
  2. 10 CFR 50.67, "Accident Source Terms." |
  3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981.
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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 Linear Heat Generation Rate (LHGR)

BASES

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**BACKGROUND** The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on the LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs). Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure or inability to cool the fuel does not occur during the anticipated operating conditions identified in UFSAR Chapters 6 and 15.

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**APPLICABLE SAFETY ANALYSES** The analytical methods and assumptions used in evaluating the fuel system design are presented in the UFSAR, Chapters 4, 6, and 15, and in References 1 and 2. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20 and 50. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

- a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the  $UO_2$  pellet; and
- b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

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

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the

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BASES

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APPLICABLE  
SAFETY ANALYSES,  
LCO, and  
APPLICABILITY  
(continued)

1. Main Steam Line Isolation

1.a. Reactor Vessel Water Level—Low Low Low, Level 1

Low reactor pressure vessel (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 the MSIVs and other interfaces with the reactor vessel occurs to prevent offsite dose limits from being exceeded. The Reactor Vessel Water Level—Low Low Low, Level 1 Function is one of the many Functions assumed to be OPERABLE and capable of providing isolation signals. The Reactor Vessel Water Level—Low Low Low, Level 1 Function associated with isolation is assumed in the analysis of the recirculation line break (Ref. 1). The isolation of the MSL on Level 1 supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level 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 of Reactor Vessel Water Level—Low Low Low, Level 1 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low Low Low, Level 1 Allowable Value is chosen to be the same as the ECCS Level 1 Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR 50.67 limits. |

This Function isolates the Group 1 and 5 valves.

1.b. Main Steam Line Pressure—Low

Low MSL pressure 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/hour 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/hour)

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(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES,  
LCO, and  
APPLICABILITY

1.b. Main Steam Line Pressure—Low (continued)

is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, 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.

1.c. Main Steam Line Flow—High

Main Steam Line Flow—High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow—High Function is directly assumed in the analysis of the main steam line break (MSLB) accident (Ref. 1). The isolation action, along with the scram function of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 50.67 limits.

The MSL flow signals are initiated from 16 transmitters that are connected to the four MSLs. The transmitters are arranged such that, even though physically separated from

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BASES

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APPLICABLE  
SAFETY ANALYSES,  
LCO, and  
APPLICABILITY  
(continued)

2. Primary Containment and Drywell Isolation

2.a, 2.e. Reactor Vessel Water Level—Low Low, Level 2

Low RPV water level indicates the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 2 supports actions to ensure that offsite dose limits of 10 CFR 50.67 are not exceeded. The Reactor Vessel Water Level—Low Low, Level 2 Function associated with isolation is implicitly assumed in the UFSAR analysis as these leakage paths are assumed to be isolated post LOCA. In addition, Function 2.a provides an isolation signal to certain drywell isolation valves. The isolation of drywell isolation valves, in combination with other accident mitigation systems, functions to ensure that steam and water releases to the drywell are channeled to the suppression pool to maintain the pressure suppression function of the drywell.

Reactor Vessel Water Level—Low Low, Level 2 signals are initiated from 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 of Reactor Vessel Water Level—Low Low, Level 2 Function are available and are required to be OPERABLE to ensure no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level—Low Low, Level 2 Allowable Value was chosen to be the same as the ECCS Reactor Vessel Water Level—Low Low, Level 2 Allowable Value (LCO 3.3.5.1), since isolation of these valves is not critical to orderly plant shutdown.

This Function isolates the Group 6A, 6B, and 7 valves.

2.b, 2.d, 2.f. Drywell Pressure—High

High drywell pressure can indicate a break in the RCPB. The isolation of some of the automatic isolation valves on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR 50.67 are not exceeded. The Drywell

(continued)

BASES

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APPLICABLE  
SAFETY ANALYSES,  
LCO, and  
APPLICABILITY

2.c. Reactor Vessel Water Level - Low Low Low, Level 1  
(continued)

to the drywell are channeled to the suppression pool to maintain the pressure suppression function of the drywell.

Reactor vessel water level signals are initiated from 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 of Reactor Vessel Water Level - Low Low Low, Level 1 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level - Low Low Low, Level 1 Allowable Value is chosen to be the same as the ECCS Reactor Vessel Water Level - Low Low Low, Level 1 Allowable Value (LCO 3.3.5.1) to ensure the valves are isolated to prevent offsite doses from exceeding 10 CFR 50.67 limits.

This Function isolates the Group 5 isolation valves.

2.g. Containment and Drywell Ventilation Exhaust Radiation - High

High ventilation exhaust radiation is an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the RCPB. When Exhaust Radiation - High is detected, valves whose penetrations communicate with the primary containment atmosphere are isolated to limit the release of fission products. In addition, this Function provides an isolation signal to certain drywell isolation valves. The isolation of drywell isolation valves, in combination with other accident mitigation systems, functions to ensure that steam and water releases to the drywell are channeled to the suppression pool to maintain the pressure suppression function of the drywell.

Additionally, the Ventilation Exhaust Radiation - High is assumed to initiate isolation of the primary containment during a fuel handling accident involving the handling of recently irradiated fuel (Ref. 2).

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BASES

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APPLICABLE  
SAFETY ANALYSES,  
LCO, and  
APPLICABILITY

2.g. Containment and Drywell Ventilation Exhaust  
Radiation-High (continued)

Four channels of Containment and Drywell Ventilation Exhaust-High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function. Two upscale-Hi Hi, one upscale-Hi Hi and one downscale, or two downscale signals from the same trip system actuate the trip system and initiate isolation of the associated containment and drywell isolation valves.

The Allowable Values are chosen to promptly detect gross failure of the fuel cladding and to ensure offsite doses remain below 10 CFR 20 and 10 CFR 50.67 limits.

The Function is required to be OPERABLE during operations with a potential for draining the reactor vessel (OPDRVs) and movement of recently irradiated fuel assemblies in the primary or secondary containment because the capability of detecting radiation releases due to fuel failures (due to fuel uncover or dropped fuel assemblies) must be provided to ensure offsite dose limits are not exceeded. Due to radioactive decay, this Function is only required to isolate primary containment during those fuel handling accidents involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

This Function isolates the Group 7 valves.

2.h. Manual Initiation

The Manual Initiation push button channels introduce signals into the primary containment and drywell isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific UFSAR safety analysis that takes credit for this Function. It is retained for the isolation function as required by the NRC in the plant licensing basis.

There are four push buttons for the logic, two manual initiation push buttons per trip system. There is no

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BASES

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APPLICABLE  
SAFETY ANALYSES,  
LCO, and  
APPLICABILITY  
(continued)

5.d. Drywell Pressure - High

High drywell pressure can indicate a break in the RCPB. The isolation of some of the PCIVs on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR 50.67 are not exceeded. The Drywell Pressure - High Function associated with isolation of the RHR Shutdown Cooling System is not modeled in any UFSAR accident or transient analysis because other leakage paths (e.g., MSIVs) are more limiting.

High drywell pressure signals are initiated from pressure transmitters that sense the pressure in the drywell. Four channels of Drywell Pressure - High 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 the same as the ECCS Drywell Pressure - High Allowable Value (LCO 3.3.5.1), since this may be indicative of a LOCA inside primary containment.

This Function isolates the Group 3 valves.

5.e. Manual Initiation

The Manual Initiation push button channels introduce signals into the RHR Shutdown Cooling System isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific UFSAR safety analysis that takes credit for this Function. It is retained for the isolation function as required by the NRC in the plant licensing basis.

There are four push buttons for the logic, two manual initiation push buttons per trip system. There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the push buttons.

Four channels of the Manual Initiation Function are available and are required to be OPERABLE.

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B 3.3 INSTRUMENTATION

B 3.3.6.2 Secondary Containment Isolation Instrumentation

BASES

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BACKGROUND

The secondary containment isolation instrumentation automatically initiates closure of appropriate secondary containment isolation valves (SCIVs) and starts the Standby Gas Treatment (SGT) System. The function of these systems, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Ref. 1), such that offsite radiation exposures are maintained within the requirements of 10 CFR 50.67 that are part of the NRC staff approved licensing basis. Secondary containment isolation and establishment of vacuum with the SGT System within the assumed time limits ensures that fission products that leak from primary containment following a DBA, or are released outside primary containment or during certain operations when primary containment is not required to be OPERABLE are maintained within applicable limits.

The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of secondary containment isolation. Most channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a secondary containment isolation signal to the isolation logic. Functional diversity is provided by monitoring a wide range of independent parameters. The input parameters to the isolation logic are (a) reactor vessel water level, (b) drywell pressure, (c) fuel handling area ventilation exhaust, and (d) fuel handling area pool sweep exhaust radiation. Redundant sensor input signals from each parameter are provided for initiation of isolation parameters. In addition, manual initiation of the logic is provided.

For all Secondary Containment Isolation instrumentation Functions, both channels in a trip system are required to trip the associated trip system. In addition to the isolation function, the SGT subsystems are initiated. There are two SGT subsystems with one subsystem being initiated by

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Specific Activity

BASES

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BACKGROUND

During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the coolant can plate out in the RCS, and, at times, an accumulation will break away to spike the normal level of radioactivity. The release of coolant during a Design Basis Accident (DBA) could send radioactive materials into the environment.

Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure, in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within the limits of 10 CFR 50.67 (Ref. 1).

This LCO contains iodine specific activity limits. The iodine isotopic activities per gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. The allowable levels are intended to limit the 2 hour radiation dose to an individual at the site boundary to a small fraction of the 10 CFR 50.67 limit.

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APPLICABLE  
SAFETY ANALYSES

Analytical methods and assumptions involving radioactive material in the primary coolant are presented in the UFSAR (Ref. 2). The specific activity in the reactor coolant (the source term) is an initial condition for evaluation of the consequences of an accident due to a main steam line break (MSLB) outside containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is assumed to end when the main steam isolation valves (MSIVs) close completely.

This MSLB release forms the basis for determining offsite doses (Ref. 2). The limits on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses at the site boundary, resulting from an MSLB outside containment during steady state operation, will not exceed 10% of the dose guidelines of 10 CFR 50.67.

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BASES

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APPLICABLE  
SAFETY ANALYSES  
(continued)

The limit on specific activity is from a parametric evaluation of typical site locations. This limit is conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site.

RCS specific activity satisfies Criterion 2 of the NRC Policy Statement.

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LCO

The specific iodine activity is limited to  $\leq 0.2 \mu\text{Ci/gm}$  DOSE EQUIVALENT I-131. This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR 50.67 limits.

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APPLICABILITY

In MODE 1, and MODES 2 and 3 with any main steam line not isolated, a limit on the primary coolant radioactivity is applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment.

In MODES 2 and 3 with the Main Steam Lines Isolated, such a limit does not apply since an escape path does not exist. In MODES 4 and 5, no limit is required since the reactor is not pressurized and the potential for leakage is reduced.

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ACTIONS

A.1 and A.2

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is  $\leq 4.0 \mu\text{Ci/gm}$ , samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems.

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BASES

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ACTIONS

A.1 and A.2 (continued)

A Note to the Required Actions of Condition A excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of a limiting event while exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to, power operation.

B.1, B.2.1, B.2.2.1, and B.2.2.2

If the DOSE EQUIVALENT I-131 cannot be restored to  $\leq 0.2$   $\mu\text{Ci/gm}$  within 48 hours, or if at any time it is  $> 4.0$   $\mu\text{Ci/gm}$ , it must be determined at least every 4 hours and all the main steam lines must be isolated within 12 hours. Isolating the main steam lines precludes the possibility of releasing radioactive material to the environment in an amount that is more than a small fraction of the requirements of 10 CFR 50.67 during a postulated MSLB accident. |

Alternately, the plant can be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for bringing the plant to MODES 3 and 4 are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

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(continued)

BASES (continued)

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.8.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level.

This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.

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REFERENCES

1. 10 CFR 50.67, "Accident Source Term."
  2. UFSAR, Section 15.6.4.
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BASES (continued)

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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 for which the consequences are mitigated by PCIVs are a loss of coolant accident (LOCA), a main steam line break (MSLB), and a fuel handling accident involving the handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours) inside primary containment (Refs. 1 and 2). In the analysis for each of these accidents, it is assumed that PCIVs are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through PCIVs are minimized. Of the events analyzed in Reference 1, the LOCA is the most limiting event due to radiological consequences. An analysis of the effect of the purge valves being open at the initiation of a LOCA has been performed. This condition was found to result in negligible dose contributions due to the small amounts of activity in the reactor coolant. It is assumed that the primary containment is isolated such that release of fission products to the environment is controlled.

PCIVs satisfy Criterion 3 of the NRC Policy Statement.

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LCO

PCIVs form a part of the primary containment boundary and some also form a part of the RCPB. The PCIV safety function is related to minimizing the loss of reactor coolant inventory, and establishing the primary containment boundary during a DBA.

The power operated isolation valves are required to have isolation times within limits. Additionally, power operated automatic valves are required to actuate on an automatic isolation signal.

(continued)

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.1.3.4 (continued)

by the applicable regulatory document (10 CFR 50.67).  
Closure times explicitly assumed in accident analyses are  
listed in UFSAR Table 6.2-44 Note d. The Frequency of this  
SR is in accordance with the Inservice Testing Program.

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(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS  
(continued)

SR 3.6.4.1.3 and SR 3.6.4.1.4

The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. To ensure that all fission products are treated, SR 3.6.4.1.3 verifies that the SGT System will rapidly establish and maintain a pressure in the secondary containment that is less than the lowest postulated pressure external to the secondary containment boundary.

SR 3.6.4.1.4 demonstrates that each OPERABLE SGT subsystem can maintain a reduced pressure in the secondary containment sufficient to allow the secondary containment to be in thermal equilibrium at steady state conditions. The test criterion specified by SR 3.6.4.1.4 includes an allowance for building degradation between performances of the surveillance. This allowance represents additional building inleakage of 125 scfm.

As discussed in B 3.6.4.2, the SGT System has the capacity to maintain secondary containment negative pressure assuming the failure of all nonqualified lines 2 inches and smaller. The number and size of these assumed failures can vary as penetrations are added or removed from the secondary containment boundary. To account for the absence of these assumed failures under test conditions the test criterion specified by SR 3.6.4.1.4 is modified. Failure of nonqualified lines 2 inches and smaller could increase secondary containment in-leakage by approximately 200 scfm. To account for this additional in-leakage, and in addition to the requirements of SR 3.6.4.1.4, each SGT subsystem must maintain  $\geq 0.303$  inches of vacuum water gauge in the secondary containment for 1 hour at a flow rate  $\leq 4000$  cfm. This value represents the minimum required differential pressure at  $\leq 4000$  scfm system flow needed to ensure that the integrity of the SGT System boundary will meet its design requirement of  $\geq 0.25$  inches of vacuum water gauge in response to postulated accidents.

Therefore, these two sets are used to ensure secondary containment boundary integrity. Since these SRs are secondary containment tests, they need not be performed with

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(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.6.4.1.3 and SR 3.6.4.1.4 (continued)

each SGT subsystem. Testing is performed on a STAGGERED TEST BASIS to ensure that both SGT subsystems are alternatively tested. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

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REFERENCES

1. UFSAR, Section 15.6.5.
  2. UFSAR, Section 15.7.4.
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B 3.7 PLANT SYSTEMS

B 3.7.6 Fuel Pool Water Level

BASES

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BACKGROUND            The minimum water level in the spent fuel storage pool and upper containment fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool and upper containment fuel storage pool design is found in the UFSAR, Section 9.1.2 (Ref. 1). The assumptions of the fuel handling accident are found in the UFSAR, Section 15.7.4 (Ref. 2).

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APPLICABLE  
SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident. A fuel handling accident is evaluated to ensure that the radiological consequences remain within the guidelines in NUREG-0800, Section 15.0.1 (Ref. 4) and 10 CFR 50.67, (Ref. 5). A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in Appendix B to Regulatory Guide 1.183 (Ref. 6).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto stored fuel bundles. The consequences of a fuel handling accident inside the auxiliary building and inside containment are documented in Reference 2. The water levels in the spent fuel storage pool and upper containment fuel storage pool provide for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The fuel pool water level satisfies Criterion 2 of the NRC Policy Statement.

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(continued)

BASES (continued)

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- REFERENCES
1. UFSAR, Section 9.1.2.
  2. UFSAR, Section 15.7.4.
  3. Deleted
  4. NUREG-0800, Section 15.0.1, Revision 0, July 2000. |
  5. 10 CFR 50.67, "Accident Source Term." |
  6. Regulatory Guide 1.183, July 2000. |
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B 3.9 REFUELING OPERATIONS

B 3.9.6 Reactor Pressure Vessel (RPV) Water Level - Irradiated Fuel

BASES

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BACKGROUND

The movement of irradiated fuel assemblies within the RPV requires a minimum water level of 22 ft 8 inches above the top of the RPV flange. During refueling, this maintains a sufficient water level in the upper containment pool. Sufficient water is necessary to retain fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine and particulate activity would be retained to limit offsite doses from the accident to within the guidelines in References 3 and 5.

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APPLICABLE  
SAFETY ANALYSES

During movement of irradiated fuel assemblies, the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Appendix B to Regulatory Guide 1.183 (Ref. 1). A minimum water level of 23 ft allows a decontamination factor of 200 (Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. In addition, all of the particulate release is retained in the water.

Analysis of the fuel handling accident is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 24 hours prior to fuel handling, the analysis demonstrates that the source term release due to a postulated fuel handling accident is adequately captured by the water, and that offsite doses are maintained within allowable limits (Refs. 3 and 5).

While the worst case assumptions include the dropping of the irradiated fuel assembly being handled onto the reactor core, the possibility exists of the dropped assembly striking the RPV flange and releasing fission products. Therefore, the minimum depth for water coverage to ensure acceptable radiological consequences is specified from the RPV flange. Since the worst case event results in failed fuel assemblies seated in the core, as well as the dropped

(continued)

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.6.1 (continued)

level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

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REFERENCES

1. Appendix B to Regulatory Guide 1.183, July 2000. |
  2. UFSAR, Section 15.7.4.
  3. NUREG-0800, Section 15.0.1. |
  4. NUREG-0831, Supplement 6, Section 16.4.2.
  5. 10 CFR 50.67, "Accident Source Term." |
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B 3.9 REFUELING OPERATIONS

B 3.9.7 Reactor Pressure Vessel (RPV) Water Level - New Fuel or Control Rods

BASES

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BACKGROUND

The movement of new fuel assemblies or handling of control rods within the RPV when fuel assemblies seated within the reactor vessel are irradiated requires a minimum water level of 23 ft above the top of irradiated fuel assemblies seated within the RPV. During refueling, this maintains a sufficient water level above the irradiated fuel. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine and particulate activity would be retained to limit offsite doses from the accident to within the guidelines in References 3 and 5.

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APPLICABLE SAFETY ANALYSES

During movement of new fuel assemblies or handling of control rods over irradiated fuel assemblies, the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Appendix B to Regulatory Guide 1.183 (Ref. 1). A minimum water level of 23 ft allows a decontamination factor of 200 (Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. In addition, all of the particulate release is retained in the water.

Analysis of the fuel handling accident inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 24 hours prior to fuel handling, the analysis demonstrates that the source term release due to a postulated fuel handling accident is adequately captured by the water, and that offsite doses are maintained within allowable limits (Refs. 3 and 5).

The related assumptions include the worst case dropping of an irradiated fuel assembly onto the reactor core loaded with irradiated fuel assemblies.

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BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.9.7.1 (continued)

met. Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

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REFERENCES

1. Appendix B to Regulatory Guide 1.183, July 2000.
  2. UFSAR, Section 15.7.4.
  3. NUREG-0800, Section 15.0.1.
  4. NUREG-0831, Supplement 6, Section 16.4.2.
  5. 10 CFR 50.67, "Accident Source Term."
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