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GNRO-2018/00039

August 20, 2018

10CFR50.71

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Report of Technical Specification Bases Changes
Grand Gulf Nuclear Station, Unit 1
Docket No. 50-416
License No. NPF-29

Dear Sir or Madam:

Pursuant to Grand Gulf Nuclear Station (GGNS) Technical Specification 5.1.11 and 10 CFR 50.71 (e), Entergy Operations Inc. herby submits an update of all changes made to the GGNS Technical Specification Bases since the last submittal (GNRO-2016/00024, dated May 17, 2016).

This letter contains no new commitments. If you have any questions or require additional information, please contact Doug Neve at 601-437-2103.

Sincerely,

E. A. La

Eric A. Larson
Site Vice President
Grand Gulf Nuclear Station
EAL/tdf

Attachments: Attachment 1: Technical Specification Bases Change Summary
Attachment 2: Technical Specification Bases Pages

cc: (See Next Page)

cc: NRC Senior Resident Inspector
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Attachment 1 to GNRO 2018-00039

Technical Specification Bases Change Summary

Grand Gulf Nuclear Station
Technical Specification Bases Changes

LBDCR Number	Topic	Affected Bases Pages
2016064	Edit Basis section 8.3.3.8.2 clarify equipment powered from RPS Buses	B3.3-232
2016070	Update Bases 3.3.1.1 2.f for Oscillation Power Range Monitor (OPRM) Upscale to reflect changes made with Maximum Extended Load Line Limit analysis Plus (MELLLA+) in Amendment 205	B3.3-9e B3.3-9f
2016271	Revise Bases SR 3.6.1.3.8 for to include MELLLA+ Pa of 12.1 psig	B3.6-25
2016272	Correct typographical and incorporation errors in Bases 3.3.1.1 introduced during Amendment 188 (Average Power Range Neutron Monitoring digital upgrade)	B3.3-9 B3.3-21 B3.3-23c
2016274	Correct typographical errors and incorrect incorporations introduced during 2010 adoption of 2004 Edition of ASME OM Code.	B3.4-20a
2017004	Deletion of SR 3.3.1.1.23 Bases, which was deleted in Amendment 205	B3.3-29d
2017036	Insert leakage limits for Standby Liquid Control system and identify orifice as ASME Code Class boundary in Bases 3.1.7.	B3.1-38
2017051	Revise MELLA+ operability requirements for OPRM Upscale function in Bases 3.3.1.1.	B3.3-6a
2017071	Update Bases 3.6.1.1 to show that Main Steam Isolation Valve leakage is included in La (0.682%/24 hours).	B3.6-2
2017092	Remove invalid turbine bypass valve precautions regarding EOC-RPT (end of cycle recirculation pump trip) bypass function from SR 3.3.4.1.5 Bases.	B3.3-75
2018011	Revise Automatic Depressurization System discussion in Bases 3.3.5.1 to correctly reflect system logic	B3.3-91
2018020	Revise Bases SR 3.6.4.3.1 to reflect 15 minutes versus 15 hours for Standby Gas Treatment System operation in accordance with Amendment #208	B3.6-100

Attachment 2 to GNRO 2018-00039

Technical Specification Bases Pages

BASES

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. The leakage limit for the SLC system is 2.87 gpm. This includes the 0.13 gpm leakage for the restricting orifice on the class boundary at the SLC test tank. (Ref. 4)
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.
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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Fixed Neutron Flux-High (continued)

The Average Power Range Monitor Fixed Neutron Flux-High Function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the SLs (e.g., MCPR and RCS pressure) being exceeded. Although the Average Power Range Monitor Fixed Neutron Flux-High Function is assumed in the CRDA analysis that is applicable in MODE 2, the Average Power Range Monitor Neutron Flux-High, Setdown Function conservatively bounds the assumed trip and, together with the assumed IRM trips, provides adequate protection. Therefore, the Average Power Monitor Fixed Neutron Flux—High Function is not required in MODE 2.

2.c. Average Power Range Monitor-Inop

Three of the four APRM/OPRM channels are required to be OPERABLE for each of the APRM Functions. This function (Inop) provides assurance that the minimum number of channels is OPERABLE.

For any APRM/OPRM channel, any time its keylock switch is in any position other than "OPER," a module is unplugged, or the automatic self-test system detects a critical fault with the APRM/OPRM channel, an Inop trip is sent to all four 2-out-of-4 voter channels. Inop trips from two or more un-bypassed APRM/OPRM channels result in a trip output from all four 2-out-of-4 voter channels to their associated trip system. This Function was not specifically credited in the accident analysis, but it is retained for the RPS as required by the NRC approved licensing basis.

There is no Allowable Value for this Function.

This Function is required to be OPERABLE in the MODES where the APRM Functions are required.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The 2-Out-Of-4 Voter Function votes APRM Functions 2.a, 2.b, and 2.d independently of Function 2.f. The voter also includes separate outputs to RPS for the two independently voted sets of functions, each of which is redundant (four total outputs). Function 2.e must be declared inoperable if any of its functionality is inoperable. However, due to the independent voting of APRM trips, and the redundancy of outputs, there may be conditions where the voter function 2.e is inoperable, but trip capability for one or more of the other APRM Functions through that voter is still maintained. This may be considered when determining the condition of other APRM Functions resulting from partial inoperability of the Voter Function 2.e.

There is no Allowable Value for this Function.

2.f. Oscillation Power Range Monitor (OPRM) Upscale

The OPRM Upscale trip function complies with GDC 10 and GDC 12, thereby providing protection from exceeding the fuel MCPR SL due to anticipated thermal-hydraulic power oscillations. This is accomplished by implementing the Detect and Suppress - Confirmation Density (DSS-CD) stability solution. DSS-CD introduces an enhanced detection algorithm, the Confirmation Density Algorithm (CDA) to the Option III stability solution, which reliably detects the inception of power oscillations and generates an early power suppression trip signal prior to any significant oscillation amplitude growth and MCPR degradation.

References 12 describes DSS-CD and licensing basis for the CDA. It also describes the DSS-CD Armed Region and the three additional algorithms for detecting thermal-hydraulic instability related neutron flux oscillations: (1) the period based detection algorithm (PBDA), (2) the amplitude based algorithm (ABA), and (3) the growth rate algorithm (GRA). All four algorithms are implemented in the OPRM Upscale trip function; however the safety analysis takes credit only for the CDA. The remaining three algorithms provide defense-in-depth and additional protection against unanticipated oscillations. OPRM Upscale trip function OPERABILITY is based only on the CDA.

The hardware design is unchanged from the Option III solution described in Reference 15 while the firmware/software is modified relative to Option III to reflect the CDA to the Option III algorithms.

The OPRM Upscale Function receives input signals from the LPRMs, which are combined into "cells" for evaluation by the OPRM algorithms DSS-CD operability requires at least eight responsive OPRM cells per channel.

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO and APPLICABILITY (continued)

The OPRM Upscale Function is required to be OPERABLE when the plant is ≥ 16.8% RTP, which is established as a power level that is greater than or equal to 5% below the lower boundary of the Armed Region. This requirement is designed to encompass the region of power-flow operation where anticipated events could lead to thermal-hydraulic instability and related neutron flux oscillations. The OPRM Upscale Function is automatically trip-enabled when THERMAL POWER, as indicated by the APRM Simulated Thermal Power, is ≥ to 21.8% RTP corresponding to the plant-specific MCPR monitoring threshold and reactor recirculation drive flow, is < 75% of rated flow. This region is the OPRM Armed Region. Note **h** allows for entry into the DSS-CD Armed Region without automatic arming of DSS-CD prior to completely passing though the DSS-CD Armed Region during both a single startup and a single shutdown following DSS-CD implementation.

An OPRM Upscale trip is issued from an OPRM channel when the CDA in that channels detects oscillatory changes in the neutron flux indicated by periodic confirmations and amplitude exceeding specified setpoints for a specified number of OPRM cells in the channel. An OPRM Upscale trip is also issued from the channel if any of the defense-in-depth algorithms (PBDA, ABA, and GRA) exceed trip conditions for one or more cells in that channel.

Three of the four channels are required to be OPERABLE. Each channel is capable of detecting thermal-hydraulic instabilities, by detecting the related neutron flux oscillations, and issuing a trip signal before the MCPR SL is exceeded

There is no Allowable Value for this function. The setpoint for the OPRM Upscale CDA is specified in the COLR.

The OPRM Upscale function settings are not traditional instrumentation setpoints determined under an instrument setpoint methodology. In accordance with the NRC Safety Evaluation for Amendment 188 (Reference 13), the OPRM Upscale trip function is not LSSS SL-related. Reference 20 confirms the OPRM Upscale trip function settings based on DSS-CD also do not have traditional instrumentation setpoints determined under an instrument setpoint methodology.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of a scram.

Alternately, if it is not desired to place the inoperable channels (or one trip system) in trip (e.g., as in the case where placing the inoperable channel or associated trip system in trip would result in a scram or RPT), Condition D must be entered and its Required Action taken.

As noted, Condition B is not applicable to APRM Functions 2.a, 2.b, 2.c, 2.d, or 2.f. Inoperability of one required APRM/OPRM channel affects both trip systems and is not associated with a specific trip system, as are the APRM 2-Out-Of-4 Voter and other non-APRM/OPRM channels for which Condition B applies. For an inoperable APRM/OPRM channel, Required Action A.1 must be satisfied, and is the only action (other than restoring OPERABILITY) that will restore capability to accommodate a single failure. Inoperability of more than one required APRM/OPRM channel of the same trip function results in loss of trip capability and entry into Condition C, as well as entry into Condition A for each channel. Because Conditions A and C provide Required Actions that are appropriate for the inoperability of APRM Functions 2.a, 2.b, 2.c, 2.d, and 2.f, and these functions are not associated with specific trip systems as are the APRM 2-Out-Of-4 Voter and other non-APRM channels, Conditions B does not apply.

C.1

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same trip system for the same Function result in the Function not maintaining RPS trip capability. A Function is considered to be maintaining RPS trip capability when sufficient channels are OPERABLE or in trip (or the associated trip system is in trip), such that both trip systems will generate a trip signal from the given Function on a valid signal. For the typical Function with one-out-of-two taken twice logic and the IRM and APRM Functions, this would require both trip systems to have one channel OPERABLE or in trip (or the associated trip system

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.1 and SR 3.3.1.1.19

Performance of the CHANNEL CHECK ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift on one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The agreement criteria include an expectation of overlap when transitioning between neutron flux instrumentation. The overlap between SRMs and IRMs must be demonstrated prior to withdrawing SRMs from the fully inserted position since indication is being transitioned from SRMs to the IRMs. This will ensure that reactor power will not be increased into a neutron flux region without adequate indication. The overlap between IRMs and APRMs is of concern when reducing power into the IRM range. On power increases, the system design will prevent further increases (by initiating a rod block) if adequate overlap is not maintained.

Overlap between IRMs and APRMs exists when sufficient IRMs and APRMs concurrently have on-scale readings such that the transition between MODE 1 and MODE 2 can be made without either APRM downscale rod block, or IRM upscale rod block. Overlap between SRMs and IRMs similarly exists when, prior to withdrawing the SRMs from the fully inserted position, IRMs are above 2/40 on range 1 before SRMs have reached the upscale rod block.

If overlap for a group of channels is not demonstrated (e.g., IRM/APRM overlap), the reason for the failure of the Surveillance should be determined and the appropriate channel(s) that are required in the current MODE or condition should be declared inoperable.

The Frequency of once every 12 hours for SR 3.3.1.1.is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.22 (continued)

In addition to these commitments, Reference 15 require that testing inputs to each RPS Trip System alternate.

Combining these frequency requirements, an acceptable test sequence is one that:

a. Tests each RPS trip system interface every other cycle,

b. Alternates testing APRM and OPRM outputs from any specific 2-Out-Of-4 Voter channel, and

c. Alternates between divisions at least every other test cycle.

Each test of an APRM or OPRM output tests each of the redundant outputs from the 2-Out-Of-4 Voter channel for that Function and each of the corresponding relays in RPS. Consequently, each of the RPS relays is tested every fourth cycle. The RPS relay testing frequency is twice the frequency justified by Reference 15.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.1.4

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The system functional test of the pump breakers is included as a part of this test, overlapping the LOGIC SYSTEM FUNCTIONAL TEST, to provide complete testing of the associated safety function. Therefore, if a breaker is incapable of operating, the associated instrument channel would also be inoperable.

The 24 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 test when performed at the 24 month Frequency.

SR 3.3.4.1.5

This SR ensures that an EOC-RPT initiated from the TSV Closure, Trip Oil Pressure - Low and TCV Fast Closure, Trip Oil Pressure - Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 35.4% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 35.4% RTP, the affected TSV Closure, Trip Oil Pressure - Low and TCV Fast Closure, Trip Oil Pressure - Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met and the channel considered OPERABLE.

The Frequency of 24 months has shown that channel bypass failures between successive tests are rare.

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BASES

BACKGROUND

Automatic Depressurization System (continued)

confirmed Reactor Vessel Water Level - Low, Level 3 in each of the two ADS trip systems. Each of these transmitters connects to a trip unit, which then drives a relay whose contacts form the initiation logic.

Each ADS trip system (trip system A and trip system B) includes a time delay between satisfying the initiation logic and the actuation of the ADS valves. The time delay chosen is long enough that the HPCS has time to operate to recover to a level above Level 1, yet not so long that the LPCI and LPCS systems are unable to adequately cool the fuel if the HPCS fails to maintain level. An alarm in the control room is annunciated when either of the timers is running. Resetting the ADS initiation signals resets the ADS Initiation Timers.

The ADS also monitors the discharge pressures of the three LPCI pumps and the LPCS pump. Each ADS trip system includes two discharge pressure permissive transmitters from each of the two low pressure ECCS pumps in the associated Division (i.e., Division 1 ECCS inputs to ADS trip system A and Division 2 ECCS inputs to ADS trip system B). The signals are used as a permissive for ADS actuation, indicating that there is a source of core coolant available once the ADS has depressurized the vessel. Any one of the four low pressure pumps provides sufficient core coolant flow to permit automatic depressurization.

The ADS logic in each trip system is arranged in two strings. One string has a contact from each of the following variables: Reactor Vessel Water Level - Low Low Low, Level 1; Drywell Pressure - High or ADS Bypass Timer; Reactor Vessel Water Level - Low, Level 3; ADS Initiation Timer; and two low pressure ECCS Discharge Pressure -High contacts. The other string has a contact from each of the following variables: Reactor Vessel Water Level - Low Low Low, Level 1; Drywell Pressure - High or ADS Bypass Timer; and two low pressure ECCS Discharge Pressure - High contacts. To initiate an ADS trip system, the following applicable contacts must close in the associated string: Reactor Vessel Water Level - Low Low Low, Level 1; Drywell Pressure - High or ADS Bypass Timer; Reactor Vessel Water Level - Low, Level 3; ADS Initiation Timer; and one of the two low pressure ECCS Discharge Pressure - High contacts.

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B 3.3 INSTRUMENTATION

B 3.3.8.2 Reactor Protection System (RPS) Electric Power Monitoring

BASES

BACKGROUND

The RPS Electric Power Monitoring System is provided to isolate the RPS bus from the motor generator (MG) set or an alternate power supply in the event of overvoltage, undervoltage, or underfrequency. This system protects the loads connected to the RPS bus against unacceptable voltage and frequency conditions (Ref. 1) and forms an important part of the primary success path for the essential safety circuits. RPS buses supply power to the RPS logic, scram solenoids, MSIV isolation solenoids, and other essential equipment.

The RPS Electric Power Monitoring assembly will detect any abnormal high or low voltage or low frequency condition in the outputs of the two MG sets or the alternate power supply and will de-energize its respective RPS bus, thereby causing all safety functions normally powered by this bus to de-energize.

In the event of failure of an RPS Electric Power Monitoring System (e.g., both inseries electric power monitoring assemblies), the RPS loads may experience significant effects from the unregulated power supply. Deviation from the nominal conditions, such as an unregulated power supply can cause damage to the scram solenoids and other Class 1E devices.

In the event of a low voltage condition for an extended period of time, the scram solenoids can chatter and potentially lose their pneumatic control capability, resulting in a loss of primary scram action.

In the event of an overvoltage condition, the RPS logic relays and scram solenoids, as well as the main steam isolation valve solenoids, may experience a voltage higher than their design voltage. If the overvoltage condition persists for an extended time period, it may cause equipment degradation and the loss of plant safety function.

Two redundant Class 1E circuit breakers are connected in series between each RPS bus and its MG set, and between each RPS bus and its alternate power supply. Each of these

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.4.3 (continued)

verify that the valve is functioning properly. This SR can be demonstrated by one of two methods. If performed by method 1), plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements (Ref. 6), prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. If performed by method 2), valve OPERABILITY has been demonstrated for all installed S/RVs based upon the successful operation of a test sample of S/RVs.

1. Manual actuation of the S/RV, with verification of the response of the turbine control valves or bypass valves, by a change in the measured steam flow, or any other method suitable to verify steam flow (e.g., tailpipe temperature or pressure). Adequate reactor steam pressure must be available to perform this test to avoid damaging the valve. Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the S/RVs divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is to be performed is consistent with the pressure recommended by the valve manufacturer.

2. The sample population of S/RVs tested each refueling outage to satisfy SR 3.4.4.1 will be stroked in the relief mode during "as-found" testing to verify proper operation of the S/RV. Just prior to installation of the to be newly-installed S/RVs to satisfy 3.4.4.1 the valve will be stroked in the relief mode during certification testing to verify proper operation of the S/RV. The successful performance of the test sample of S/RVs will perform in a similar fashion. After the S/RVs are replaced, the electrical and pneumatic connections shall be verified either through mechanical / electrical inspection or test prior to the resumption of electric power generation to ensure that no damage has occurred to the S/RV during transportation and installation.

   This verifies that each replaced S/RV will properly perform it intended function.

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BASES

BACKGROUND (continued)	This Specification ensures that the performance of the primary containment, in the event of a DBA, meets the assumptions used in the safety analyses of References 1 and 2. SR 3.6.1.1.1 leakage rate requirements are in conformance with 10 CFR 50, Appendix J (Ref. 3), as modified by approved exemptions.
APPLICABLE SAFETY ANALYSES	The safety design basis for the primary containment is that it must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate. The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage. Analytical methods and assumptions involving the primary containment are presented in References 1 and 2. The safety analyses assume a nonmechanistic fission product release following a DBA, which forms the basis for determination of offsite doses. The fission product release is, in turn, based on an assumed leakage rate from the primary containment. OPERABILITY of the primary containment ensures that the leakage rate assumed in the safety analyses is not exceeded. The maximum allowable leakage rate for the primary containment (L_a) including MSIV Leakage is 0.682% by weight of the containment and drywell air per 24 hours at the maximum peak containment pressure (P_a) of 12.1 psig (Ref. 4). Primary containment satisfies Criterion 3 of the NRC Policy Statement.
LCO	Primary containment OPERABILITY is maintained by limiting leakage to $\leq$ 1.0 L_a, except prior to the first unit startup after performing a required 10 CFR 50, Appendix J leakage test. At this time, the combined Type B and Type C leakage must be < 0.6 L_a, and the overall Type A leakage must be < 0.75 L_a. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.7 (continued)

each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.1.7 overlaps this SR to provide complete testing of the safety function. The 24 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 that these components usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.8

The analyses in Reference 2 is based on leakage that is less than the specified leakage rate. Leakage through any single main steam line must be ≤ 100 scfh when tested at a pressure of 12.1 psig. Leakage through all four steam lines must be ≤ 250 scfh when tested at P_a (12.1 psig). The MSIV leakage rate must be verified to be in accordance with the leakage test requirements of Reference 3, as modified by approved exemptions. A Note is added to this SR which states that these valves are only required to meet this leakage limit in MODES 1, 2 and 3. In the other conditions, the Reactor Coolant System is not pressurized and specific primary containment leakage limits are not required.

SR 3.6.1.3.9

Surveillance of hydrostatically tested lines provides assurance that the calculation assumptions of Reference 2 is met.

This SR is modified by a Note that states these valves are only required to meet the combined leakage rate in MODES 1, 2, and 3 since this is when the Reactor Coolant System is

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ACTIONS

D.1 (continued)

Required Action D.1 is modified by a Note that states that LCO 304a is not applicable when entering MODE 3. This Note prohibits the use of LCO 304a to enter MODE 3 during startup with the LCO not met. However, there is no restriction on the use of LCO 304b, if applicable, because LCO 304b requires performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering MODE 3, and establishment of risk management actions, if appropriate. LCO 304 is not applicable to, and the Note does not preclude, changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1 and E.2

When two SGT subsystems are inoperable, if applicable, movement of recently irradiated fuel assemblies in the primary and secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating each SGT subsystem from the control room for ≥ 15 continuous minutes ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

SR 3.6.4.3.2

This SR verifies that the required SGT filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations).

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