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SOUTHERN COMPANY
Energy to Serve Your World℠

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50-425

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

Ladies and Gentlemen:

VOGTLE ELECTRIC GENERATING PLANT
CHANGES TO TECHNICAL SPECIFICATION BASES

The Vogtle Electric Generating Plant (VEGP) Unit 1 and Unit 2 Technical Specifications, section 5.5.14, Technical Specifications (TS) Bases Control Program, provide for changes to the Bases without prior NRC approval. In addition, TS section 5.5.14 requires that Bases changes made without prior NRC approval be provided to the NRC on a frequency consistent with 10 CFR 50.71 (e). Pursuant to TS section 5.5.14, Southern Nuclear Operating Company hereby submits Bases changes made to the VEGP TS Bases under the provisions of TS section 5.5.14. This submittal reflects changes since April 20, 1998 through November 5, 1999.

Sincerely,

J. B. Beasley, Jr.

JBB/NJS

Enclosure: Bases Changes

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BASES

LCO 3.0.3 (continued)

an Applicability of "During movement of irradiated fuel assemblies in the fuel storage pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.15 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.15 of "Suspend movement of irradiated fuel assemblies in the fuel storage pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

LCO 3.0.4 establishes limitations on changes in MODES or other specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a MODE or other specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Unit conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the unit being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a MODE or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the MODE change. Therefore, in such cases, entry into a MODE or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

(continued)

BASES

LCO 3.0.4 (continued)

The provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 3.0.4 is only applicable for MODE changes when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODES 1, 2, 3, or 4. The requirements of LCO 3.0.4 do not apply in MODES 5 and 6, or in other specfified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken. In some cases (e.g., LCO 3.1.1) these ACTIONS provide a Note that states "While this LCO is not met, entry into a MODE or other specified condition in the Applicability is not permitted, unless required to comply with ACTIONS." This Note is a requirement explicitly precluding entry into a MODE or other specified condition of the Applicability.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, in compliance with LCO 3.0.4 or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

(continued)

BASES

LCO 3.0.5 (continued)

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed SRs. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the SRs.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of an SR on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of an SR on another channel in the same trip system.

LCO 3.0.6

LCO 3.0.6 establishes an exception to LCO 3.0.2 for support systems that have an LCO specified in the Technical Specifications (TS). This exception is provided because LCO 3.0.2 would require that the Conditions and Required Actions of the associated inoperable supported system LCO be entered solely due to the inoperability of the support system. This exception is justified because the actions that are required to ensure the unit is maintained in a safe condition are specified in the support system LCO's Required Actions. These Required Actions may include entering the supported system's Conditions and Required Actions or may specify other Required Actions.

When a support system is inoperable and there is an LCO specified for it in the TS, the supported system(s) are required to be declared inoperable if determined to be inoperable as a result of the support system inoperability. However, it is not necessary to enter into the supported systems' Conditions and Required Actions unless directed to do so by the support system's Required Actions. The potential confusion and inconsistency of requirements related to the entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the unit is maintained in a safe condition in the support system's Required Actions.

(continued)

BASES

SR 3.0.4 (continued)

Applicability for which these systems and components ensure safe operation of the unit.

The provisions of this Specification should not be interpreted as endosing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a MODE change or other specified condition change. When a system, subsystem, division, component, device, or variable is inoperable or outside its specified limits, the associated SR(s) is not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on inoperable equipment. When equipment is inoperable, SR 3.0.4 does not apply to the associated SR(s), since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillance(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing MODES or other specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the MODE or other specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been, reached. Further discussion of the specific formats of SRs annotation is found in Section 1.4, Frequency.

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BASES

SR 3.0.4 (continued)

SR 3.0.4 is only applicable for MODE changes when entering MODE 4 from MODE 5, MODE 3 from MODE 4, MODE 2 from MODE 3, or MODE 1 from MODE 2. Furthermore, SR 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating MODES 1, 2, 3, or 4. The requirements of SR 3.0.4 do not apply in MODES 5 and 6, or in other specified conditions of the Applicability (unless in MODES 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

BASES

BACKGROUND

Signal Process Control and Protection System (continued)

the other channels providing the protection function actuation. Again, a single failure will neither cause nor prevent the protection function actuation. These requirements are described in IEEE-279-1971 (Ref. 4). The actual number of channels required for each unit parameter is specified in Reference 1.

Two logic channels are required to ensure no single random failure of a logic channel will disable the RTS. The logic channels are designed such that testing required while the reactor is at power may be accomplished without causing trip. Provisions to allow removing logic channels from service during maintenance are unnecessary because of the logic system's designed reliability.

Trip Setpoints and Allowable Values

The Trip Setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION tolerance.

The Trip Setpoints used in the bistables are based on the analytical limits stated in Reference 1. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those RTS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 5), the Trip Setpoints and Allowable Values specified in Table 3.3.1-1 in the accompanying LCO are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the Trip Setpoints, including their explicit uncertainties, is provided in the "RTS/ESFAS Setpoint Methodology Study" (Ref. 6). The actual nominal Trip Setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a COT. One example of such a change in measurement error is drift during the surveillance interval.

(continued)

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

Setpoints in accordance with the Allowable Value ensure that SLs are not violated during AOOs (and that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed). For the purpose of demonstrating compliance with 10 CFR 50.36 to the extent that the Technical Specifications are required to specify Limiting Safety System Settings (LSSS), the LSSS for VEGP are comprised of both the Nominal Trip Setpoints and the Allowable Values specified in Table 3.3.1-1. The Nominal Trip Setpoint is the expected value to be achieved during calibrations. The Nominal Trip Setpoint considers all factors which may affect channel performance by statistically combining rack drift, rack measurement and test equipment effects, rack calibration accuracy, rack comparator setting accuracy, rack temperature effects, sensor measurement and test equipment effects, sensor calibration accuracy, primary element accuracy, and process measurement accuracy. The Nominal Trip Setpoint is the value that will always ensure that safety analysis limits are met (with margin) given all of the above effects. The Allowable Value has been established by considering the values assumed for rack effects only. The Allowable Value serves as an operability limit for the purpose of the quarterly CHANNEL OPERATIONAL TESTS.

Each channel of the process control equipment can be tested on line to verify that the signal or setpoint accuracy is within the specified allowance requirements of Reference 2. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. SRs for the channels are specified in the SRs section.

The Trip Setpoints and Allowable Values listed in Table 3.3.1-1 are based on the methodology described in Reference 6, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal

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BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

Solid State Protection System

The SSPS equipment is used for the decision logic processing of outputs from the signal processing equipment bistables. To meet the redundancy requirements, two trains of SSPS, each performing the same functions, are provided. If one train is taken out of service for maintenance or test purposes, the second train will provide reactor trip and/or ESF actuation for the unit. If both trains are taken out of service or placed in test, a reactor trip will result. Each train is packaged in its own cabinet for physical and electrical separation to satisfy separation and independence

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BASES

BACKGROUND

Reactor Trip Switchgear (continued)

trip mechanism is sufficient by itself, thus providing a diverse trip mechanism.

The decision logic matrix Functions are described in the functional diagrams included in Reference 1. In addition to the reactor trip or ESF, these diagrams also describe the various "permissive interlocks" that are associated with unit conditions. Each train has a built in testing device that can automatically test the decision logic matrix Functions and the actuation devices while the unit is at power. When any one train is taken out of service for testing, the other train is capable of providing unit monitoring and protection until the testing has been completed. The testing device is semiautomatic to minimize testing time.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The RTS functions to maintain the SLs during all AOOs and mitigates the consequences of DBAs in all MODES in LCO, and which the RTBs are closed.

Each of the analyzed accidents and transients can be detected by one or more RTS Functions. The accident analysis described in Reference 3 takes credit for most RTS trip Functions. RTS trip Functions not specifically credited in the accident analysis are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the unit. These RTS trip Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. They may also serve as backups to RTS trip Functions that were credited in the accident analysis.

The LCO requires all instrumentation performing an RTS Function, listed in Table 3.3.1-1 in the accompanying LCO, to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions. The Nominal Trip Setpoint column is modified by a Note that requires the as-left condition for a channel to be within the calibration tolerance for that channel. In addition, the as-left condition may be more conservative than the specified Nominal Trip Setpoint.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The conservative direction is established by the direction of the inequality applied to the Allowable Value. It is consistent with the setpoint methodology for the as-left trip setpoint to be outside the calibration tolerance but in the conservative direction with respect to the Nominal Trip Setpoint. For example, the Power Range Neutron Flux High trip setpoint may be set to a value less than 109% during initial startup following a refueling outage until a sufficiently high reactor power is achieved so that the power range channels may be calibrated. In addition, certain Required Actions may require that the Power Range Neutron Flux High trip setpoints and/or the Overpower Delta-T setpoints be reduced based on plant conditions.

The LCO generally requires OPERABILITY of four or three channels in each instrumentation Function, two channels of Manual Reactor Trip in each logic Function, and two trains in each Automatic Trip Logic Function. Four OPERABLE

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4. Intermediate Range Neutron Flux (continued)

Above the P-10 setpoint, the Power Range Neutron Flux—High Setpoint trip and the Power Range Neutron Flux—High Positive Rate trip provide core protection for a rod withdrawal accident. In MODE 3, 4, or 5, the Intermediate Range Neutron Flux trip does not have to be OPERABLE because the reactor cannot be started up in this condition. The core also has the required SDM to mitigate the consequences of a positive reactivity addition accident. In MODE 6, all rods are fully inserted and the core has a required increased SDM. Also, the NIS intermediate range indication is typically low off-scale in this MODE.

5. Source Range Neutron Flux

The LCO requirement for the Source Range Neutron Flux trip (NI-0031B, D, & E, NI-0032B, D, & G) Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to the Power Range Neutron Flux — Low Setpoint and Intermediate Range Neutron Flux trip Functions. In MODES 3, 4, and 5, administrative controls also prevent the uncontrolled withdrawal of rods. The NIS source range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS source range detectors do not provide any inputs to control systems. The source range trip is the only RTS automatic protection function required in MODES 3, 4, and 5. Therefore, the functional capability at the specified Trip Setpoint is assumed to be available.

The LCO requires two channels of Source Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function. The LCO also requires two channels of the Source Range Neutron Flux to be OPERABLE in MODE 3, 4, or 5 with RTBs closed.

The Source Range Neutron Flux Function provides protection for control rod withdrawal from

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5. Source Range Neutron Flux (continued)

subcritical, boron dilution (see LCO 3.3.8) and control rod ejection events. The Function also provides visual neutron flux indication in the control room.

In MODE 2 when below the P-6 setpoint during a reactor startup, the Source Range Neutron Flux trip must be OPERABLE. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux — Low Setpoint trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the Source Range Neutron Flux trip is blocked.

In MODE 3, 4, or 5 with the reactor shut down, the Source Range Neutron Flux trip Function must also be OPERABLE. If the Rod Control System is capable of rod withdrawal, the Source Range Neutron Flux trip must be OPERABLE to provide core protection against a rod withdrawal accident. If the Rod Control System is not capable of rod withdrawal, the source range detectors are not required to trip the reactor. Source range detectors also function to monitor for high flux at shutdown. This function is addressed in Specification 3.3.8. Requirements for the source range detectors in MODE 6 are addressed in LCO 3.9.3.

6. Overtemperature ΔT

The Overtemperature ΔT trip Function (TDI-0411C, TDI-0421C, TDI-0431C, TDI-0441C, TDI-0411A, TDI-0421A, TDI-0431A, TDI-0441A) is provided to ensure that the design limit DNBR is met. This trip Function also limits the range over which the Overpower ΔT trip Function must provide protection. The inputs to the Overtemperature ΔT trip include pressure, coolant temperature, axial power distribution, and reactor power as indicated by loop ΔT assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. The Function monitors both variation in power and flow since a decrease in flow

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Overtemperature ΔT (continued)

This results in a two-out-of-four trip logic. Section 7.2.2.3 of Reference 1 discusses control and protection system interactions for this function. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent a reactor trip.

Delta-T_0, as used in the overtemperature and overpower ΔT trips, represents the 100% RTP value as measured for each loop. This normalizes each loop's ΔT trips to the actual operating conditions existing at the time of measurement, thus forcing the trip to reflect the equivalent full power conditions as assumed in the accident analyses. These differences in RCS loop ΔT can be due to several factors, e.g., differences in RCS loop flows and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT values. Therefore, loop specific ΔT_0 values are measured as needed to ensure they represent actual core conditions.

The parameter K_1 is the principal setpoint gain, since it defines the function offset. The parameters K_2 and K_3 define the temperature gain and pressure gain, respectively. The values for T' and P' are key reference parameters corresponding directly to plant safety analyses initial conditions assumptions for the Overtemperature ΔT function. For the purposes of performing a CHANNEL CALIBRATION, the values for K_1, K_2, K_3, T', and P' are utilized in the safety analyses without explicit tolerances, but should be considered as nominal values for instrument settings. That is, while an exact setting is not expected, a setting as close as reasonably possible is desired. Note that for T', the value for the hottest RCS loop will be set

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABLITY

6. Overtemperature ΔT (continued)

as close as possible to 588.4° F. The value of T' for the remaining RCS loops will be set appropriately less than 588.4°F based on the actual loop specific indicated T_{avg}. The engineering scaling calculations use each of the referenced parameters as an exact gain or reference value. Tolerances are not applied to the individual gain or reference parameters. Tolerances are applied to each calibration module and the overall string calibration. In order to ensure that the Overtemperature ΔT setpoint is consistent with the assumptions of the safety analyses, it is necessary to verify during the CHANNEL OPERATIONAL TEST that the Overtemperature ΔT setpoint is within the appropriate calibration tolerances for the defined calibration conditions (Ref. 9).

The LCO requires all four channels of the Overtemperature ΔT trip Function to be OPERABLE. Note that the Overtemperature ΔT Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Overpower ΔT (continued)

Delta-T_0, as used in the overtemperature and overpower ΔT trips, represents the 100% RTP value as measured for each loop. This normalizes each loop's ΔT trips to the actual operating conditions existing at the time of measurement, thus forcing the trip to reflect the equivalent full power conditions as assumed in the accident analyses. These differences in RCS loop ΔT can be due to several factors, e.g., difference in RCS loop flows and slightly asymmetric power distributions between quadrants. While RCS loop flows are not expected to change with cycle life, radial power redistribution between quadrants may occur, resulting in small changes in loop specific ΔT values. Therefore, loop specific ΔT_0 values are measured as needed to ensure they represent actual core conditions.

The value for T" is a key reference parameter corresponding directly to plant safety analyses initial conditions assumptions for the Overpower ΔT function. For the purposes of performing a CHANNEL CALIBRATION, the values for K_4, K_5, K_6, and T" are utilized in the safety analyses without explicit tolerances, but should be considered as nominal values for instrument settings. That is, while an exact setting is not expected, a setting as close as reasonably possible is desired. Note that for T", the value for the hottest RCS loop will be set as close as possible to 588.4° F. The value of T" for the remaining RCS loops will be set appropriately less than 588.4°F based on the actual loop specific indicated T_{avg}. The engineering scaling calculations use each of the referenced parameters as an exact gain or reference value. Tolerances are not applied to the individual gain or reference parameters. Tolerances are applied to each calibration module and the overall string calibration. In order to ensure that the Overpower ΔT setpoint is consistent with the assumptions of the safety analyses, it is necessary to verify during the CHANNEL OPERATIONAL TEST that the Overpower ΔT setpoint is within the appropriate calibration tolerances for defined calibration conditions (Ref. 9). Note that for the parameter K_5,

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Overpower ΔT (continued)

in the case of decreasing temperature, the gain setting must be $\geq$ 0 to prevent generating setpoint margin on decreasing temperature rates. Similarly, the setting for K_6 is required to be equal to 0 for conditions where T $\leq$ T".

The LCO requires four channels of the Overpower ΔT trip Function to be OPERABLE. Note that the Overpower ΔT trip Function receives input from channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

8. Pressurizer Pressure

The same sensors (PI-0455A, B, & C, PI-0456, PI-0456A, PI-0457, PI-0457A, PI-0458, PI-0458A) provide input to the Pressurizer Pressure — High and — Low trips and the Overtemperature ΔT trip. Since the Pressurizer Pressure channels are also used to provide input to the Pressurizer Pressure Control System, the actuation logic must be able to withstand an input failure to

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

16. Reactor Trip System Interlocks

Reactor protection interlocks are provided to ensure reactor trips are in the correct configuration for the current unit status. They back up operator actions to ensure protection system Functions are not bypassed during unit conditions under which the safety analysis assumes the Functions are not bypassed. Therefore, the interlock Functions do not need to be OPERABLE when the associated reactor trip functions are outside the applicable MODES. These are:

a. Intermediate Range Neutron Flux, P-6

The Intermediate Range Neutron Flux, P-6 interlock (NI-0035B, D, & E, NI-0036B, D, & G) is actuated when any NIS intermediate range channel goes approximately one decade above the minimum channel reading. If both channels drop below the setpoint, the permissive will automatically be defeated. The LCO requirement for the P-6 interlock ensures that the following Functions are performed:

- on increasing power, the P-6 interlock allows the manual block of the NIS Source Range, Neutron Flux reactor trip. This prevents a premature block of the source range trip and allows the operator to ensure that the intermediate range is OPERABLE prior to leaving the source range.
- on decreasing power, the P-6 interlock automatically enables the NIS Source Range Neutron Flux reactor trip.

The LCO requires two channels of Intermediate Range Neutron Flux, P-6 interlock to be OPERABLE in MODE 2 when below the P-6 interlock setpoint.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

d. Power Range Neutron Flux, P-9 (continued)

reactor is not at a power level sufficient to have a load rejection beyond the capacity of the Steam Dump System.

e. Power Range Neutron Flux, P-10

The Power Range Neutron Flux, P-10 interlock (NI-0041B & C, NI-0042B & C, NI-0043B & C, NI-0044B & C) is actuated at approximately 10% power, as determined by two-out-of-four NIS power range detectors. If power level falls below 10% RTP on 3 of 4 channels, the nuclear instrument trips will be automatically unblocked. The LCO requirement for the P-10 interlock ensures that the following Functions are performed:

- on increasing power, the P-10 interlock allows the operator to manually block the Intermediate Range Neutron Flux reactor trip. Note that blocking the reactor trip also blocks the signal to prevent automatic and manual rod withdrawal;
- on increasing power, the P-10 interlock allows the operator to manually block the Power Range Neutron Flux — Low reactor trip;
- on increasing power, the P-10 interlock automatically provides a backup signal to block the Source Range Neutron Flux reactor trip;
- the P-10 interlock provides one of the two inputs to the P-7 interlock; and
- on decreasing power, the P-10 interlock automatically enables the Power Range Neutron Flux — Low reactor trip and the Intermediate Range Neutron Flux reactor trip (and rod stop).

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BASES

ACTIONS

C.1 and C.2 (continued)

- Manual Reactor Trip;
- RTBs;
- RTB Undervoltage and Shunt Trip Mechanisms; and
- Automatic Trip Logic.

This action addresses the train orientation of the SSPS for these Functions. With one channel or train inoperable, the inoperable channel or train must be restored to OPERABLE status within 48 hours. If the affected Function(s) cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be placed in a MODE in which the requirement does not apply. To achieve this status, the RTBs must be opened within the next hour. The additional hour provides sufficient time to accomplish the action in an orderly manner. With the RTBs open, these Functions are no longer required. This Condition is modified by a Note that prohibits closing the RTBs in MODE 5 if any of the above Functions (Function 1, 17, 18, or 19 of Table 3.3.1-1) are not met. Closing the RTBs in MODES 3 or 4 with any of these Functions not met is prohibited by LCO 3.0.4.

The Completion Time is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function, and given the low probability of an event occurring during this interval.

D.1.1, D.1.2, D.2.1, D.2.2, and D.3

Condition D applies to the Power Range Neutron Flux—High Function.

The NIS power range detectors provide input to the CRD System and the SG Water Level Control System and, therefore, have a two-out-of-four trip logic. A known inoperable channel must be placed in the tripped condition. This results in a partial trip condition requiring only one-out-of-three logic for actuation. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in WCAP-10271-P-A (Ref. 7).

In addition to placing the inoperable channel in the tripped condition, THERMAL POWER must be reduced to ≤ 75% RTP within 12 hours. Reducing the power level prevents operation of

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.10

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint methodology. The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology.

The Frequency of 18 months is based on the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology for some instrument functions, and the need to perform this Surveillance for some instrument functions under the conditions that apply during a plant outage and the potential for an unplanned plant transient if the Surveillance were performed at power. Operating experience has shown these components usually pass the Surveillance when performed on the 18 month Frequency.

SR 3.3.1.10 is modified by a Note stating that this test shall include verification that the time constants are adjusted to the prescribed values where applicable.

SR 3.3.1.11

SR 3.3.1.11 is the performance of a CHANNEL CALIBRATION, as described in SR 3.3.1.10, every 18 months. This SR is modified by a Note that states that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range neutron detectors includes a normalization of the detectors based on a power calorimetric and flux map performed above 75% RTP. The CHANNEL CALIBRATION for the source range neutron detectors includes obtaining the detector preamp discriminator curves and evaluating those curves.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.11 (continued)

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 on the 18 month Frequency.

SR 3.3.1.12

SR 3.3.1.12 is the performance of a COT of RTS interlocks every 18 months.

The Frequency is based on the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.1.13

SR 3.3.1.13 is the performance of a TADOT of the Manual Reactor Trip and the SI Input from ESFAS. This TADOT is as described in SR 3.3.1.4, except that the test is performed every 18 months.

The manual reactor trip TADOT shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the manual reactor trip function. This test shall also verify the OPERABILITY of the Bypass breaker trip circuit(s), including the automatic undervoltage trip.

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.13 (continued)

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.

SR 3.3.1.14

SR 3.3.1.14 is the performance of a TADOT of the turbine stop valve closure Turbine Trip Functions. This TADOT is as described in SR 3.3.1.4, except that this test is performed after each entry into MODE 3 for a unit shutdown and prior to exceeding the P-9 interlock trip setpoint. A Note states that this Surveillance is not required if it has been performed within the previous 31 days. Verification of the Trip Setpoint does not have to be performed for this Surveillance. Performance of this test will ensure that the turbine trip Function is OPERABLE prior to taking the reactor critical. This test cannot be performed with the reactor at power and must therefore be performed prior to reactor startup.

SR 3.3.1.15

SR 3.3.1.15 verifies that the individual channel/train actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in FSAR, Chapter 16 (Ref. 8). Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the trip setpoint value at the sensor to the point at which the equipment reaches the required functional state (i.e., control and shutdown rods fully inserted in the reactor core).

For channels that include dynamic transfer Functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer function set to one or with the time constants set to their nominal value. The results must be compared to properly defined acceptance criteria. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.15 (continued)

Response time may be verified by actual response time tests in any series of sequential, overlapping, or total channel measurements; or by the summation of allocation sensor, signal processing, and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable resonse time tests (hydraulic, noise, or power interrupt tests), (2) in place, onsite, or offsite (e.g., vendor) test measurements, or (3) using vendor engineering specifications. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements", provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

WCAP-14036-P Revision 1, "Elimination of Periodic Protection Channel Response Time Tests", provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for sensor, signal conditioning and actuation logic response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. Specific components identified in the WCAP may be replaced without verification testing. One example where response time could be affected is replacing the sensing assembly of a transmitter.

As appropriate, each channel's response must be verified every 18 months on a STAGGERED TEST BASIS. Testing of the final actuation devices is included in the testing. Response times cannot be determined during unit operation because equipment operation is required to measure response

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.15 (continued)

times. Experience has shown that these components usually pass this surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.3.1.15 is modified by a Note stating that neutron detectors are excluded from RTS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

SR 3.3.1.16

SR 3.3.1.16 is the performance of a COT for the low fluid oil pressure portion of the Turbine Trip Functions as described in SR 3.3.1.7 except that the Frequency is after each entry into MODE 3 for a unit shutdown and prior to exceeding the P-9 interlock trip setpoint. The surveillance is modified by two Notes. Note 1 states that the surveillance may be satisfied if performed within the previous 31 days. Note 2 states that verification of the setpoint is not required. The Frequency ensures that the turbine trip on low fluid oil pressure channels is OPERABLE after each unit shutdown and prior to entering the Mode of Applicability (above the P-9 power range neutron flux interlock) for this instrument function.

REFERENCES

1. FSAR, Chapter 7.

(continued)

BASES

REFERENCES (continued)

2. FSAR, Chapter 6.

3. FSAR, Chapter 15.

4. IEEE-279-1971.

5. 10 CFR 50.49.

6. WCAP-11269, Westinghouse Setpoint Methodology for Protection Systems; as supplemented by:

   - Amendments 34 (Unit 1) and 14 (Unit 2), RTS Steam Generator Water Level – Low Low, ESFAS Turbine Trip and Feedwater Isolation SG Water Level – High High, and ESFAS AFW SG Water Level – Low Low.

   - Amendments 48 and 49 (Unit 1) and Amendments 27 and 28 (Unit 2), deletion of RTS Power Range Neutron Flux High Negative Rate Trip.

   - Amendments 60 (Unit 1) and 39 (Unit 2), RTS Overtemperature ΔT setpoint revision.

   - Amendments 57 (Unit 1) and 36 (Unit 2), RTS Overtemperature and Overpower ΔT time constants and Overtemperature ΔT setpoint.

   - Amendments 43 and 44 (Unit 1) and 23 and 24 (Unit 2), revised Overtemperature and Overpower ΔT trip setpoints and allowable values.

   - Amendments 104 (Unit 1) and 82 (Unit 2), revised RTS Intermediate Range Neutron Flux, Source Range Neutron Flux, and P-6 trip setpoints and allowable values.

7. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.

8. FSAR, Chapter 16.

9. Westinghouse Letter GP-16696, November 5, 1997.

10. WCAP-13632-P-A Revision 1, “Elimination of Periodic Sensor Response Time Testing Requirements,” January 1996.

11. WCAP-14036-P-A Revision 1, “Elimination of Periodic Protection Channel Response Time Tests,” October 1998.

BASES

BACKGROUND

Sequencer Output Relays (continued)

sequencer and are part of the control circuitry of these ESF loads. There are two independent trains of sequencers and each is powered by the respective train of 120-Vac ESF electrical power supply. The power supply for the output relays is the sequencer power supply. The applicable output relays are tested in the slave relay testing procedures, and in particular, in conjunction with the specific slave relay also required to actuate to energize the applicable ESF load.

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

Each of the analyzed accidents can be detected by one or more ESFAS Functions. One of the ESFAS Functions is the primary actuation signal for that accident. An ESFAS Function may be the primary actuation signal for more than one type of accident. An ESFAS Function may also be a secondary, or backup, actuation signal for one or more other accidents. For example, Pressurizer Pressure—Low is a primary actuation signal for small loss of coolant accidents (LOCAs) and a backup actuation signal for steam line breaks (SLBs) outside containment. Functions such as manual initiation, not specifically credited in the accident safety analysis, are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the unit. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions may also serve as backups to Functions that were credited in the accident analysis (Ref. 3).

The LCO requires all instrumentation performing an ESFAS Function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions. The Nominal Trip Setpoint column is modified by a Note that requires the as-left conditions for a channel to be within the calibration tolerance for that channel. In addition, the as-left condition may be more conservative than the specified Nominal Trip Setpoint. The conservative direction is established by the direction of the inequality applied to the Allowable Value. It is consistent with the setpoint methodology for the as-left trip setpoint to be outside the calibration tolerance but in the conservative direction with respect to the Nominal Trip Setpoint.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

The LCO generally requires OPERABILITY of four or three channels in each instrumentation function and two channels in each logic and manual initiation function. The two-out-of-three and the two-out-of-four configurations allow one channel to be tripped during maintenance or testing without causing an ESFAS initiation. If an instrument channel is equipped with installed bypass capability, such that no jumpers or lifted leads are

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Engineered Safety Feature Actuation System Interlocks — Reactor Trip, P-4 (continued)

This Function must be OPERABLE in MODES 1, 2, and 3 when the reactor may be critical, approaching criticality, or the automatic SI function is required to be OPERABLE. This Function does not have to be OPERABLE in MODE 4, 5, or 6 because the main turbine, the MFW System, and the automatic SI function are not required to be OPERABLE. The P-4 function to trip the turbine and isolate main feedwater are only required in MODES 1 and 2 when these systems may be in service.

b. Engineered Safety Feature Actuation System Interlocks — Pressurizer Pressure, P-11

The P-11 interlock (PT-0455, PT-0456, PT-0457) permits a normal unit cooldown and depressurization without actuation of SI or main steam line isolation. With two-out-of-three pressurizer pressure channels (discussed previously) less than the P-11 setpoint, the operator can manually block the Pressurizer Pressure — Low and Steam Line Pressure — Low SI signals and the Steam Line Pressure — Low steam line isolation signal (previously discussed). When the Steam Line Pressure — Low steam line isolation signal is manually blocked, a main steam isolation signal on Steam Line Pressure — Negative Rate — High is enabled. This provides protection for an SLB by closure of the MSIVs. With two-out-of-three pressurizer pressure channels above the P-11 setpoint, the Pressurizer Pressure — Low and Steam Line Pressure — Low SI signals and the Steam Line Pressure — Low steam line isolation signal are automatically enabled. The operator can also enable these trips by use of the respective manual reset buttons. When the Steam Line Pressure — Low steam line isolation signal is enabled, the main steam isolation on Steam Line Pressure — Negative Rate — High is disabled. The Trip Setpoint

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.7 (continued)

The Frequency of 18 months is based on the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

This SR is modified by a Note stating that this test should include verification that the time constants are adjusted to the prescribed values where applicable. The steam line pressure-low and steam line pressure negative rate-high functions have time constants specified in their setpoints.

SR 3.3.2.8

This SR ensures the individual channel ESF RESPONSE TIMES are less than or equal to the maximum values assumed in the accident analysis. Response Time testing acceptance criteria are included in the FSAR, Chapter 16 (Ref. 8). Individual component response times are not modeled in the analyses. The analyses model the overall or total elapsed time, from the point at which the parameter exceeds the Trip Setpoint value at the sensor, to the point at which the equipment in both trains reaches the required functional state (e.g., pumps at rated discharge pressure, valves in full open or closed position).

For channels that include dynamic transfer functions (e.g., lag, lead/lag, rate/lag, etc.), the response time test may be performed with the transfer functions set to one or with the time constants set to their nominal value. The results must be compared to properly defined acceptance criteria. The response time may be measured by a series of overlapping tests such that the entire response time is measured.

Response time may be verified by actual response time tests in any series of sequential, overlapping, or total channel measurements; or by the summation of allocated sensor, signal processing, and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from:

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.8 (continued)

(1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) inplace, onsite, or offsite (e.g., vendor) test measurements, or (3) using vendor engineering specifications. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements", provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

WCAP-14036-P Revision 1, "Elimination of Periodic Protection Channel Response Time Tests", provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for sensor, signal conditioning and actuation logic response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. Specific components identified in the WCAP may be replaced without verification testing. One example where response time could be affected is replacing the sensing assembly of a transmitter.

ESF RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. Testing of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. The final actuation device in one train is tested with each channel. Therefore, staggered testing results in response time

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.8 (continued)

verification of these devices every 18 months. The 18 month Frequency is consistent with the typical refueling cycle and is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

This SR is modified by a Note that clarifies that the turbine driven AFW pump is tested within 24 hours after reaching 900 psig in the SGs.

SR 3.3.2.9

SR 3.3.2.9 is the performance of a TADOT as described in SR 3.3.2.6 for the P-4 Reactor Trip Interlock, and the Frequency is once per 18 months. This Frequency is based on operating experience. The SR is modified by a note that excludes verification of setpoints during the TADOT. The function tested has no associated setpoint.

REFERENCES

1. FSAR, Chapter 6.
2. FSAR, Chapter 7.
3. FSAR, Chapter 15.
4. IEEE-279-1971.
5. 10 CFR 50.49.
6. WCAP-11269, Westinghouse Setpoint Methodology for Protection Systems; as supplemented by:
    - Amendments 38 (Unit 1) and 18 (Unit 2), ESFAS Safety Injection Pressurizer — Low allowable value revision.
    - Amendments 34 (Unit 1) and 14 (Unit 2), RTS Steam Generator Water Level — Low Low, ESFAS Turbine Trip and Feedwater Isolation SG Water Level — High High, and ESFAS AFW SG Water Level — Low Low.

(continued)

BASES

REFERENCES (continued)

- Amendments 43 and 44 (Unit 1) and 23 and 24 (Unit 2), revised ESFAS Interlocks Pressurizer P-11 trip setpoint and allowable value.

7. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.

8. FSAR, Chapter 16.

9. Westinghouse Letter GP-16696, November 5, 1997.

10. WCAP-13632-P-A Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," January 1996.

11. WCAP-14036-P-A Revision 1, "Elimination of Periodic Protection Channel Response Time Tests," October 1998.

BASES (continued)

TABLE B 3.3.4-1

REMOTE SHUTDOWN SYSTEM MONITORING INSTRUMENTATION

	INSTRUMENT FUNCTION	READOUT[1] LOCATION	CHANNELS AVAILABLE
1.	Source Range Neutron Flux	A	1 (NI-31E)
2.	Extended Range Neutron Flux	B	1 (NI-13135 C&D)
3.	RCS Cold Leg Temperature	A, B	1/Loop (Loop 1 TI-0413D, Panel A) (Loop 2 TI-0423D, Panel B) (Loop 3 TI-0433D, Panel B) (Loop 4 TI-0443D, Panel A)
4.	RCS Hot Leg Temperature	A	2 (Loop 1 TI-0413C Loop 4 TI-0443C)
5.	Core Exit Thermocouples	B	2 (Loop 2 Core Quadrant 1TI-10055) (Loop 3 Core Quadrant 1TI-10056) (Loop 1 Core Quadrant 2TI-10055) (Loop 4 Core Quadrant 2TI-10056)
6.	RCS Wide Range Pressure	A, B	2 (PI-405A, Panel A) (PI-403A, Panel B)
7.	Steam Generator Level Wide Range	A, B	1/Loop (Loop 1 LI-501B, Panel A) (Loop 2 LI-502B, Panel B) (Loop 3 LI-503B, Panel B) (Loop 4 LI-504B, Panel A)
8.	Pressurizer Level	A, B	2 (LI-459C, Panel A) (LI-460C, Panel B)
9.	RWST Level	L	1 (LI-0990C)
10.	BAST Level	L	1 (PI-10115)[2]
11.	CST Level	L	2 (Tank 1 LI-5100) (Tank 2 LI-5115)

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event, within approximately 60 seconds. The isolation of the purge supply and exhaust valves has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed. The containment purge supply and exhaust isolation radiation monitors act as backup to the SI signal to ensure closing of the purge supply and exhaust valves for events occurring in MODES 1 through 4. Manual isolation (using individual valve handswitches) following a radiation alarm is the assumed means for isolating containment in the event of a fuel handling accident during shutdown. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accidental offsite radiological doses are below 10 CFR 100 (Ref. 1) limits.

The containment ventilation isolation instrumentation satisfies Criterion 3 of the NRC Policy Statement.

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Ventilation Isolation, listed in Table 3.3.6-1, is OPERABLE.

1. Manual Initiation

The LCO requires two channels OPERABLE. The operator can initiate Containment ventilation isolation at any time by using either of two switches in the control room (containment isolation Phase A switches). Either switch actuates both trains. This action will cause actuation of all components in the same manner as any of the automatic actuation signals.

The LCO for Manual Initiation ensures the proper amount of redundancy is maintained in the manual actuation circuitry to ensure the operator has manual initiation capability.

Each channel consists of one CIA handswitch and the interconnecting wiring to the actuation logic cabinet.

(continued)

BASES

LCO (continued)

2. Automatic Actuation Logic and Actuation Relays

The LCO requires two channels of Automatic Actuation Logic and Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation.

Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b, SI. The applicable MODES and specified conditions for the Containment ventilation isolation portion of these Functions are different and less restrictive than those for their SI roles. If one or more of the SI Functions becomes inoperable in such a manner that only the Containment Ventilation Isolation Function is affected, the Conditions applicable to their SI Functions need not be entered. The less restrictive Actions specified for inoperability of the Containment Ventilation Isolation Functions specify sufficient compensatory measures for this case.

3. Containment Radiation

The LCO specifies two required channels of radiation monitors to ensure that the radiation monitoring instrumentation necessary to initiate Containment ventilation isolation remains OPERABLE. During CORE ALTERATIONS or movement of irradiated fuel assemblies in containment, the required channels provide input to control room alarms to ensure prompt operator action to manually close the containment purge and exhaust valves. It is also acceptable during CORE ALTERATIONS or movement of irradiated fuel to meet the requirements of this LCO by maintaining the radiation monitoring instrumentation necessary to initiate containment ventilation isolation OPERABLE, in accordance with the requirements stated for MODES 1, 2, 3, and 4 operability. The purge exhaust radiation detectors (RE-2565A, B&C) are treated as one channel which is considered OPERABLE if the particulate (RE-2565A) and iodine (RE-2565B) monitors are OPERABLE or the noble gas monitor (RE-2565C) is OPERABLE. In addition, two individual channels of containment area low range gamma monitors (RE-0002 & RE-0003) are provided. The two required radiation monitoring channels may be made up of any combination of the above described channels.

(continued)

BASES

LCO

3. Containment Radiation (continued)

For sampling systems, channel OPERABILITY involves more than OPERABILITY of the channel electronics. OPERABILITY may also require correct valve lineups, sample pump operation, and filter motor operation, as well as detector OPERABILITY, if these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.

(continued)

BASES

LCO (continued)

4. Safety Injection

Refer to LCO 3.3.2, Function 1, for all initiating Functions and requirements. The safety injection initiation function is applicable in MODES 1, 2, 3, and 4 only.

APPLICABILITY

The Manual Initiation, Automatic Actuation Logic and Actuation Relays, Containment Radiation, and Safety Injection Functions are required OPERABLE in MODES 1, 2, 3, and 4. Under these conditions, the potential exists for an accident that could release fission product radioactivity into containment. Therefore, the Containment ventilation isolation instrumentation must be OPERABLE in these MODES.

During CORE ALTERATIONS or movement of irradiated fuel assemblies in containment, the air locks may be open provided they are isolable per LCO 3.9.4. Since the air locks can only be closed manually, it is assumed that containment ventilation isolation is accomplished by manually closing the purge and exhaust ventilation valves. Therefore, only OPERABLE radiation monitors are required to alert the operators of the need for containment ventilation isolation.

While in MODES 5 and 6 without fuel handling in progress, the containment ventilation isolation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of Reference 1.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.6-1. The Completion Time(s) of

(continued)

BASES

ACTIONS (continued)

the inoperable channel(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

A.1

Condition A applies to the failure of one required containment ventilation isolation radiation monitor channel. The failed channel must be restored to OPERABLE status. Four hours are allowed to restore the affected channel based on the low likelihood of events occurring during this interval, and recognition that one or more of the remaining channels will respond to most events.

B.1

Condition B applies to all Containment Ventilation Isolation Functions and addresses the train orientation of the Solid State Protection System (SSPS) and the master and slave relays for these Functions. It also addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1.

If a manual or automatic actuation channel is inoperable, no radiation monitoring channels operable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is met for each valve made inoperable by failure of isolation instrumentation.

A Note is added stating that Condition B is only applicable in MODE 1, 2, 3, or 4.

C.1 and C.2

Condition C addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

Required Action A.1. If no radiation monitoring channels are operable or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action to place and maintain containment purge supply and exhaust isolation valves in their closed position is met or the applicable Conditions of LCO 3.9.4, "Containment Penetrations," are met for each penetration not in the required status. The Completion Time for these Required Actions is Immediately.

A Note states that Condition C is applicable during CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.6-1 determines which SRs apply to which Containment Ventilation Isolation Functions.

SR 3.3.6.1

Performance of the CHANNEL CHECK once every 12 hours 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 two instrument channels could be an indication of excessive instrument drift in one of the channels or of 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 unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.4

A COT is performed every 92 days on each required channel to ensure the entire channel will perform the intended Function. The Frequency is based on the staff recommendation for increasing the availability of radiation monitors according to NUREG-1366 (Ref. 2). For MODES 1, 2, 3, and 4, this test verifies the capability of the instrumentation to provide the containment purge and exhaust system isolation. During CORE ALTERATIONS and movement of irradiated fuel in containment, this test verifies the capability of the required channels to generate the signals required for input to the control room alarm. The setpoint shall be left consistent with the current unit specific calibration procedure tolerance.

SR 3.3.6.5

SR 3.3.6.5 is the performance of a SLAVE RELAY TEST. The SLAVE RELAY TEST is the energizing of the slave relays. Contact operation is verified in one of two ways. Actuation equipment that may be operated in the design mitigation mode is either allowed to function or is placed in a condition where the relay contact operation can be verified without operation of the equipment. Actuation equipment that may not be operated in the design mitigation mode is prevented from operation by the SLAVE RELAY TEST circuit. For this latter case, contact operation is verified by a continuity check of the circuit containing the slave relay. This test is performed every 92 days. The Frequency is acceptable based on instrument reliability and industry operating experience.

SR 3.3.6.6

SR 3.3.6.6 is the performance of a TADOT. This test is a check of the Manual Actuation Functions and is performed every 18 months. Each Manual Actuation Function is tested up to, and including, the master relay coils. In some instances, the test includes actuation of the end device (i.e., pump starts, valve cycles, etc.).

The test also includes trip devices that provide actuation signals directly to the SSPS, bypassing the analog process control equipment. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The

(continued)

B 3.3 INSTRUMENTATION

B 3.3.8 High Flux at Shutdown Alarm (HFASA)

BASES

BACKGROUND

The primary purpose of the HFASA is to warn the operator of an unplanned boron dilution event in sufficient time (15 minutes prior to loss of shutdown margin) to allow manual action to terminate the event. The HFASA is used for this purpose in MODES 3 and 4, and MODE 5 with the loops filled.

The HFASA consists of two channels of alarms, with each channel receiving input from one source range channel. An alarm setpoint of ≤2.3 times background provides at least 15 minutes from the time the HFASA occurs to the total loss of shutdown margin due to an unplanned dilution event. This meets the Standard Review Plan criteria for mitigating the consequences of an unplanned dilution event by relying on operator action.

APPLICABLE SAFETY ANALYSES

The analysis presented in Reference 1 identifies credible boron dilution initiators. Time intervals from the HFASA until loss of shutdown margin were calculated. The results demonstrate that sufficient time for operator response is available to terminate an inadvertent dilution event taking credit for one HFASA with a setpoint of ≤2.3 times background.

The HFASA satisfied Criterion 3 of the NRC Policy Statement.

LCO

The LCO requires two channels of HFASA to be OPERABLE with input from two source range channels to provide protection against single failure.

APPLICABILITY

The HFASA must be OPERABLE in MODES 3, 4, and 5.

The Applicability is modified by a Note which allows the HFASA to be blocked in MODE 3 during reactor startup so that spurious alarms are not generated.

(continued)

BASES

APPLICABILITY (continued)

In MODES 1 and 2, operators are alerted to an unplanned dilution event by a reactor trip on overtemperature delta-T or power range neutron flux high, low setpoint, respectively. As a protective measure in addition to HFASA, in MODE 5 with the loops not filled, unplanned dilution events are precluded by requiring the unborated water source (reactor makeup water storage tank (RMWST)) to be isolated.

ACTIONS

A.1

With one channel of HFASA inoperable, Required Action A.1 requires the inoperable channel to be restored within 48 hours. In this condition, one channel of HFASA remains available to provide protection. The 48 hour Completion Time is consistent with that required for an inoperable source range channel. Required Action A.1 is modified by a Note providing an exception to LCO 3.0.4. When Condition A (and Required Action A.1) are applicable, the Note permits MODE changes provided that Required Action B.1 and B.2 are met. LCO 3.0.4 allows MODE changes when the associated ACTIONS to be entered provide for continued operation for an unlimited period of time, or to comply with ACTIONS, or to facilitate a shutdown of the unit. The associated ACTIONS of LCO 3.3.8 provide for continued operation for an unlimited period of time. Therefore, with one channel of HFASA inoperable, LCO 3.0.4 would permit entry into the Applicability of LCO 3.3.8 and MODE changes within the 48 hour Completion Time allowed by Required Action A.1, before Condition B and Required Actions B.1 and B.2 would become applicable. In particular, when transitioning down through MODES 3, 4, and 5, the shutdown margin requirements become more restrictive to compensate for a postulated boron dilution event. Required Action B.1 is a periodic verification of shutdown margin, and Required Action B.2 ensures that the unborated water source isolation valves are shut, precluding a boron dilution event. With one channel of HFASA inoperable, it is prudent to take the compensatory actions of Required Actions B.1 and B.2 if MODE changes are desired or required.

B.1 and B.2

With the Required Action A.1 and associated Completion Time not met, or with both channels of HFASA inoperable, the appropriate ACTIONS are to verify that the required SDM is present and isolate the unborated water source by performing

(continued)

BASES

ACTIONS	B.1 and B.2 (continued) SR 3.9.2.1. This places the unit in a condition that precludes an unplanned dilution event. The Completion Times of 1 hour and once per 12 hours thereafter for verifying SDM provide timely assurance that no unintended dilution occurred while the HFASA was inoperable and that SDM is maintained. The Completion Times of 4 hours and once per 14 days thereafter for verifying that the unborated source is isolated provide timely assurance that an unplanned dilution event cannot occur while the HFASA is inoperable and that this protection is maintained until the HFASA is restored.
SURVEILLANCE REQUIREMENTS	The HFASA channels are subject to a COT and a CHANNEL CALIBRATION. SR 3.3.8.1 SR 3.3.8.1 requires the performance of a COT every 92 days to ensure that each channel of the HFASA and its setpoint are OPERABLE. This test shall include verification that the HFASA setpoint is less than or equal to 2.3 times background. The frequency of 92 days is consistent with the requirements for the source range channels. This Surveillance Requirement is modified by a Note that provides a 4-hour delay in the requirement to perform this surveillance for the HFASA instrumentation upon entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without delay for the performance of the surveillance to meet the applicability requirements in MODE 3. SR 3.3.8.2 SR 3.3.8.2 requires the performance of a CHANNEL CALIBRATION every 18 months. This test verifies that each channel responds to a measured parameter within the necessary range and accuracy. It encompasses the HFASA portion of the instrument loop. The frequency is based on operating experience and consistency with the typical industry refueling cycle.
REFERENCES	1. FSAR, Subsection 15.4.6.

BASES

LCO (continued)

b. RCS pressure maintained > 100 psig since the most recent filling and venting.

The loops are not considered to be filled if these requirements are not satisfied.

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least two SGs is required to be above the highest point of the SG U-tubes.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops—MODES 1 and 2";
LCO 3.4.5, "RCS Loops—MODE 3";
LCO 3.4.6, "RCS Loops—MODE 4";
LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level" (MODE 6).

ACTIONS

A.1 and A.2

If one RHR loop is inoperable and the required SGs have secondary side water levels below the highest point of the SG U-tubes, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water levels. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no RHR loop is in operation, except during conditions permitted by Note 1, or if no loop is OPERABLE, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RHR loop to

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops—MODE 5, Loops Not Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops not filled, the primary function of the reactor coolant is the removal of decay heat generated in the fuel, and the transfer of this heat to the component cooling water via the residual heat removal (RHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

In MODE 5 with loops not filled, only RHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR pump for decay heat removal and transport and to require that two paths be available to provide redundancy for heat removal.

The specified condition of Applicability "Loops not filled" is defined as when RCS pressure is not maintained >100 psig since the most recent filling and venting. It is in this specified condition of Applicability, where the smallest active volume for the RCS can occur during midloop operation. Based on the smallest active volume considered for the boron dilution transient, it was determined that each valve used to isolate unborated water sources shall be secured closed in MODE 5 with the RCS loops not filled. At least one valve in each flowpath from the Reactor Makeup Water Storage Tank (RMWST) to the suction of each charging pump shall be closed and secured in position. The applicable valve(s) will be controlled by plant procedures, which will ensure proper valve position. This action effectively isolates the unborated water source of the chemical and volume control system (CVCS) from the RCS, thereby precluding an uncontrolled boron dilution event in MODE 5 with the RCS loops not filled. However, the maximum possible flow rate from the RMWST, through the chemical mixing tank, to the suction of the charging pumps is sufficiently small that the applicable valve(s) can be allowed open under administrative control provided the applicable shutdown margin requirements of LCO 3.1.1 are met and the high flux at shutdown alarm is

(continued)

BASES

LCO (continued)

Note 3 allows valves in the flowpath from the RMWST, through the chemical mixing tank, to the suction of the charging pumps to be open under administrative control provided the SDM requirements of LCO 3.1.1 are met and the high flux at shutdown alarm is OPERABLE. (OPERABILITY of the high flux at shutdown alarm is defined by LCO 3.3.8.) This permits the addition of chemicals to the RCS as necessary in this MODE of operation while minimizing the risk of an uncontrolled boron dilution transient.

An OPERABLE RHR loop is comprised of an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required.

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the RHR System.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops—MODES 1 and 2";
LCO 3.4.5, "RCS Loops—MODE 3";
LCO 3.4.6, "RCS Loops—MODE 4";
LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level" (MODE 6).

ACTIONS

The ACTIONS table is modified by a Note prohibiting entry into MODE 5 with the loops not filled while the LCO is not met.

A.1

If only one RHR loop is OPERABLE and in operation, redundancy for RHR is lost. Action must be initiated to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

(continued)

BASES

BACKGROUND (continued)	Two groups of pressurizer heaters can be administratively loaded onto the non-Class 1E emergency buses. The Class 1E 4160-V breakers supplying the non-Class 1E buses are automatically opened upon a safety injection signal, but they can be closed under administrative procedure.
APPLICABLE SAFETY ANALYSES	In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. Safety analyses performed for lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present. Safety analyses presented in the FSAR (Ref. 1) do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure. The maximum pressurizer water level limit satisfies Criterion 2 of the NRC Policy Statement. Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 2), is the reason for providing an LCO.
LCO	The LCO requirement for the pressurizer to be OPERABLE with a water volume ≤ 1656 cubic feet, which is equivalent to 92% (LI-0459A, LI-0460A, LI-0461A), ensures that a steam bubble exists. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions. The LCO requires two groups of OPERABLE pressurizer heaters, each with a capacity ≥ 150 kW, capable of being powered from an emergency power supply. This means that the two required groups of pressurizer heaters must be capable of being powered from a Class 1E 4160-V power supply. This is accomplished by administratively loading the two required

(continued)

BASES

LCO (continued)

groups of pressurizer heaters onto the non-Class 1E emergency buses. These non-Class 1E emergency buses are in turn fed from the Class 1E 4160-V buses which can in turn be supplied from the emergency diesel generators or offsite power sources. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops.

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

In MODES 1, 2, and 3, there is the need to maintain the availability of pressurizer heaters, capable of being powered from an emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal (RHR) System is in service, and therefore, the LCO is not applicable.

ACTIONS

A.1 and A.2

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions. Normally the plant will trip in this event since the upper limit of this LCO is the same as the Pressurizer Water Level—High Trip.

If the pressurizer water level is not within the limit, action must be taken to restore the plant to operation within the bounds of the safety analyses. To achieve this status, the unit must be brought to MODE 3, with the reactor trip breakers open, within 6 hours and to MODE 4 within 12 hours. This takes the unit out of the applicable MODES

(continued)

BASES

APPLICABILITY (continued)

OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

The Applicability is modified by a Note stating that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

ACTIONS

Two Notes modify the ACTIONS table. Note 1 prohibits entry into MODE 6 with the vessel head on from MODE 6 and MODE 5 from MODE 6 with the vessel head on. Entry into MODE 4 from MODE 5 is already prohibited by LCO 3.0.4. Note 2 permits entry into MODE 4 from MODE 3 with a PORV that is inoperable for the purpose of cold overpressure protection provided that RCS temperature is maintained above 275°F, and, within 36 hours, either: the PORV is restored to OPERABLE status; or, an RHR suction relief valve is placed in service so that the requirements of LCO 3.4.12 are met. Otherwise, the reactor vessel must be depressurized and vented in accordance with Required Action F.1. With only one PORV OPERABLE, the COPS remains capable of mitigating a design basis cold overpressurization event. However, the system cannot withstand a single failure of the remaining PORV. The current COPS enable temperature is established very conservatively at 350°F. However, the application of ASME Code Case N-514 would allow the enable temperature to be lowered to less than 275°F. Therefore, when entering this LCO from MODE 3 with one required PORV inoperable, maintaining RCS temperature above 275°F minimizes actual exposure to a cold overpressure event. Furthermore, requiring action within 36 hours minimizes the exposure to a single failure while allowing sufficient time to either restore the inoperable PORV or to place RHR in service. Note 2 is only applicable to the condition of entering MODE 4 from MODE 3 with one required PORV inoperable for the purpose of cold overpressure protection. If operating in MODE 4 and a failure of a required RCS relief valve occurs, Condition D applies.

(continued)

BASES

ACTIONS (continued)

A.1

With one or more safety injection pumps capable of injecting into the RCS, RCS overpressurization is possible.

Rendering the safety injection pumps incapable of injecting into the RCS within 4 hours to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

B.1, C.1, and C.2

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required Action C.1 and Required Action C.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > 350°F, an accumulator pressure of 678 psig cannot exceed the COPS limits if the accumulators are fully injected. Depressurizing the accumulators below the COPS limit from the PTLR also gives this protection.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and that the likelihood that an event requiring COPS during this time is small.

D.1

In MODE 4, with one required RCS relief valve inoperable, the RCS relief valve must be restored to OPERABLE status within a Completion Time of 7 days. Two RCS relief valves in any combination of the PORVS and the RHR suction relief valves are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The Completion Time considers the facts that only one of the RCS relief valves is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.

(continued)

BASES

ACTIONS (continued)

E.1

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. 7). Thus, with one of the two RCS relief valves inoperable in MODE 5 or in MODE 6 with the head on, the Completion Time to restore two valves to OPERABLE status is 24 hours.

The Completion Time represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE RCS relief valve to protect against overpressure events.

F.1

The RCS must be depressurized and a vent must be established within 12 hours when:

a. Both required RCS relief valves are inoperable; or

b. A Required Action and associated Completion Time of Condition A, C, D, or E is not met; or

c. The COPS is inoperable for any reason other than Condition A, B, C, D, or E.

The vent must be sized ≥2.14 square inches (based on an equivalent length of 10 feet of pipe) to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1 and SR 3.4.12.2

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, both safety injection pumps are verified incapable of injecting into the RCS, and the accumulator discharge isolation valves are verified closed and locked out.

The safety injection pumps are rendered incapable of injecting into the RCS through at least two independent means such that a single failure or single action will not result in an injection into the RCS.

The Frequency of within 4 hours after initial entry into MODE 4 from MODE 3 and prior to RCS cold leg temperature decreasing below 325°F (for the safety injection pumps) and 12 hours thereafter (for the safety injection pumps and accumulators) is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

SR 3.4.12.3

Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction isolation valves are open and by testing it in accordance with the Inservice Testing Program. This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO. For Train A, the RHR suction relief valve is PSV-8708A and the suction isolation valves are HV-8701A and B. For Train B, the RHR suction relief valve is PSV-8708B and the suction isolation valves are HV-8702A and B.

The RHR suction valves are verified to be opened every 12 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction isolation valves remain open.

The ASME Code, Section XI (Ref. 8), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.

SR 3.4.12.4

The RCS vent of ≥ 2.14 square inches (based on an equivalent length of 10 feet of pipe) is proven OPERABLE by verifying its open condition either:

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.4 (continued)

a. Once every 12 hours for a valve that cannot be locked.

b. Once every 31 days for a valve that is locked, sealed, or secured in position. A removed pressurizer safety valve fits this category.

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12b.

SR 3.4.12.5

The PORV block valve must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. This Surveillance is performed if the PORV satisfies the LCO.

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.

SR 3.4.12.6

Performance of a COT is required within 12 hours after decreasing RCS temperature to ≤ 350°F and every 31 days on each required PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the PTLR allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is not required.

A Note has been added indicating that this SR is required to be performed 12 hours after decreasing RCS cold leg temperature to ≤ 350°F. The 12 hours considers the unlikelihood of a low temperature overpressure event during this time.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.7

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 18 months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. ASME, Boiler and Pressure Vessel Code, Section III.
4. FSAR, Chapter 15
5. 10 CFR 50, Section 50.46.
6. 10 CFR 50, Appendix K.
7. Generic Letter 90-06.
8. ASME, Boiler and Pressure Vessel Code, Section XI.
9. Westinghouse Letter GP-13419, RHR Open Permissive Setpoint.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator every 31 days since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage. Sampling the affected accumulator within 6 hours after a 1% volume increase (7% of indicated level) will identify whether inleakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the refueling water storage tank (RWST), because the water contained in the RWST is within the accumulator boron concentration requirements. This is consistent with the recommendation of NUREG-1366 (Ref. 5).

SR 3.5.1.5

Verification every 31 days that power is removed from each accumulator isolation valve operator when the pressurizer pressure is > 1000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.

This SR allows power to be supplied to the motor operated isolation valves when pressurizer pressure is ≤ 1000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns.

(continued)

BASES (continued)

REFERENCES

1. Deleted.
2. FSAR, Chapter 6.
3. 10 CFR 50.46.
4. FSAR, Chapter 15.
5. NUREG-1366, February 1990.

BASES

APPLICABILITY (continued)

the containment air locks are based on a fuel handling accident inside containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.4, "Containment Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed to repair. If the inner door is the one that is inoperable, however, then a short time exists when the containment boundary is not intact (during access through the outer door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock.

In the event the air lock leakage results in exceeding the overall containment leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment."

A.1, A.2, and A.3

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

(continued)

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

Note that for the purpose of Required Actions A.1, A.2, and A.3, the bulkhead associated with an air lock door is considered to be part of the door. For example, an air lock door may be declared inoperable if the associated door shaft seal(s) are replaced or the equalizing valve becomes inoperable, etc. It is appropriate to treat the associated bulkhead as part of the door because a leak path through the bulkhead is no different than a leak path past the door seals. The remaining OPERABLE door/bulkhead provides the necessary barrier between the containment atmosphere and the environs.

In addition, the affected air lock penetration must be isolated by locking closed the OPERABLE air lock door within

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

offsite radiological doses, and a most limiting single failure. The loss of offsite power assumption results in the ARVs being relied on to reduce RCS temperature to recover from an SGTR and also to reduce RCS temperature and pressure to RHR entry conditions. In addition, the SGTR analysis considers SG overfill. SG overfill during an SGTR event is a concern due to the potential liquid release via the ARV or Main Steam Safety Valves (MSSVs) to the atmosphere that must be assumed and the resulting increase in the offsite radiological dose. The limiting single failure with respect to SG overfill is the failure of one ARV on an intact SG to open when required for cooldown of the RCS. The analysis assumes three ARVs are OPERABLE at the start of the event. One of the ARVs is on the ruptured SG, another ARV is assumed to fail to open, and the remaining ARV is used to perform the RCS cooldown. However, there is also a scenario where the limiting single failure is the loss of control power for the two remaining ARVs. In this case, the ARVs cannot be controlled from the control room to initiate cooldown. The ARVs are equipped with local handpumps that can be used to manually open them. Given a tube rupture on one of the steam generators with an operable ARV, and the limiting single failure being a loss of control power to the remaining operable ARVs, only one ARV must be capable of being manually actuated using its handpump. If the ARV on the ruptured generator also has one of the functional handpumps, then only one of the remaining ARVs need have a functional handpump in order to meet the safety analysis. Because no additional failures need to be postulated in addition to the loss of control power, only two functional ARV handpumps are required. The analysis shows that cooldown using a single ARV does not result in SG overfill. The failure of one ARV to open does not represent the most limiting single failure with respect to offsite radiological doses. The failure open of the ARV on the ruptured SG is the limiting failure for offsite radiological doses. This failure results in an uncontrolled depressurization of the ruptured SG until the local manual isolation valve for that ARV is closed. This failure maximizes the activity release from the ruptured SG to the atmosphere.

The recovery from the SGTR requires a rapid cooldown to establish adequate subcooling as a necessary step to allow depressurization of the RCS to terminate the primary to secondary break flow in the ruptured steam generator. The time required to terminate the primary to secondary break flow in the SGTR event is more critical than the time required to cool the RCS down to RHR conditions for this

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

event and other accident analyses. After primary to secondary break flow termination, it is assumed that one ARV on an intact SG is used to cool the RCS down to 350°F, at the maximum allowable cooldown rate of 100°F/hour.

The offsite radiological dose analyses show that the failure open of the ARV on the ruptured SG represents the limiting single failure. The resulting offsite radiological doses at the exclusion area boundary, low population zone, and control room are well within the allowable guidelines as specified by Standard Review Plan 15.6.3 and 10 CFR 100. A detailed description of the SGTR analyses can be found in WCAP-11731 and associated supplements (Ref. 3).

The ARVs are equipped with manual block valves in the event an ARV spuriously fails open or fails to close during use.

The ARVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

Three ARV lines are required to be OPERABLE. One ARV line is required from each of three steam generators to ensure that at least one ARV line is available to conduct a unit cooldown following an SGTR, in which one steam generator becomes unavailable, accompanied by a single, active failure of a second ARV line on an unaffected steam generator. A block valve for each required ARV must be OPERABLE to isolate a failed open ARV line.

Failure to meet the LCO can result in the inability to cool the unit to RHR entry conditions following an SGTR event in which the condenser is unavailable for use with the Steam Dump System.

An ARV is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing on demand. Additionally, it is required that at least two of the three OPERABLE ARVs maintain the capability for local manual actuation via their associated handpumps.

APPLICABILITY

In MODES 1, 2, and 3, the ARVs are required to be OPERABLE.

In MODE 4, the pressure and temperature limitations are such that the probability of an SGTR event requiring ARV operation is low. In addition, the RHR system is available

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a controlled cooldown of the RCS, the ARVs must be able to be opened either remotely or locally and throttled through their full range. This SR ensures that the ARVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an ARV during a unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. The Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section 10.3.
2. FSAR, Subsection 15.6.3.
3. WCAP-11731, LOFTTR2 Analysis for a Steam Generator Tube Rupture Event for the Vogtle Electric Generating Plant Units 1 and 2, January 1988, and Westinghouse letter GP-16886, J. L. Tain to J.B. Beasley, Jr., SGTR Analysis With Revised Operator Action Times and SECL 98-124, Revision 0, dated December 4, 1998.

BASES

LCO (continued)

established in Reference 4 and exceeds the volume required by the accident analysis.

The OPERABILITY of the CST is determined by maintaining the tank level at or above the minimum required level. Either CST V4001 or CST V4002 may be used to satisfy the LCO requirement.

APPLICABILITY

In MODES 1, 2, and 3, the CST is required to be OPERABLE.

Due to the reduced heat removal requirements and short period of time in MODE 4 and the availability of RHR in MODE 4, the LCO does not require a CST to be OPERABLE in this MODE.

In MODE 5 or 6, the CST is not required because the AFW System is not required.

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B 3.7 PLANT SYSTEMS

B 3.7.17 Fuel Storage Pool Boron Concentration

BASES

BACKGROUND

Fuel assemblies are stored in high density racks. The Unit 1 spent fuel storage racks contain storage locations for 1476 fuel assemblies, and the Unit 2 spent fuel storage racks contain storage locations for 2098 fuel assemblies. The Unit 1 racks use boral as a neutron absorber in a flux trap design. The Unit 2 racks contain Boraflex, however, no credit is taken for Boraflex. Westinghouse 17x17 fuel assemblies with initial enrichments of up to and including 5.0 weight percent U-235 can be stored in any location in the Unit 1 or Unit 2 fuel storage pool provided the fuel burnup-enrichment combinations are within the limits that are specified in Figures 3.7.18-1 (Unit 1) or 3.7.18-2 (Unit 2) of the Technical Specifications. Fuel assemblies that do not meet the burnup-enrichment combination of Figures 3.7.18-1 or 3.7.18-2 may be stored in the storage pools of Units 1 or 2 in accordance with checkerboard storage configurations described in Figures 4.3.1-2 through 4.3.1-9. The acceptable fuel assembly storage configurations are based on the Westinghouse Spent Fuel Rack Criticality Methodology, described in WCAP-14416-NP-A, Rev. 1, (Reference 4). This methodology includes computer code benchmarking, spent fuel rack criticality calculations methodology, reactivity equivalencing methodology, accident methodology, and soluble boron credit methodology.

The Westinghouse Spent Fuel Rack Criticality Methodology ensures that the multiplication factor, K_{eff}, of the fuel and spent fuel storage racks is less than or equal to 0.95 as recommended by ANSI 57.2-1983 (Reference 3) and NRC guidance (References 1, 2 and 6). The codes, methods, and techniques contained in the methodology are used to satisfy this criterion on K_{eff}.

The methodology of the NITAWL-II, XSDRNPM-S, and KENO-Va codes is used to establish the bias and bias uncertainty. PHOENIX-P, a nuclear design code used primarily for core reactor physics calculations is used to simulate spent fuel storage rack geometries.

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BASES

BACKGROUND (continued)

Reference 4 describes how credit for fuel storage pool soluble boron is used under normal storage configuration conditions. The storage configuration is defined using K_{eff} calculations to ensure that the K_{eff} will be less than 1.0 with no soluble boron under normal storage conditions including tolerances and uncertainties. Soluble boron credit is then used to maintain K_{eff} less than or equal to 0.95. The Unit 1 pool requires 600 ppm and the Unit 2 pool requires 500 ppm to maintain K_{eff} less than or equal to 0.95 for all allowed combinations of storage configurations, enrichments, and burnups. The analyses assumed 19.9% of the boron atoms have atomic weight 10 (B-10). The effects of B-10 depletion on the boron concentration for maintaining $K_{eff} \leq 0.95$ are negligible. The treatment of reactivity equivalencing uncertainties, as well as the calculation of postulated accidents crediting soluble boron is described in WCAP-14416-NP-A, Rev. 1.

This methodology was used to evaluate the storage of fuel with initial enrichments up to and including 5.0 weight percent U-235 in the Vogtle fuel storage pools. The resulting enrichment, and burnup limits for the Unit 1 and Unit 2 pools, respectively, are shown in Figures 3.7.18-1 and 3.7.18-2. Checkerboard storage configurations are defined to allow storage of fuel that is not within the acceptable burnup domain of Figures 3.7.18-1 and 3.7.18-2. These storage requirements are shown in Figures 4.3.1-2 through 4.3.1-9. A boron concentration of 2000 ppm assures that no credible dilution event will result in a K_{eff} of > 0.95.

APPLICABLE SAFETY ANALYSES

Most fuel storage pool accident conditions will not result in an increase in K_{eff}. Examples of such accidents are the drop of a fuel assembly on top of a rack, and the drop of a fuel assembly between rack modules, or between rack modules and the pool wall.

From a criticality standpoint, a dropped assembly accident occurs when a fuel assembly in its most reactive condition is dropped onto the storage racks. The rack structure from a criticality standpoint is not excessively deformed. Previous accident analysis with unborated water showed that the dropped assembly which comes to rest horizontally on top of the rack has sufficient water separating it from the

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BASES

APPLICABLE SAFETY ANALYSES (continued)

active fuel height of stored assemblies to preclude neutronic interaction. For the borated water condition, the interaction is even less since the water contains boron, an additional thermal neutron absorber.

However, three accidents can be postulated for each storage configuration which could increase reactivity beyond the analyzed condition. The first postulated accident would be a change in pool temperature to outside the range of temperatures assumed in the criticality analyses (50°F to 185°F). The second accident would be dropping a fuel assembly into an already loaded cell. The third would be the misloading of a fuel assembly into a cell for which the restrictions on location, enrichment, or burnup are not satisfied.

An increase in the temperature of the water passing through the stored fuel assemblies causes a decrease in water density which results in an addition of negative reactivity for flux trap design racks such as the Unit 1 racks. However, since Boraflex is not considered to be present for the Unit 2 racks and the fuel storage pool water has a high concentration of boron, a density decrease causes a positive reactivity addition. The reactivity effects of a temperature range from 32° F to 240° F were evaluated. The increase in reactivity due to the increase in temperature is bounded by the misload accident, for the Unit 2 racks. The increase in reactivity due to the decrease in temperature below 50° F is bounded by the misplacement of a fuel assembly between the rack and pool walls for the Unit 1 racks.

For the accident of dropping a fuel assembly into an already loaded cell, the upward axial leakage of that cell will be reduced, however, the overall effect on the rack reactivity will be insignificant. This is because the total axial leakage in both the upward and downward directions for the entire fuel array is worth about 0.003 Δk. Thus, minimizing the upward-only leakage of just a single cell will not cause any significant increase in reactivity. Furthermore, the neutronic coupling between the dropped assembly and the already loaded assembly will be low due to several inches of assembly nozzle structure which would separate the active fuel regions. Therefore, this accident would be bounded by the misload accident.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The fuel assembly misloading accident involves placement of a fuel assembly in a location for which it does not meet the requirements for enrichment or burnup, including the placement of an assembly in a location that is required to be left empty. The result of the misloading is to add positive reactivity, increasing K_{eff} toward 0.95. A fourth accident was evaluated for the Unit 1 fuel storage racks containing boral. The fourth accident was the misplacement of a fuel assembly between the rack and pool wall. This was the limiting accident for the Unit 1 racks. The

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES (continued)	maximum required additional boron to compensate for this event is 1250 ppm for Unit 2, and 800 ppm for Unit 1 which is well below the limit of 2000 ppm. The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of the NRC Policy Statment.
LCO	The fuel storage pool boron concentration is required to be ≥ 2000 ppm. The specified concentration of dissolved boron in the fuel storage pool preserves the assumptions used in the analyses of the potential criticality accident scenarios as described in reference 5. The amount of soluble boron required to offset each of the above postulated accidents was evaluated for all of the proposed storage configurations. That evaluation established the amount of soluble boron necessary to ensure that K_{eff} will be maintained less than or equal to 0.95 should pool temperature exceed the assumed range or a fuel assembly misload occur. The amount of soluble boron necessary to mitigate these events was determined to be 1250 ppm for Unit 2 and 800 ppm for Unit 1. The specified minimum boron concentration of 2000 ppm assures that the concentration will remain above these values. In addition, the boron concentration is consistent with the boron dilution evaluation that demonstrated that any credible dilution event could be terminated prior to reaching the boron concentration for a K_{eff} of > 0.95. These values are 600 ppm for Unit 1 and 500 ppm for Unit 2.
APPLICABILITY	This LCO applies whenever fuel assemblies are stored in the spent fuel storage pool.
ACTIONS	<u>A.1, A.2.1, and A.2.2</u> The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply. When the concentration of boron in the fuel storage pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most

(continued)

BASES

ACTIONS (continued)

efficiently achieved by immediately suspending the movement of fuel assemblies. Immediate action to restore the concentration of boron is also required simultaneously with suspending movement of fuel assemblies. This does not preclude movement of a fuel assembly to a safe position

If the LCO is not met while moving irradiated fuel assemblies in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.17.1

This SR verifies that the concentration of boron in the fuel storage pool is within the required limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over such a short period of time. The gate between the Unit 1 and Unit 2 fuel storage pool is normally open. When the gate is open the pools are considered to be connected for the purpose of conducting the surveillance.

REFERENCES

1. USNRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition. NUREG-0800, June 1987.
2. USNRC Spent Fuel Storage Facility Design Bases (for Comment) Proposed Revision 2, 1981. Regulatory Guide 1.13.
3. ANS, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," ANSI/ANS-57.2-1983.
4. WCAP-14416 NP-A, Rev. 1, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," November 1996.
5. Vogtle FSAR, Section 4.3.2.
6. Nuclear Regulatory Commission, Letter to All Power Reactor Licensees from B. K. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978.

B 3.7 PLANT SYSTEMS

B 3.7.18 Fuel Assembly Storage in the Fuel Storage Pool

BASES

BACKGROUND

The Unit 1 spent fuel storage racks contain storage locations for 1476 fuel assemblies, and the Unit 2 spent fuel storage racks contain storage locations for 2098 fuel assemblies.

Westinghouse 17X17 fuel assemblies with an enrichment of up to and including 5.0 weight percent U-235 can be stored in the acceptable storage configurations that are specified in Figures 3.7.18-1 (Unit 1), 3.7.18-2 (Unit 2), and 4.3.1-2 through 4.3.1-9. The acceptable fuel assembly storage locations are based on the Westinghouse Spent Fuel Rack Criticality Methodology, described in WCAP-14416-NP-A, Rev. 1 (reference 1). Additional background discussion can be found in B 3.7.17.

Westinghouse 17x17 fuel assemblies with nominal enrichments no greater than 3.50 w/o^{235}U may be stored in all storage cell locations of the Unit 1 pool. Fuel assemblies with initial nominal enrichment greater than 3.50 w/o^{235}U must satisty a minimum burnup requirement as shown in Figure 3.7.18-1. Fuel assemblies having a K_∞ of 1.431 at cold reactor core conditions may also be stored in all cells of the Unit 1 fuel storage racks.

Westinghouse 17x17 fuel assemblies with nominal enrichments no greater than 5.0 w/o^{235}U may be stored in a 3-out-of-4 checkerboard arrangement with empty cells in the Unit 1 pool. There are no minimum burnup requirements for this configuration.

Westinghouse 17x17 fuel assemblies with nominal enrichments no greater that 5.0 w/o^{235}U may be stored in a 2-out-of-4 checkerboard arrangement with empty cells in the Unit 2 pool. There are no minimum burnup requirements for this configuration.

(continued)

BASES

BACKGROUND (continued)

Westinghouse 17x17 fuel assemblies with nominal enrichments no greater than 1.77 w/o^{235}U may be stored in all storage cell locations of the Unit 2 pool. Fuel assemblies with initial nominal enrichment greater than 1.77 w/o^{235}U must satisfy a minimum burnup requirement as shown in Figure 3.7.18-2.

(continued)

BASES

BACKGROUND (continued)

Westinghouse 17x17 fuel assemblies with nominal enrichments no greater than 2.40 w/o^{235}U may be stored in a 3-out-of-4 checkerboard arrangement with empty cells in the Unit 2 pool. Fuel assemblies with initial nominal enrichment greater than 2.40 w/o^{235}U must satisfy a minimum burnup requirement as shown in Figure 4.3.1-2.

Westinghouse 17x17 fuel assemblies may be stored in the Unit 2 pool in a 3x3 array. The center assembly must have an initial enrichment no greater than 3.20 w/o^{235}U. Alternatively, the center of the 3x3 array may be loaded with any assembly which meets a maximum infinite multiplication factor (K_{∞}) value of 1.410 at 68°F. One method of achieving this value of K_{∞} is by the use of IFBAs. The surrounding fuel assemblies must have an initial nominal enrichment no greater than 1.48 w/o^{235}U or satisfy a minimum burnup requirement for higher initial enrichments as shown in Figure 4.3.1-3.

APPLICABLE SAFETY ANALYSIS

Most fuel storage pool accident conditions will not result in an increase in K_{eff}. Examples of such accidents are the drop of a fuel assembly on top of a rack and the drop of a fuel assembly between rack modules or between rack modules and the pool wall. However, accidents can be postulated for each storage configuration which could increase reactivity beyond the analyzed condition. A discussion of these accidents is contained in B 3.7.17.

The configuration of fuel assemblies in the fuel storage pool satisfies Criterion 2 of the NRC Policy Statement.

LCO

The restrictions on the placement of fuel assemblies within the fuel storage pool ensure the K_{eff} of the fuel storage pool will always remain < 0.95, assuming the pool to be flooded with borated water.

The combination of initial enrichment and burnup are specified in Figures 3.7.18-1 and 3.7.18-2 for all cell storage in the Unit 1 and Unit 2 pools, respectively. Other acceptable enrichment burnup and checkerboard combinations are described in Figures 4.3.1-1 through 4.3.1-9.

(continued)

BASES (continued)

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in the fuel storage pool.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the configuration of fuel assemblies stored in the fuel storage pool is not in accordance with the acceptable combination of initial enrichment, burnup, and storage configurations, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figures 3.7.18-1 (Unit 1), 3.7.18-2 (Unit 2), or Specification 4.3.1.1, (Unit 1) or 4.3.1.2 (Unit 2).

If unable to move irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If unable to move irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the action is independent of reactor operation. Therefore inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.18.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is within the acceptable burnup domain of Figures 3.7.18-1 (Unit 1) or 3.7.18-2 (Unit 2). For fuel assemblies in the unacceptable range of Figures 3.7.18-1 and 3.7.18-2, performance of this SR will also ensure compliance with Specification 4.3.1.1 (Unit 1) or 4.3.1.2 (Unit 2).

Fuel assembly movement will be in accordance with preapproved plans that are consistent with the specified fuel enrichment, burnup, and storage configurations. These plans are administratively verified prior to fuel movement. Each assembly is verified by visual inspection to be in accordance with the preapproved plan prior to storage in the fuel storage pool. Storage commences following unlatching of the fuel assembly in the fuel storage pool.

(continued)

BASES (continued)

REFERENCES	1.	WCAP-14416-NP-A, Revision 1, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," November 1996.

BASES

LCO (continued)

train. For the DGs, separation and independence are complete.

For the offsite AC sources, separation and independence are to the extent practical. A circuit may be connected to more than one ESF bus while the bus is being transferred to the other circuit.

APPLICABILITY

The AC sources and sequencers are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The AC power requirements for MODES 5 and 6 are covered in LCO 3.8.2, "AC Sources — Shutdown."

ACTIONS

A.1

To ensure a highly reliable power source remains with one offsite circuit inoperable, it is necessary to verify the OPERABILITY of the remaining required offsite circuit on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action not met. However, if a second required circuit fails SR 3.8.1.1, the second offsite circuit is inoperable, and Condition D, for two offsite circuits inoperable, is entered.

A.2

Required Action A.2, which only applies if the train cannot be powered from an offsite source, is intended to provide assurance that an event coincident with a single failure of the associated DG will not result in a complete loss of safety function of critical redundant required features.

(continued)

BASES

ACTIONS

A.2 (continued)

These features are powered from the redundant AC electrical power train. This includes motor driven auxiliary feedwater pumps. Single train systems, such as turbine driven auxiliary feedwater pumps, may not be included.

The Completion Time for Required Action A.2 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. The train has no offsite power supplying its loads; and

b. A required feature on the other train is inoperable.

If at any time during the existence of Condition A (one offsite circuit inoperable) a redundant required feature subsequently becomes inoperable, this Completion Time begins to be tracked.

Discovering no offsite power to one train of the onsite Class 1E Electrical Power Distribution System coincident with one or more inoperable required support or supported features, or both, that are associated with the other train that has offsite power, results in starting the Completion Times for the Required Action. Twenty-four hours is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

The remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to Train A and Train B of the onsite Class 1E Distribution System. The 24 hour Completion Time takes into account the component OPERABILITY of the redundant counterpart to the inoperable required feature. Additionally, the 24 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

(continued)

BASES

ACTIONS (continued)

A.3

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition A for a period that should not exceed 72 hours. With one required offsite circuit inoperable, the reliability of the offsite system is degraded, and the potential for a loss of offsite power is increased, with attendant potential for a challenge to the unit safety systems. In this Condition, however, the remaining OPERABLE offsite circuit and DGs are adequate to supply electrical power to the onsite Class 1E Distribution System.

The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

The second Completion Time for Required Action A.3 establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition A is entered while, for instance, a DG is inoperable and that DG is subsequently returned OPERABLE, the LCO may already have been not met for up to 11 days. This could lead to a total of 14 days, since initial failure to meet the LCO, to restore the offsite circuit. At this time, a DG could again become inoperable, the circuit restored OPERABLE, and an additional 72 hours, or 14 days depending on SAT availability, allowed prior to complete restoration of the LCO. The 14 day Completion Time provides a limit on the time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The "AND" connector between the 72 hour and 14 day Completion Times means that both Completion Times apply simultaneously, and the more restrictive Completion Time must be met.

Tracking the 14 day Completion Time is a requirement for beginning the Completion Time "clock" that is in addition to the normal Completion Time requirements. With respect to the 14 day Completion Time, the "time zero" is specified as

(continued)

BASES

ACTIONS

A.3 (continued)

commencing at the time LCO 3.8.1 was initially not met, instead of at the time Condition A was entered. This results in the requirement when in this Condition to track the time elapsed from both the Condition A "time zero" and the "time zero" when LCO 3.8.1 was initially not met.

B.1

To ensure a highly reliable power source remains with an inoperable DG, it is necessary to verify the availability of the offsite circuits on a more frequent basis. Since the Required Action only specifies "perform," a failure of SR 3.8.1.1 acceptance criteria does not result in a Required Action being not met. However, if a circuit fails to pass SR 3.8.1.1, it is inoperable. Upon offsite circuit inoperability, additional Conditions and Required Actions must then be entered.

B.2

The 13.8/4.16 kV Standby Auxiliary Transformer (SAT) is a qualified offsite circuit that may be connected to the onsite Class 1E distribution system independently of the RATs and may be utilized to meet the LCO 3.8.1 requirements for an offsite circuit. Its availability permits an extension of the allowable out-of-service time for a DG to 14 days from the discovery of failure to meet LCO 3.8.1. The SAT is available when it is:

- Operable in accordance with plant procedures;
- Not already being applied to any of the four 4.16 kV ESF buses for Units 1 and 2 in accordance with Specification 3.8.1 as either an offsite source or to meet the requirements of an LCO 3.8.1 Condition; and,
- Not providing power to the other unit when that unit is in MODE 5 or 6 or defueled.

(continued)

BASES

ACTIONS

B.2 (continued)

Furthermore, the SAT can be applied to only one of the four 4.16 kV ESF buses at any given time for Units 1 and 2 to meet the requirements of an LCO 3.8.1 Condition.

When one or more of these criteria are not satisfied, the SAT is not available. These criteria are structured to ensure that the SAT is available as an alternate offsite source to support the extended DG Completion Time of 14 days. Therefore, when a DG is inoperable, it is necessary to verify the availability of the SAT within one hour and once per 12 hours thereafter. If Required Action B.2 is not met or the status of the SAT changes after Required Action B.2 is initially met, Condition C must be entered concurrently.

B.3

Required Action B.3 is intended to provide assurance that a loss of offsite power, during the period that a DG is inoperable, does not result in a complete loss of safety function of critical systems. These features are designed with redundant safety related trains. This includes motor driven auxiliary feedwater pumps. Single train systems, such as turbine driven auxiliary feedwater pumps, are not included. Redundant required feature failures consist of inoperable features associated with a train, redundant to the train that has an inoperable DG.

The Completion Time for Required Action B.3 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." In this Required Action, the Completion Time only begins on discovery that both:

a. An inoperable DG exists; and

b. A required feature on the other train (Train A or Train B) is inoperable.

(continued)

BASES

ACTIONS

B.3 (continued)

If at any time during the existence of this Condition (one DG inoperable) a required feature subsequently becomes inoperable, this Completion Time would begin to be tracked.

Discovering one required DG inoperable coincident with one or more inoperable required support or supported features, or both, that are associated with the OPERABLE DG, results in starting the Completion Time for the Required Action. Four hours from the discovery of these events existing concurrently is acceptable because it minimizes risk while allowing time for restoration before subjecting the unit to transients associated with shutdown.

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

B.4.1 and B.4.2

Required Action B.3.1 provides an allowance to avoid unnecessary testing of the OPERABLE DG. If it can be determined that the cause of the inoperable DG does not exist on the OPERABLE DG, SR 3.8.1.2 does not have to be performed. If the cause of inoperability exists on the other DG, the other DG would be declared inoperable upon discovery and Condition F of LCO 3.8.1 would be entered. Once the failure is repaired, the common cause failure no longer exists, and Required Action B.4.1 is satisfied. If the cause of the initial inoperable DG cannot be confirmed not to exist on the remaining DG, performance of SR 3.8.1.2 suffices to provide assurance of continued OPERABILITY of that DG.

(continued)

BASES

ACTIONS

B.4.1 and B.4.2 (continued)

In the event the inoperable DG is restored to OPERABLE status prior to completing either B.4.1 or B.4.2, the applicable plant procedures will continue to require the evaluation of the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed while in Condition B.

According to Generic Letter 84-15 (Ref. 7), 24 hours is reasonable to confirm that the OPERABLE DG is not affected by the same problem as the inoperable DG.

B.5.1 and B.5.2

Required Action B.5.1 provides assurance that an enhanced black-start combustion turbine generator (CTG) is functional when a DG is out of service for greater than 72 hours. Required Action B.5.1 is modified by a Note that states that it is only applicable provided that the two enhanced black-start CTGs and black-start diesel generator have a combined reliability of ≥ 95% based on a minimum of 20 tests per enhanced black-start CTG and quarterly testing thereafter. This quarterly testing will subject each enhanced black-start CTG to a start and load-run test. The black-start diesel generator will also be tested quarterly, but separately from the enhanced black-start CTGs. Required Action B.5.1 may be met by starting either of the enhanced black-start CTGs and the black-start diesel generator and verifying that they achieve steady state voltage and frequency. The black-start diesel generator may be started separately.

If a DG is to be removed from service voluntarily for greater than 72 hours, it may be advantageous to test an enhanced black-start CTG prior to taking the DG out of service. In such cases where advanced notice of removing a DG from service is available, Required Action B.5.1 may be performed up to 72 hours prior to entry into Condition B. In other cases, Required Action B.5.1 must be performed within 72 hours after entry into Condition B.

(continued)

BASES

ACTIONS

B.5.1 and B.5.2 (continued)

If the combined reliability of the enhanced black-start CTGs has not been demonstrated or maintained ≥ 95%, the option of starting and running any one of the six CTGs while in Condition B is available in the form of Required Action B.5.2. In the event of preplanned maintenance that would exceed 72 hours, any one of the six CTGs must be started prior to entry into Condition B and allowed to run for the duration of Condition B. Otherwise, any one of the six CTGs must be started within 72 hours (and allowed to run) after entry into Condition B if the DG is to be out of service for more than 72 hours. Note that Required Action B.5.1 requires that one of the two enhanced black-start CTGs be started, but any one of the six CTGs could be started to satisfy Required Action B.5.2. Since a CTG is started and running while the DG is inoperable, it is not necessary that the CTG have enhanced black-start capability.

B.6

The availability of the SAT provides an additional AC source which permits operation to continue for a period not to exceed 14 days from discovery of failure to meet the LCO.

In Condition B, the remaining OPERABLE DG and offsite circuits are adequate to supply electrical power to the onsite Class 1E Distribution System. The 14 day Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

In addition, the Configuration Risk Management Program (CRMP) is used to assess changes in core damage frequency resulting from applicable plant configurations. The CRMP uses the equipment out of service risk monitor, a computer based tool that may be used to aid in the risk assessment of on-line maintenance and to evaluate the change in risk from a component failure. The equipment out of service risk monitor uses the plant probabilistic risk assessment model to evaluate the risk of removing equipment from service

(continued)

BASES

ACTIONS

B.6 (continued)

based on current plant configuration and equipment condition. The CRMP is used when a DG is intentionally taken out of service for a planned activity excluding short duration activities (e.g., performing an air roll on the EDG prior to a routine surveillance). In addition, the CRMP is used for unplanned maintenance or repairs of a DG.

Planned activities involving an extended DG AOT will be synchronized with other maintenance activities as much as possible in order to maximize equipment reliability while minimizing the time equipment is unavailable. In addition, Required Action B.3 requires that features supported by the inoperable DG be declared inoperable within 4 hours of discovery when redundant features are discovered to be inoperable. The combination of planned maintenance centered around the extended DG AOT, Required Action B.3, and use of the CRMP provides an appropriate level of assurance that risk significant activities with an unacceptable risk achievement worth will be minimized during an extended DG AOT.

The Completion Time for Required Action B.6 also establishes a limit on the maximum time allowed for any combination of required AC power sources to be inoperable during any single contiguous occurrence of failing to meet the LCO. If Condition B is entered while, for instance, an offsite circuit is inoperable, the LCO may already have been not met for up to 72 hours. If the offsite circuit is restored within the required 72 hours, this could lead to a total of 17 days, since initial failure to meet the LCO, to restore compliance with the LCO (i.e., restore the DG). However, the 14 day Completion Time provides a

(continued)

BASES

ACTIONS

B.6 (continued)

limit on time allowed in a specified condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B (and consequently Condition E) are entered concurrently.

Tracking the 14 day Completion Time is a requirement for beginning the Completion Time "clock" that is in addition to the normal Completion Time requirements. With respect to the Completion Time, the "time zero" is specified as commencing at the time LCO 3.8.1 was initially not met, instead of at the time Condition B was entered. This results in the requirement when in this Condition to track the time elapsed from both the Condition B "time zero" and the "time zero" when LCO 3.8.1 was initially not met.

C.1

If the availability of the SAT cannot be verified, or if no CTG meets the requirements of either Required Action B.5.1 or B.5.2, the DG must be restored to OPERABLE status within 72 hours. The 72 hour Completion Time begins upon entry into Condition C. However, the total time to restore an inoperable DG cannot exceed 14 days (per the Completion Time of Required Action B.6).

The Completion Time of 72 hours (in the absence of the SAT) is consistent with Regulatory Guide 1.93 (Ref.6). The 72 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and low probability of a DBA occurring this period.

D.1 and D.2

Required Action D.1, which applies when two offsite circuits are inoperable, is intended to provide assurance that an event with a coincident single failure will not result in a complete loss of redundant required safety functions. The Completion Time for this failure of redundant required features is reduced to 12 hours from that allowed for one train without offsite power (Required Action A.2). The rationale for the reduction to 12 hours is that Regulatory Guide 1.93 (Ref. 6) allows a Completion Time of 24 hours for

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

two required offsite circuits inoperable, based upon the assumption that two complete safety trains are OPERABLE. When a concurrent redundant required feature failure exists, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate. These features are powered from redundant AC safety trains. This includes motor driven auxiliary feedwater pumps. Single train features, such as turbine driven auxiliary pumps, are not included in the list.

The Completion Time for Required Action D.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

beginning the allowed outage time "clock." In this Required Action the Completion Time only begins on discovery that both:

a. All required offsite circuits are inoperable; and

b. A required feature is inoperable.

If at any time during the existence of Condition D (two offsite circuits inoperable) a required feature becomes inoperable, this Completion Time begins to be tracked.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition D for a period that should not exceed 24 hours. This level of degradation means that the offsite electrical power system does not have the capability to effect a safe shutdown and to mitigate the effects of an accident; however, the onsite AC sources have not been degraded. This level of degradation generally corresponds to a total loss of the immediately accessible offsite power sources.

Because of the normally high availability of the offsite sources, this level of degradation may appear to be more severe than other combinations of two AC sources inoperable that involve one or more DGs inoperable. However, two factors tend to decrease the severity of this level of degradation:

a. The configuration of the redundant AC electrical power system that remains available is not susceptible to a single bus or switching failure; and

b. The time required to detect and restore an unavailable offsite power source is generally much less than that required to detect and restore an unavailable onsite AC source.

With both of the required offsite circuits inoperable, sufficient onsite AC sources are available to maintain the unit in a safe shutdown condition in the event of a DBA or transient. In fact, a simultaneous loss of offsite AC sources, a LOCA, and a worst case single failure were

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

postulated as a part of the design basis in the safety analysis. Thus, the 24 hour Completion Time provides a period of time to effect restoration of one of the offsite circuits commensurate with the importance of maintaining an AC electrical power system capable of meeting its design criteria.

According to Reference 6, with the available offsite AC sources, two less than required by the LCO, operation may continue for 24 hours. If two offsite sources are restored within 24 hours, unrestricted operation may continue. If only one offsite source is restored within 24 hours, power operation continues in accordance with Condition A.

E.1 and E.2

Pursuant to LCO 3.0.6, the Distribution System ACTIONS would not be entered even if all AC sources to it were inoperable, resulting in de-energization. Therefore, the Required Actions of Condition E are modified by a Note to indicate that when Condition E is entered with no AC source to one or more trains, the Conditions and Required Actions for LCO 3.8.9, "Distribution Systems — Operating," must be immediately entered. This allows Condition E to provide requirements for the loss of one offsite circuit and one DG, without regard to whether a train is de-energized. LCO 3.8.9 provides the appropriate restrictions for a de-energized train.

According to Regulatory Guide 1.93 (Ref. 6), operation may continue in Condition E for a period that should not exceed 12 hours.

In Condition E, individual redundancy is lost in both the offsite electrical power system and the onsite AC electrical power system. Since power system redundancy is provided by two diverse sources of power, however, the reliability of the power systems in this Condition may appear higher than that in Condition D (loss of both required offsite circuits). This difference in reliability is offset by the susceptibility of this power system configuration to a

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

single bus or switching failure. The 12 hour Completion Time takes into account the capacity and capability of the remaining AC sources, a reasonable time for repairs, and the low probability of a DBA occurring during this period.

F.1

With Train A and Train B DGs inoperable, there are no remaining standby AC sources. Thus, with an assumed loss of offsite electrical power, insufficient standby AC sources are available to power the minimum required ESF functions. Since the offsite electrical power system is the only source of AC power for this level of degradation, the risk associated with continued operation for a very short time could be less than that associated with an immediate controlled shutdown (the immediate shutdown could cause grid instability, which could result in a total loss of AC power). Since any inadvertent generator trip could also result in a total loss of offsite AC power, however, the time allowed for continued operation is severely restricted. The intent here is to avoid the risk associated with an immediate controlled shutdown and to minimize the risk associated with this level of degradation.

According to Reference 6, with both DGs inoperable, operation may continue for a period that should not exceed 2 hours.

G.1

The sequencer(s) is an essential support system to both the offsite circuit and the DG associated with a given ESF bus. Furthermore, the sequencer is on the primary success path for most major AC electrically powered safety systems powered from the associated ESF bus. The sequencers are required to provide the system response to both an SI signal and a loss of or degraded ESF bus voltage signal. Therefore, loss of an ESF bus sequencer affects every major ESF system in the train. The 12 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining sequencer

(continued)

BASES

ACTIONS

G.1 (continued)

OPERABILITY. This time period also ensures that the probability of an accident (requiring sequencer OPERABILITY) occurring during periods when the sequencer is inoperable is minimal.

H.1 and H.2

If the inoperable AC electric power sources or an automatic load sequencer cannot be restored to OPERABLE status within the required Completion Time, or Required Actions B.1, B.3, B.4.1, B.4.2, or B.6 cannot be met within the required Completion Times, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

I.1

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

SURVEILLANCE REQUIREMENTS

The AC sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function, in accordance with 10 CFR 50, Appendix A, GDC 18 (Ref. 8). Periodic component tests are supplemented by extensive functional tests during refueling outages (under simulated accident conditions). The SRs for demonstrating the OPERABILITY of the DGs are in accordance with the recommendations of Regulatory Guide 1.9 (Ref. 3), Regulatory Guide 1.108 (Ref. 9), and Regulatory Guide 1.137 (Ref. 10), as addressed in the FSAR.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The RCS boron concentration satisfies Criterion 2 of the NRC Policy Statement.
LCO	The LCO requires that a minimum boron concentration be maintained in all filled portions of the RCS, the refueling canal, and the refueling cavity while in MODE 6. The boron concentration limit specified in the COLR ensures that a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.
APPLICABILITY	This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $k_{eff} \leq 0.95$. In MODES 1 and 2, LCO 3.1.4, "Rod Group Alignment Limits," LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits," ensure an adequate amount of negative reactivity is available to shut down the reactor. In MODES 3, 4, and 5, LCO 3.1.1, "SHUTDOWN MARGIN" ensures an adequate amount of negative reactivity is available to shut down the reactor.
ACTIONS	The ACTIONS table is modified by a Note prohibiting entry into MODE 6 if the RCS boron concentration specified in the COLR is not met. <u>A.1 and A.2</u> Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the filled portions of the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately. Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position or normal cooldown of the coolant volume for the purpose of system temperature control.

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B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

BASES

BACKGROUND

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors (NI-0031 and NI-0032) are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core. Temporary neutron flux detectors which provide equivalent indication may be utilized in place of installed instrumentation.

The installed source range neutron flux monitors are fission chamber detectors. The detectors monitor the neutron flux in counts per second. The instrument range covers seven decades of neutron flux (1E-1 cps to 1E +6 cps) with a 2% instrument accuracy. The detectors also provide continuous visual indication in the control room. The NIS is designed in accordance with the criteria presented in Reference 1.

APPLICABLE SAFETY ANALYSES

Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as an improperly loaded fuel assembly. The need for a safety analysis for an uncontrolled boron dilution accident is minimized by isolating all unborated water sources except as provided for by LCO 3.9.2, "Unborated Water Source Isolation Valves."

The source range neutron flux monitors satisfy Criterion 3 of the NRC Policy Statement.

LCO

This LCO requires that two source range neutron flux monitors be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity. To be OPERABLE each monitor must provide visual indication.

(continued)

BASES

ACTIONS

B.2 (continued)

are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

The Completion Time of once per 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration and to ensure that unplanned changes in boron concentration would be identified. The 12 hour Completion Time is reasonable, considering the low probability of a change in core reactivity during this time period.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

SR 3.9.3.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation."

SR 3.9.3.2

SR 3.9.3.2 is the performance of a CHANNEL CALIBRATION every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitors includes obtaining the detector preamp discriminator curves and evaluating those curves. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

(continued)

BASES

BACKGROUND (continued)

required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the door interlock mechanism may remain disabled, but one air lock door must always must be isolable by at least one air lock door with a designated individual available to close the air lock door, or at least one air lock door must be closed.

The emergency air lock will not normally be open during core alterations or fuel movement inside containment. Therefore, in the event the emergency air lock is open at the same time the personnel air lock is open, a separate individual shall be responsible for closing the emergency air lock (within 15 minutes) in addition to the individual designated to close the personnel air lock.

The requirements for containment penetration closure are sufficient to ensure fission product radiactivity release from containment due to a fuel handling accident during refueling is maintained to within the acceptance criteria of Standard Review Plan Section 15.7.4 and General Design Criteria 19.

The Containment Ventilation System consists of two 24 inch penetrations for purge and exhaust of the containment atmosphere. Each main or shutdown purge and exhaust system contains one motor operated 24 inch valve inside containment and one motor operated 24 inch valve outside containment (HV-2626A, HV-2627A, HV-2628A, and HV-2629A). A second 14 inch mini-purge and exhaust system shares each 24 inch penetration and consists of one 14 inch pneumatically operated valve inside containment and one outside of containment (HV-2626B, HV-2627B, HV-2628B, and HV-2629B). A 14 inch mini-purge line is connected to each 24 inch line between the 24 inch isolation valve and the penetration both inside and outside containment.

In MODES 1, 2, 3 and 4 the 24 inch main or shutdown purge and exhaust valves are secured in the closed position. The 14 inch mini-purge and exhaust valves may be opened in these MODES in accordance with LCO 3.6.3, Containment Isolation Valves, and are automatically closed by a Containment Ventilation Isolation signal. The instrumentation that provides the automatic isolation function for these valves is listed in LCO 3.3.6, Containment Ventilation Isolation Instrumentation.

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BASES

BACKGROUND (continued)

In MODE 6, the 24 inch main or shutdown purge and exhaust valves are used to exchange large volumes of containment air to support refueling operations or other maintenance activities. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment any open 24 inch valves are capable of being closed (LCO 3.3.6). The 14 inch mini-purge and exhaust valves, though typically not opened during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, if opened are also capable of being closed (LCO 3.3.6).

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by a closed automatic isolation valve, a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods allowed under the provisions of 10 CFR 50.59 may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment (Ref. 1).

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 2). Fuel handling accidents, analyzed in Reference 3, include dropping a single irradiated fuel assembly onto another irradiated fuel assembly.

To support the plant configuration of both air lock doors open (personnel and/or emergency air locks), it was assumed in FSAR calculations for dose analysis that the designated individual for closure of the air lock would have the air lock closed within 15 minutes of the fuel handling accident. The 15 minute duration was chosen as the limit for the response capability for the person who is designated for closing the air lock door. The NRC

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

acceptance of this specification was based on doses for a 2 hour release as well as a licensee commitment for a person designated to close the door quickly.

Also, the requirements of LCO 3.9.7, "Refueling Cavity Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 3), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). The radiological consequences of a fuel handling accident in containment have been evaluated assuming that the containment is open to the outside atmosphere. All airborne activity reaching the containment atmosphere is assumed to be exhausted to the environment within 2 hours of the accident. The calculated offsite and control room operator doses are within the acceptance criteria of Standard Review Plan 15.7.4 and GDC 19. Therefore, although the containment penetrations do not satisfy any of the NRC Policy Statement criteria, LCO 3.9.4 provides containment closure capability to minimize potential offsite doses.

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires the equipment hatch and any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed. Personnel air lock closure capability is provided by the availability of at least one door and a designated individual to close it. Emergency air lock closure capability is provided by the availability of at least one door and a designated individual to close it. For the OPERABLE containment ventilation penetrations, this LCO ensures that each penetration is isolable by the Containment Ventilation Isolation valves. The OPERABILITY requirements for LCO 3.3.6, Containment Ventilation Isolation Instrumentation ensure that radiation monitor inputs to the control room alarm exist so that operators can take timely

(continued)

BASES

LCO (continued)

action to close containment penetrations to minimize potential offsite doses. The LCO requirements for penetration closure may also be met by the automatic isolation capability of the CVI system.

Item b of this LCO includes requirements for both the emergency air lock and the personnel air lock. The personnel and emergency air locks are required by Item b of this LCO to be isolable by at least one air lock door in each air lock. Both containment personnel and emergency air lock doors may be open during movement of irradiated fuel in the containment and during CORE ALTERATIONS provided at least one air lock door is isolable in each air lock. An air lock is isolable when the following criteria are satisfied:

1. one air lock door is OPERABLE,

2. at least 23 feet of water shall be maintained over the top of the reactor vessel flange in accordance with Specification 3.9.7,

3. a designated individual is available to close the door.

OPERABILITY of a containment air lock door requires that the door seal protectors are easily removed, that no cables or hoses are being run through the air lock, and that the air lock door is capable of being quickly closed.

(continued)

BASES (continued)

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1, "Containment." In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the required open containment ventilation isolation valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each required valve operator has motive power, which will ensure that each valve is capable of being closed.

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. Including a surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1 (continued)

product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

SR 3.9.4.2

This Surveillance demonstrates that each containment ventilation isolation valve in each open containment ventilation penetration actuates to its isolation position. The 18 month Frequency maintains consistency with other similar testing requirements. Also, SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances Performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

REFERENCES

1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.
2. FSAR, Subsection 15.7.4.
3. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.

BASES

LCO (continued)

Additionally, one loop of RHR must be in operation in order to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop consists of an RHR pump, a heat exchanger, valves, piping, instruments and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold legs.

APPLICABILITY

Two RHR loops are required to be OPERABLE, and one RHR loop must be in operation in MODE 6, with the water level < 23 ft above the top of the reactor vessel flange, to provide decay heat removal and mixing of the borated coolant. Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level ≥ 23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation — High Water Level."

ACTIONS

The ACTIONS table is modified by a Note that prohibits entry into the Applicability while this LCO is not met.

A.1 and A.2

If less than the required number of RHR loops are OPERABLE, action shall be immediately initiated and continued until the RHR loop is restored to OPERABLE status and to operation or until ≥ 23 ft of water level is established above the reactor vessel flange. When the water level is ≥ 23 ft above the reactor vessel flange, the Applicability changes to that of LCO 3.9.5, and only one RHR loop is required to be OPERABLE and in operation. An immediate Completion Time is necessary for an operator to initiate corrective actions.

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