



A Subsidiary of RGS Energy Group, Inc.

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ROBERT C. MECREDDY
Vice President
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April 16, 2001

U.S. Nuclear Regulatory Commission
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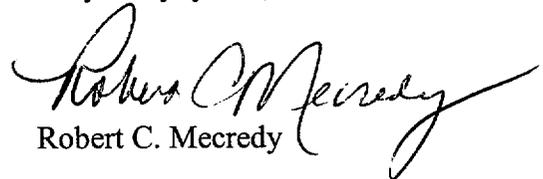
Subject: Submittal of Changes to Technical Specification Bases
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Vissing:

In accordance with Technical Specification 5.5.13, Technical Specification (TS) Bases Control Program, Rochester Gas and Electric has prepared its 2001 submittal of changes to the Technical Specification Bases. Enclosed are the revised pages, together with a list of effective pages. Change bars indicate those revisions which have been incorporated. Also, as a result of a change in the software utilized to maintain this document, the entire Bases for chapters 3.3 and 3.7 are being reissued.

The information presented in this submittal reflects changes to the Bases pursuant to commission requirements for the period of October 21, 1999 through April 15, 2001.

Very truly yours,


Robert C. Mecreddy

xc: Mr. Guy S. Vissing (Mail Stop 8C2)
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1000290

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B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Trip System (RTS) Instrumentation

BASES

BACKGROUND

Atomic Industry Forum (AIF) GDC 14 (Ref. 1) requires that the core protection systems, together with associated engineered safety features equipment, be designed to prevent or suppress conditions that could result in exceeding acceptable fuel design limits. The RTS initiates a plant shutdown, based on the values of selected plant parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during anticipated operational occurrences (AOOs) and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

The installed protection and monitoring systems have been designed to assure safe operation of the reactor at all times. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RTS, as well as specifying LCOs with respect to these parameters and other reactor system parameters and equipment.

The LSSS, defined in this specification as the Trip Setpoints, in conjunction with the associated LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs). These acceptable limits are:

- a. The Safety Limit (SL) values shall be maintained to prevent departure from nucleate boiling (DNB);
- b. Fuel centerline melt shall not occur; and
- c. The RCS pressure SL of 2735 psig shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," maintains the above values and assures that offsite dose will be within 10 CFR 100 limits (Ref. 2) during AOOs.

DBAs are events that are analyzed even though they are not expected to occur during the plant life. The DBA acceptance limit is that offsite doses shall be maintained within an acceptable fraction of 10 CFR 100 limits (Ref. 2). There are five different accident categories which are organized based on the probability of occurrence (Ref. 3). Each accident category is allowed a different fraction of the 10 CFR 100 limits, inversely proportioned to the probability of occurrence. Meeting the acceptable dose limit for an accident category is considered as having acceptable consequences for that event.

The RTS instrumentation is segmented into three distinct but interconnected modules as described in UFSAR, Chapter 7 (Ref. 4):

- a. Field transmitters or process sensors;
- b. Signal process control and protection equipment; and
- c. Reactor trip switchgear.

These modules are shown in Figure B 3.3.1-1 and discussed in more detail below.

Field Transmitters and Process Sensors

Field transmitters and process sensors provide a measurable electronic signal based on the physical characteristics of the parameter being measured. To meet the design demands for redundancy and reliability, two, three, and up to four field transmitters or sensors are used to measure required plant parameters. To account for the calibration tolerances and instrument drift, which is assumed to occur between calibrations, statistical allowances are provided. These statistical allowances provide the basis for determining acceptable "as left" and "as found" calibration values for each transmitter or sensor as provided in established plant procedures.

Signal Process Control and Protection Equipment

The process control equipment provides signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with setpoints established by safety analyses. These setpoints are defined in UFSAR, Chapter 7 (Ref. 4), Chapter 6 (Ref. 5), and Chapter 15 (Ref. 6). If the measured value of a plant parameter exceeds the predetermined setpoint, an output from a bistable is forwarded to the logic relays.

Generally, three or four channels of process control equipment are used for the signal processing of plant parameters measured by the field transmitters and sensors. If a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are typically sufficient to provide the required reliability and redundancy. If one channel fails in a direction that would not result in a partial Function trip, the Function can still be accomplished with a two-out-of-two logic. If one channel fails in a direction that a partial Function trip occurs, a trip will not occur unless a second channel fails or trips in the remaining one-out-of-two logic.

If a parameter has no measurable setpoint and is only used as an input to the protection circuits (e.g., manual trip functions) two channels with a one-out-of-two logic are sufficient. A third channel is not required since no surveillance testing is required during the time period in which the parameter is required.

If a parameter is used for input to the protection system and a control function, four channels with a two-out-of-four logic are typically sufficient to provide the required reliability and redundancy. This ensures that the circuit is able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Therefore, a single failure will neither cause nor prevent the protection function actuation. These requirements are described in IEEE-279-1971 (Ref. 7).

The two, three, and four process control channels discussed above all feed two logic trains. Figure B 3.3.1-1 shows a two-out-of-four logic function which provides input into two logic trains (Train A and B). Two logic trains are required to ensure that no single failure of one logic train will disable the RTS. Provisions to allow removing logic trains from service during maintenance are unnecessary because of the logic system's designed reliability. During normal operation, the two logic trains remain energized.

Reactor Trip Switchgear

The reactor trip switchgear includes the reactor trip breakers (RTBs) and bypass breakers as shown on Figure B 3.3.1-1. The RTBs are in the electrical power supply line from the control rod drive motor generator set power supply to the control rod drive mechanisms (CRDMs). Opening of the RTBs interrupts power to the CRDMs, which allows the shutdown rods and control rods to fall into the core by gravity and shutdown the reactor. Each RTB may be bypassed with a bypass breaker to allow testing of the RTB while the plant is at power. During normal operation, the output from the protection system is a voltage signal that energizes the undervoltage coils in the RTBs and bypass breakers, if in use. When the required logic matrix combination is completed, the protection system output voltage signal is removed, the undervoltage coils are de-energized, the breaker trip lever is actuated by the de-energized undervoltage coil, and the RTBs and bypass breakers are tripped open allowing the shutdown rods and control rods to fall into the core. Therefore, a loss of power to the protection system or RTBs will cause a reactor trip. In addition to the de-energization of the undervoltage coils, each breaker is also equipped with a shunt trip device that is energized to trip the breaker open upon receipt of a reactor trip signal from the protection system (except for the zirconium guide tube trip which

only utilizes the undervoltage coils). Either the undervoltage coil or the shunt trip mechanism is sufficient by itself to open the RTBs, thus providing diverse trip mechanisms.

APPLICABLE
SAFETY
ANALYSES,
LCO, AND
APPLICABILITY

The RTS functions to maintain the SLs during all AOOs and mitigates the consequences of DBAs which initiate in any MODE in 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 6 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 plant. 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 anticipatory trips 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 LCO generally requires OPERABILITY of three or four 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 instrumentation channels in a two-out-of-four configuration are required when one RTS channel is also used as a control system input. This configuration accounts for the possibility of the shared channel failing in such a manner that it creates a transient that requires RTS action. In this case, the RTS will still provide protection, even with random failure of one of the other three protection channels. Three operable instrumentation channels in a two-out-of-three configuration are generally required when there is no potential for control system and protection system interaction that could simultaneously create a need for a RTS trip and disable one RTS channel. The two-out-of-three and two-out-of-four configurations allow one channel to be tripped or bypassed during maintenance or testing without causing a reactor trip. Specific exceptions to the above general philosophy exist and are discussed below.

The LCO and Applicability of each RTS Function are provided in Table 3.3.1-1. Included on Table 3.3.1-1 are Trip Setpoints for all applicable RTS Functions. Trip Setpoints for RTS Functions not specifically modeled in the safety analysis are based on established limits provided in the UFSAR (Reference 4). Note that in the accompanying LCO 3.3.1, the Trip Setpoints of Table 3.3.1-1 are the LSSS. The Trip Setpoints are

the limiting values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the allowable tolerance band for CHANNEL CALIBRATION accuracy as specified within plant procedures. The channel containing the bistable is considered inoperable when the "as found" value exceeds the Trip Setpoint specified in Table 3.3.1-1.

The Trip Setpoints used in the bistables are based on the analytical limits stated in References 4, 5, and 6. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays, calibration tolerances, instrumentation uncertainties, and instrument drift are taken into account. The Trip Setpoints specified in Table 3.3.1-1 are therefore conservatively adjusted with respect to the analytical limits used in the accident analysis. A detailed description of the methodology used to verify the adequacy of the existing Trip Setpoints, including their explicit uncertainties, is provided in Reference 8.

The RTS utilizes various permissive signals to ensure reactor trip Functions are in the correct configuration for the current plant status. These permissives back up operator actions to ensure protection system Functions are not bypassed during plant conditions under which the safety analysis assumes the Function is available.

In addition to the RTS Functions listed in Table 3.3.1-1, a zirconium guide tube trip function exists. This trip function was added by RG&E to prevent potential damage to the control rod drive mechanisms when cooling down due to the different thermal expansion rates of zirconium and stainless steel. This trip function is not credited in the accident analysis, and as such, is not addressed by this LCO. However, the trip function is used for testing the RTBs since the function only actuates the RTB undervoltage mechanism (versus shunt trip).

The safety analyses and OPERABILITY requirements applicable to each RTS Function and permissive provided in Table 3.3.1-1 are discussed below:

1. Manual Reactor Trip

The Manual Reactor Trip Function ensures that the control room operator can initiate a reactor trip at any time by using either of two reactor trip pushbuttons on the main control board. A Manual Reactor Trip energizes the shunt trip device and de-energizes the undervoltage coils for the RTBs and bypass breakers. It is used at the discretion of the control room operators to shut down the reactor whenever any parameter is rapidly trending toward its Trip Setpoint or during other degrading plant conditions.

The LCO requires two Manual Reactor Trip channels to be OPERABLE. Each channel is controlled by a manual reactor trip pushbutton which actuates the reactor trip breaker in both trains. Two independent channels are required to be OPERABLE so that no single failure will disable the Manual Reactor Trip Function. This function has no adjustable trip setpoint with which to associate an LSSS, therefore no setpoints are provided.

In MODE 1 or 2, manual initiation capability of a reactor trip must be OPERABLE. These are the MODES in which the shutdown rods and/or control rods are partially or fully withdrawn from the core. In MODE 3, 4, or 5, the manual initiation Function must also be OPERABLE if the RTBs are closed and the Control Rod Drive (CRD) System is capable of withdrawing the shutdown rods or the control rods. In this condition, inadvertent control rod withdrawal is possible. In MODE 3, 4, or 5, manual initiation of a reactor trip is not required to be OPERABLE if the CRD System is not capable of withdrawing the shutdown rods or control rods, or if one or more RTBs are open. If the rods cannot be withdrawn from the core, there is no need to be able to trip the reactor because all of the rods are inserted. In MODE 6, neither the shutdown rods nor the control rods are permitted to be withdrawn and the CRDMs are disconnected from the control rods and shutdown rods. Therefore, the manual initiation Function is not required.

2. Power Range Neutron Flux

The Power Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident. The Nuclear Instrumentation System (NIS) power range detectors (N-41, N-42, N-43, and N-44) are located external to the reactor vessel and measure neutrons leaking from the core. The NIS power range detectors provide input to the CRD System for determination of automatic rod speed and direction. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation.

a. Power Range Neutron Flux-High

The Power Range Neutron Flux-High trip Function ensures that protection is provided, from all power levels, against a positive reactivity excursion leading to DNB during power operations. These reactivity excursions can be caused by rod withdrawal or reductions in RCS temperature. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

The LCO requires all four of the Power Range Neutron Flux-High trip Function channels to be OPERABLE.

In MODE 1 or 2, when a positive reactivity excursion could occur, the Power Range Neutron Flux-High trip must be OPERABLE. This Function will terminate the reactivity excursion and shut down the reactor prior to reaching a power level that could damage the fuel. In MODE 3, 4, 5, or 6, the NIS power range detectors cannot detect neutron levels in this range. In these MODES, the Power Range Neutron Flux-High trip Function is not required to be OPERABLE because the reactor is shut down and reactivity excursions into the power range are extremely unlikely. Other RTS Functions and administrative controls provide protection against reactivity additions when in MODE 3, 4, 5, or 6.

b. Power Range Neutron Flux-Low

The LCO requirement for the Power Range Neutron Flux-Low trip Function ensures that protection is provided against a positive reactivity excursion from low power or subcritical conditions.

The LCO requires all four of the Power Range Neutron Flux-Low trip Function channels (N-41, N-42, N-43, and N-44) to be OPERABLE.

In MODE 1, below 6% RTP, and in MODE 2, the Power Range Neutron Flux-Low trip must be OPERABLE. This Function may be manually blocked by the operator when two-out-of-four power range channels are greater than approximately 8% RTP (P-10 setpoint). This Function is automatically unblocked when three-out-of-four power range channels are below the P-10 setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux-High trip Function.

In MODE 3, 4, 5, or 6, the Power Range Neutron Flux-Low trip Function is not required to be OPERABLE because the reactor is shut down and the NIS power range detectors cannot detect neutron levels in this range. Other RTS trip Functions and administrative controls provide protection against positive reactivity additions or power excursions in MODE 3, 4, 5, or 6.

3. Intermediate Range Neutron Flux

The Intermediate Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition. This trip Function provides redundant protection to the Power Range Neutron Flux-Low trip Function and is not specifically modeled in the accident analysis. The NIS intermediate range detectors (N-35 and N-36) are located external to the reactor vessel and measure neutrons leaking from the core. The NIS intermediate range detectors do not provide any input to control systems. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

The LCO requires two channels of the Intermediate Range Neutron Flux trip Function to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single failure will disable this trip Function. Because this trip Function is important only during low power conditions, there is generally no need to disable channels for testing while the Function is required to be OPERABLE. Therefore, a third channel is unnecessary.

In MODE 1 below 6% RTP, and in MODE 2, The Intermediate Range Neutron Flux trip Function must be OPERABLE since there is a potential for an uncontrolled RCCA bank rod withdrawal accident. This Function may be manually blocked by the operator when two-out-of-four power range channels are greater than approximately 8% RTP (P-10 setpoint). Above 8% RTP (P-10 setpoint), the Power Range Neutron Flux-High trip provides core protection for a rod withdrawal accident. In MODE 3, 4, or 5, the Intermediate Range Neutron Flux trip Function is not required to be OPERABLE because the NIS intermediate range detectors cannot detect neutron levels in this range. Other RTS trip Functions and administrative controls provide protection against reactivity additions or power excursions in MODE 3, 4, 5, or 6.

4. Source Range Neutron Flux

The LCO requirement for the Source Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition and provides protection against boron dilution and rod ejection events. This trip Function provides redundant protection to the Power Range Neutron Flux-Low and Intermediate Range Neutron Flux trip Functions in MODE 2 and is not specifically credited in the accident analysis at these conditions. The NIS source range detectors (N-31 and N-32) 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 trip Function to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single failure will disable this trip Function. The LCO also requires one channel of the Source Range Neutron Flux trip Function to be OPERABLE in MODE 3, 4, or 5 with the CRD System not capable of rod withdrawal and all rods fully inserted. In this case, the source range Function is to provide control room indication. The outputs of the Function to RTS logic are not required to be OPERABLE when the CRD system is not capable of rod withdrawal and all rods fully inserted.

The Source Range Neutron Flux Trip Function provides protection for control rod withdrawal from subcritical, boron dilution and control rod ejection events. The Function also provides visual neutron flux indication in the control room.

In MODE 2 when both intermediate range channels are $< 5E-11$ amps (below the P-6 setpoint), the Source Range Neutron Flux trip Function must be OPERABLE. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux-Low trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the NIS source range detectors are manually de-energized by the operator and are inoperable.

In MODE 3, 4, or 5 with the CRD System capable of rod withdrawal or all rods are not fully inserted, the Source Range Neutron Flux trip Function must be OPERABLE to provide core protection against a rod withdrawal accident. If the CRD System is not capable of rod withdrawal and all rods are fully inserted, the source range detectors are not required to trip the reactor. However, their monitoring Function must be OPERABLE to monitor core neutron levels and provide indication of reactivity changes that may occur as a result of events like a boron dilution. The requirements for the NIS source range detectors in MODE 6 are addressed in LCO 3.9.2, "Nuclear Instrumentation."

5. Overtemperature ΔT

The Overtemperature ΔT trip Function is provided to ensure that the design limit departure from nucleate boiling ratio (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, T_{avg} , 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 Overtemperature ΔT trip Function monitors both variation in power and flow since a decrease in flow has the same effect on ΔT as a power increase. The Overtemperature ΔT trip Function uses the ΔT of each loop as a measure of reactor power and is compared with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature - the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;
- pressurizer pressure - the Trip Setpoint is varied to correct for changes in system pressure; and
- axial power distribution $f(\Delta I)$ - the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

Dynamic compensation is included for system piping delays from the core to the temperature measurement system.

The Overtemperature ΔT trip Function is calculated in two channels for each loop as described in Note 1 of Table 3.3.1-1. A reactor trip occurs if the Overtemperature ΔT Trip Setpoint is reached in two-out-of-four channels. Since the pressure and temperature signals are used for other control functions, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Section 7.2.5 of Reference 4 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 an unnecessary reactor trip.

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 is not required to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

6. Overpower ΔT

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding failure) under all possible overpower conditions. This trip Function also limits the required range of the Overtemperature ΔT trip Function and provides a backup to the Power Range Neutron Flux-High Setpoint trip. The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the ΔT of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature - the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;
- rate of change of reactor coolant average temperature - including dynamic compensation for the delays between the core and the temperature measurement system; and

- axial power distribution $f(\Delta I)$ - the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 2 of Table 3.3.1-1.

The Overpower ΔT trip Function is calculated in two channels for each loop as described in Note 2 of Table 3.3.1-1. A reactor trip occurs if the Overpower ΔT trip setpoint is reached in two-out-of-four channels. Since the temperature signals are used for other control functions, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation and a single failure in the remaining channels providing the protection function actuation. Section 7.2.5 of Reference 4 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 Overpower ΔT condition and may prevent an unnecessary reactor trip.

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 MODES where 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 is not required to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

7. Pressurizer Pressure

The same sensors (PT-429, PT-430, and PT-431) provide input to the Pressurizer Pressure-High and -Low trips and the Overtemperature ΔT trip with the exception that the Pressurizer Pressure-Low and Overtemperature ΔT trips also receive input from PT-449. Since the Pressurizer Pressure channels are also used for other control functions, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Section 7.2.5 of Reference 4 discusses control and protection system interactions for this function.

a. Pressurizer Pressure-Low

The Pressurizer Pressure-Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure. The LCO requires four channels of the Pressurizer Pressure-Low trip Function to be OPERABLE.

In MODE 1, when DNB is a major concern, the Pressurizer Pressure-Low trip function must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock (8.5% RTP). On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, the Pressurizer Pressure-Low trip Function is not required to be OPERABLE because no conceivable power distributions can occur that would cause DNB concerns.

b. Pressurizer Pressure-High

The Pressurizer Pressure-High trip Function ensures that protection is provided against overpressurizing the RCS. This trip Function operates in conjunction with the pressurizer relief and safety valves to prevent RCS overpressure conditions. The LCO requires three channels of the Pressurizer Pressure-High trip Function to be OPERABLE.

In MODE 1 or 2, the Pressurizer Pressure-High trip Function must be OPERABLE to help prevent RCS overpressurization and minimize challenges to the relief and safety valves. In MODE 3, 4, 5, or 6, the Pressurizer Pressure-High trip Function is not required to be OPERABLE because transients that could cause an overpressure condition will be slow to occur. Therefore, the operator will have sufficient time to evaluate plant conditions and take corrective actions. Additionally, low temperature overpressure protection systems provide overpressure protection when in or below MODE 4.

8. Pressurizer Water Level-High

The Pressurizer Water Level-High trip Function provides a backup signal for the Pressurizer Pressure-High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. This trip Function is not specifically modeled in the accident analysis.

The LCO requires three channels of the Pressurizer Water Level-High trip Function to be OPERABLE. The pressurizer level channels (LT-426, LT-427, and LT-428) are also used for other control functions. Section 7.2.5 of Reference 4 discusses control and protection system interactions for this function. The level channels do not actuate the safety valves, and the high pressure reactor trip is set below the safety valve setting. Therefore, with the slow rate of charging available, pressure overshoot due to level channel failure cannot cause the safety valve to lift before the reactor high pressure trip.

In MODE 1 or 2, when there is a potential for overfilling the pressurizer, the Pressurizer Water Level-High trip Function must be OPERABLE. In MODES 3, 4, 5, or 6, the Pressurizer Water Level-High trip Function is not required to be OPERABLE because transients that could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate plant conditions and take corrective actions.

9. Reactor Coolant Flow-Low

The Reactor Coolant Flow-Low (Single Loop) and (Two Loops) trip Functions utilize three common flow transmitters per RCS loop to generate a reactor trip above 8.5% RTP (P-7 setpoint). Flow transmitters FT-411, FT-412, and FT-413 are used for RCS Loop A and FT-414, FT-415, and FT-416 are used for RCS Loop B.

a. Reactor Coolant Flow-Low (Single Loop)

The Reactor Coolant Flow-Low (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in the RCS loop, while avoiding reactor trips due to normal variations in loop flow. Above the P-8 setpoint, (50% RTP), a loss of flow in either RCS loop will actuate a reactor trip. Each RCS loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

The LCO requires three Reactor Coolant Flow-Low (Single Loop) trip Function channels per RCS loop to be OPERABLE in MODE 1 \geq 50% RTP (above P-8 setpoint). Each loop is considered a separate Function for the purpose of this LCO.

In MODE 1 above the P-8 setpoint, a loss of flow in one RCS loop could result in DNB conditions in the core. In MODE 1 below the P-8 setpoint the Reactor Coolant Flow-Low (Single Loop) trip Function is not required to be OPERABLE because a loss of flow in one loop has been evaluated and found to be acceptable (Ref. 6).

b. Reactor Coolant Flow-Low (Two Loops)

The Reactor Coolant Flow-Low (Two Loops) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in both RCS loops while avoiding reactor trips due to normal variations in loop flow.

The LCO requires three Reactor Coolant Flow-Low (Two Loops) trip Function channels per loop to be OPERABLE in MODE 1 above 8.5% RTP (P-7 setpoint) and before the Reactor Coolant Flow-Low (Single Loop) trip Function is OPERABLE (below the P-8 setpoint). Each loop is considered a separate Function for the purpose of this LCO.

Above the P-7 setpoint and below the P-8 setpoint, a loss of flow in both loops will initiate a reactor trip. Each loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

Below the P-7 setpoint, this trip Function is not required to be OPERABLE because all reactor trips on low flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on low flow in both RCS loops is automatically enabled. Above the P-8 setpoint, the Reactor Coolant Flow-Low (Two Loops) trip Function is not required to be OPERABLE because loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

10. RCP Breaker Position

Both RCP Breaker Position trip Functions (Single Loop and Two Loops) utilize a common auxiliary contact located on each RCP. These Functions anticipate the Reactor Coolant Flow-Low trips to avoid RCS heatup that would occur before the low flow trip actuates but are not specifically credited in the accident analysis.

a. Reactor Coolant Pump Breaker Position (Single Loop)

The RCP Breaker Position (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in one RCS loop. The position of each RCP breaker is monitored. If one RCP breaker is open above 50% RTP, a reactor trip is initiated. This trip Function will generate a reactor trip before the Reactor Coolant Flow-Low (Single Loop) Trip Setpoint is reached.

The LCO requires one RCP Breaker Position trip Function channel per RCP to be OPERABLE in MODE 1 \geq 50% RTP (above the P-8 setpoint). Each RCP is considered a separate Function for the purpose of this LCO. One OPERABLE channel is sufficient for this trip Function because the RCS Flow-Low trip alone provides sufficient protection of plant SLs for loss of flow events. The RCP Breaker Position trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of a pump.

This Function measures only the discrete position (open or closed) of the RCP breaker, using a position switch. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.

In MODE 1 above the P-8 setpoint, when a loss of flow in any RCS loop could result in DNB conditions in the core, the RCP Breaker Position (Single Loop) trip Function must be OPERABLE. In MODE 1 below the P-8 setpoint, the RCP Breaker Position (Single Loop) trip Function is not required to be OPERABLE because a loss of flow in one loop has been evaluated and found to be acceptable (Ref. 6).

b. RCP Breaker Position (Two Loops)

The RCP Breaker Position (Two Loops) trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in both RCS loops. The position of each RCP breaker is monitored. If both RCP breakers are open above 8.5% RTP (P-7 setpoint) and before the RCP Breaker Position (Single Loop) trip Function is OPERABLE (below the P-8 setpoint), a reactor trip is initiated. This trip Function will generate a reactor trip before the Reactor Coolant Flow-Low (Two Loops) Trip Setpoint is reached.

The LCO requires one RCP Breaker Position trip Function channel per RCP to be OPERABLE in MODE 1 above the P-7 and below the P-8 setpoints. Each RCP is considered a separate Function for the purpose of this LCO. One OPERABLE channel is sufficient for this Function because the RCS Flow-Low trip alone provides sufficient protection of plant SLs for loss of flow events. The RCP Breaker Position trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of an RCP.

This Function measures only the discrete position (open or closed) of the RCP breaker, using a position switch. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.

In MODE 1 above the P-7 setpoint and below the P-8 setpoint, the RCP Breaker Position (Two Loops) trip Function must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow (including RCP breaker position) are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on loss of flow in both RCS loops is automatically enabled. Above the P-8 setpoint, the RCP Breaker Position (Two Loops) trip Function is not required to be OPERABLE because a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

11. Undervoltage-Bus 11A and 11B

The Undervoltage-Bus 11A and 11B reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in both RCS loops from a major network voltage disturbance. The voltage to each RCP is monitored. Above 8.5% RTP (the P-7 setpoint), an undervoltage condition detected on both Buses 11A and 11B will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow-Low (Two Loops) Trip Setpoint is reached. Time delays are incorporated into the Undervoltage Bus 11A and 11B channels to prevent reactor trips due to momentary electrical power transients.

The LCO requires two Undervoltage-Bus 11A and 11B trip Function channels per bus to be OPERABLE in MODE 1 above the P-7 setpoint. Each bus is considered a separate Function for the purpose of this LCO.

Below the P-7 setpoint, the Undervoltage-Bus 11A and 11B trip Function is not required to be OPERABLE because all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on Undervoltage-Bus 11A and 11B is automatically enabled.

12. Underfrequency-Bus 11A and 11B

The Underfrequency-Bus 11A and 11B reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in both RCP loops from a major network frequency disturbance. An underfrequency condition will slow down the pumps, thereby reducing their coastdown time following a pump trip. The proper coastdown time is required so that reactor heat can be removed immediately after reactor trip. The frequency of each RCP bus is monitored. Above 8.5% RTP (the P-7 setpoint), a loss of frequency detected on both RCP buses will initiate a

reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow-Low (Two Loops) Trip Setpoint is reached. Time delays are incorporated into the Underfrequency RCPs channels to prevent reactor trips due to momentary electrical power transients.

The LCO requires two Underfrequency-Bus 11A and 11B channels per bus to be OPERABLE in Mode 1 above the P-7 setpoint. Each bus is considered a separate Function for the purpose of this LCO.

Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on Underfrequency-Bus 11A and 11B is automatically enabled.

13. Steam Generator Water Level-Low Low

The Steam Generator (SG) Water Level-Low Low trip Function ensures that protection is provided against a loss of heat sink and actuates the Auxiliary Feedwater (AFW) System prior to uncovering the SG tubes. The SGs are the heat sink for the reactor. In order to act as a heat sink, the SGs must contain a minimum amount of water. A narrow range low low level in any SG is indicative of a loss of heat sink for the reactor. Three level transmitters per SG (LT-461, LT-462, and LT-463 for SG A and, LT-471, LT-472, and LT-473 for SG B) provide input to the SG Level Control System. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. This Function also performs the Engineered Safety Feature Actuation System (ESFAS) function of starting the AFW pumps on low low SG level. The AFW System is the safety related backup source of water to ensure that the SGs remain the heat sink for the reactor.

The LCO requires three trip Function channels of SG Water Level-Low Low per SG to be OPERABLE in MODES 1 and 2. Each SG is considered a separate Function for the purpose of this LCO.

In MODE 1 or 2, the SG Water Level-Low Low trip Function must be OPERABLE to ensure that a heat sink is available to the reactor. In MODE 3, 4, 5, or 6, the SG Water Level-Low Low trip Function is not required to be OPERABLE because the reactor is not operating. Decay heat removal is accomplished by the AFW System in MODE 3 and by the Residual Heat Removal (RHR) System in MODE 4, 5, or 6.

14. Turbine Trip

Credit for these trip Functions is not credited in the accident analysis.

a. Turbine Trip-Low Autostop Oil Pressure

The Turbine Trip-Low Autostop Oil Pressure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip from a power level above the P-9 setpoint. Below the P-9 setpoint this action will not actuate a reactor trip. The trip Function anticipates the loss of secondary heat removal capability that occurs when the stop valves close. Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. Three pressure switches monitor the control oil pressure in the Autostop Oil System. A low pressure condition sensed by two-out-of-three pressure switches will actuate a reactor trip. These pressure switches do not provide any input to the control system. The plant is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure-High trip Function and RCS integrity is ensured by the pressurizer safety valves.

The LCO requires three trip Function channels of Turbine Trip-Low Autostop Oil Pressure to be OPERABLE in MODE 1 above P-9.

Below the P-9 setpoint, the Turbine Trip-Low Autostop Oil Pressure trip Function is not required to be OPERABLE because load rejection can be accommodated by the steam dump system. Therefore, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, the turbine is not operating, therefore, there is no potential for a turbine trip.

b. Turbine Trip-Turbine Stop Valve Closure

The Turbine Trip-Turbine Stop Valve Closure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip from a power level above the P-9 setpoint. Below the P-9 setpoint this action will not actuate a reactor trip. The trip Function anticipates the loss of secondary heat removal capability that occurs when the stop valves close. Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. This trip Function will not and is not required to operate in the presence of a single channel failure. The plant is designed to withstand a

complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure-High trip Function, and RCS integrity is ensured by the pressurizer safety valves. This trip Function is diverse to the Turbine Trip-Low Autostop Oil Pressure trip Function. Each turbine stop valve is equipped with one limit switch that inputs to the RTS. If both limit switches indicate that the stop valves are closed, a reactor trip is initiated.

This Function only measures the discrete position (open or closed) of the turbine stop valves. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.

The LCO requires two Turbine Trip-Turbine Stop Valve Closure trip Function channels, one per valve, to be OPERABLE in MODE 1 above P-9. Both channels must trip to cause reactor trip.

Below the P-9 setpoint, the Turbine Trip-Turbine Stop Valve Closure trip Function is not required to be OPERABLE because a load rejection can be accommodated by the steam dump system. Therefore, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, the turbine is not operating, therefore there is no potential for a turbine trip.

15. Safety Injection Input from Engineered Safety Feature Actuation System

The Safety Injection (SI) Input from ESFAS ensures that if a reactor trip has not already been generated by the RTS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal that initiates SI. This trip is assumed in the safety analyses for the loss of coolant accident (LOCA). However, other transients and accidents take credit for varying levels of ESF performance and rely upon rod insertion, except for the most reactive rod that is assumed to be fully withdrawn, to ensure reactor shutdown. Therefore, a reactor trip is initiated every time an SI signal is present.

Trip Setpoints are not applicable to this Function. The SI Input is provided by relays in the ESFAS. Therefore, there is no measurement signal with which to associate an LSSS.

The LCO requires two trip Function channels of SI Input from ESFAS to be OPERABLE in MODE 1 or 2.

A reactor trip is initiated every time an SI signal is present. Therefore, this trip Function must be OPERABLE in MODE 1 or 2, when the reactor is critical, and must be shut down in the event of an accident. In MODE 3, 4, 5, or 6, the reactor is not critical, and this trip Function does not need to be OPERABLE.

16. Reactor Trip System Interlocks

Reactor protection interlocks (i.e., permissives) are provided to ensure reactor trips are in the correct configuration for the current plant status. They back up operator actions to ensure protection system Functions are not bypassed during plant 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 Permissive

The Intermediate Range Neutron Flux, P-6 permissive is actuated when any NIS intermediate range channel goes approximately one decade (1 E-10 amps) 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 permissive 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 by use of two defeat push buttons. 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. When the source range trip is blocked, the high voltage to the detectors is also removed; and
- on decreasing power, the P-6 interlock automatically energizes the NIS source range detectors and enables the Source Range Neutron Flux reactor trip at 5E-11 amps.

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

Above the P-6 permissive setpoint, the Source Range Neutron Flux reactor trip will be blocked, and this Function is no longer required.

In MODE 3, 4, 5, or 6 the P-6 permissive does not have to be OPERABLE because the Source Range is providing the required core protection.

b. Low Power Reactor Trips Block, P-7 Permissive

The Low Power Reactor Trips Block, P-7 interlock is actuated by input from either the Power Range Neutron Flux, P-10, or from first stage turbine pressure. The LCO requirement for the P-7 permissive allows the bypass of the following Functions:

- Pressurizer Pressure-Low;
- Reactor Coolant Flow-Low (Two Loops);
- RCP Breaker Position (Two Loops);
- Undervoltage-Bus 11A and 11B; and
- Underfrequency-Bus 11A and 11B.

These reactor trip functions are not required below the P-7 setpoint since the RCS is capable of providing sufficient natural circulation without any RCP running.

The LCO requires four channels of Low Power Reactor Trips Block, P-7 permissive to be OPERABLE in MODE 1 \geq 8.5% RTP.

In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the permissive performs its Function when power level drops below 8.5% power, which is in MODE 1.

c. Power Range Neutron Flux, P-8 Permissive

The Power Range Neutron Flux, P-8, permissive is actuated at approximately 49% power as determined by two-out-of-four NIS power range detectors. The P-8 interlock allows the Reactor Coolant Flow-Low (Single Loop) and RCP Breaker Position (Single Loop) reactor trips on low flow in one or more RCS loops to be blocked so that a loss of a single loop will not cause a reactor trip. The LCO requirement for this trip Functions ensures that protection is provided against a loss of flow in any RCS loop that could result in DNB conditions in the core when $\geq 50\%$ power.

The LCO requires four channels of Power Range Neutron Flux, P-8 interlock to be OPERABLE in MODE 1 $\geq 50\%$ RTP.

In MODE 1, a loss of flow in one RCS loop could result in DNB conditions, so the Power Range Neutron Flux, P-8 permissive must be OPERABLE. In MODE 1 $< 50\%$ RTP, this function is not required to be OPERABLE because a loss of flow in one loop will not result in DNB. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the core is not producing sufficient power to be concerned about DNB conditions.

d. Power Range Neutron Flux, P-9 Permissive

The Power Range Neutron Flux, P-9 permissive is actuated at approximately 50% power as determined by two-out-of-four NIS power range detectors if the Steam Dump System is available and at 8% if the Steam Dump System is unavailable. The LCO requirement for this Function ensures that the Turbine Trip-Low Autostop Oil Pressure and Turbine Trip-Turbine Stop Valve Closure reactor trips are enabled above the P-9 setpoint. Above the P-9 setpoint, a turbine trip will cause a load rejection beyond the capacity of the Steam Dump System and RCS. A reactor trip is automatically initiated on a turbine trip when it is above the P-9 setpoint, to minimize the transient on the reactor.

The LCO require four channels of Power Range Neutron Flux, P-9 permissive to be OPERABLE in MODE 1 above the permissive setpoint.

In MODE 1 above the permissive setpoint, a turbine trip could cause a load rejection beyond the capacity of the Steam Dump System and RCS, so the Power Range Neutron Flux interlock must be OPERABLE. In MODE 1 below the permissive setpoint and MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the 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 Permissive

The Power Range Neutron Flux, P-10 permissive is actuated at approximately 8% power, as determined by two-out-of-four NIS power range detectors. If power level falls below 8% RTP on 3 of 4 channels, the nuclear instrument trips will be automatically unblocked. The LCO requirement for the P-10 permissive ensures that the following Functions are performed:

- on increasing power, the P-10 permissive allows the operator to manually block the Intermediate Range Neutron Flux and Power Range Neutron Flux-low reactor trips;
- on increasing power, the P-10 permissive automatically provides a backup signal to the P-6 permissive to block the Source Range Neutron Flux reactor trip, and also to de-energize the NIS source range detector;
- 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).

The LCO requires four channels of Power Range Neutron Flux, P-10 interlock to be OPERABLE in MODE 1 < 6% RTP and MODE 2.

OPERABILITY in MODE 1 < 6% RTP ensures the Function is available to perform its decreasing power Functions in the event of a reactor shutdown. This Function must also be OPERABLE in MODE 2 to ensure that core protection is providing during a startup or shutdown by the Power Range Neutron Flux-Low and Intermediate Range Neutron Flux reactor trips. In MODE 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at power and the Source Range Neutron Flux reactor trip provides core protection.

17. Reactor Trip Breakers

This trip Function applies to the RTBs exclusive of individual trip mechanisms. The OPERABILITY requirement for the individual trip mechanisms is provided in Function 18 below. The LCO requires two OPERABLE trains of trip breakers. A trip breaker train consists of all trip breakers associated with a single RTS logic train that are racked in, closed, and capable of supplying power to the CRD System. Thus, the train may consist of the main breaker, bypass breaker, or main breaker and bypass breaker, depending upon the system configuration. Two OPERABLE trains ensure no single failure can disable the RTS trip capability.

These trip Functions must be OPERABLE in MODE 1 or 2 because the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the CRD System is capable of rod withdrawal and all rods are not fully inserted.

18. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms

The LCO requires both the Undervoltage and Shunt Trip Mechanisms to be OPERABLE for each RTB that is in service. The trip mechanisms are not required to be OPERABLE for trip breakers that are open, racked out, incapable of supplying power to the CRD System, or declared inoperable under Function 17 above. OPERABILITY of both trip mechanisms on each breaker ensures that no single trip mechanism failure will prevent opening any breaker on a valid signal.

These trip Functions must be OPERABLE in MODE 1 or 2 because the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the CRD System is capable of rod withdrawal and all rods are not fully inserted.

19. Automatic Trip Logic

The LCO requirement for the RTBs (Functions 17 and 18) and Automatic Trip Logic (Function 19) ensures that means are provided to interrupt the power to allow the rods to fall into the reactor core. Each RTB is equipped with an undervoltage coil and a shunt trip coil to trip the breaker open when needed. Each RTB is also equipped with a redundant bypass breaker to allow testing of the trip breaker while the plant is at power. The reactor trip signals generated by the RTS Automatic Trip Logic cause the RTBs and associated bypass breakers to open and shut down the reactor.

The LCO requires two trains of RTS Automatic Trip Logic to be OPERABLE. Having two OPERABLE trains ensures that failure of a single logic train will not prevent reactor trip.

These trip Functions must be OPERABLE in MODE 1 or 2 because the reactor is critical. In MODE 3, 4, or 5, these RTS trip Functions must be OPERABLE when the CRD System is capable of rod withdrawal and all rods are not fully inserted.

The RTS instrumentation satisfies Criterion 3 of the NRC Policy Statement.

ACTIONS

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 3.3.1-1.

In the event a channel's Trip Setpoint is found nonconservative with respect to analytical values specified in plant procedures, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected.

As shown on Figure B 3.3.1-1, the RTS is comprised of multiple interconnected modules and components. For the purpose of this LCO, a channel is defined as including all related components from the field instrument to the Automatic Trip Logic (Function 19 in Table 3.3.1-1). Therefore, a channel may be inoperable due to the failure of a field instrument or a bistable failure which affects one or both RTS trains that is comprised of the RTBs and Automatic Trip Logic Function. The only exception to this are the Manual Reactor Trip and SI Input from ESFAS trip Functions which are defined strictly on a train basis (i.e., failure of these Functions may only affect one RTS train).

A.1

Condition A applies to all RTS protection functions. Condition A addresses the situation where one required channel for one or more Functions is inoperable or if both source range channels are inoperable. The Required Action is to refer to Table 3.3.1-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

When the number of inoperable channels in a trip Function exceed those specified in all related Conditions associated with a trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 must be immediately entered if the trip Function is applicable in the current MODE of operation. This essentially applies to the loss of more than one channel of any RTS Function except with respect to Condition H.

B.1

Condition B applies to the Manual Reactor Trip Function in MODE 1 or 2 and in MODES 3, 4, and 5 with the CRD system capable of rod withdrawal or all rods not fully inserted. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 48 hours. In this Condition, the remaining OPERABLE channel is adequate to perform the required safety function.

The Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE, and the low probability of an event occurring during this interval.

C.1, C.2, and C.3

If the Manual Reactor Trip Function cannot be restored to OPERABLE status within the allowed 48 hour Completion Time of Condition B, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours, action must be initiated within 6 hours to ensure that all rods are fully inserted, and the Control Rod Drive System must be placed in a condition incapable of rod withdrawal within 7 hours. The Completion Times provide adequate time to exit the MODE of Applicability from full power operation in an orderly manner without challenging plant systems based on operating experience.

D.1

Condition D applies to the following reactor trip Functions:

- Power Range Neutron Flux-High;
- Power Range Neutron Flux-Low;

- Overtemperature ΔT ;
- Overpower ΔT ;
- Pressurizer Pressure-High;
- Pressurizer Water Level-High; and
- SG Water Level-Low Low.

With one channel inoperable, the channel must be restored to OPERABLE status or placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition. For the Power Range Neutron Flux-High, Power Range Neutron Flux-Low, Overtemperature ΔT , and Overpower ΔT functions, this results in a one-out-of-three logic for actuation. For the Pressurizer Pressure-High and Pressurizer Water Level-High Functions, this results in a one-out-of-two logic for actuation. For the SG Water Level-Low Low Function, this results in a one-out-of-two logic per each affected SG for actuation. The 6 hours allowed to place the inoperable channel in the tripped condition is consistent with Reference 9.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypass condition for up to 4 hours while performing surveillance testing of other channels. This includes placing the inoperable channel in the bypass condition to allow setpoint adjustments of other channels when required to reduce the setpoint in accordance with other Technical Specifications. This 4 hours is applied to each of the remaining OPERABLE channels. The 4 hour time limit is consistent with Reference 9.

E.1 and E.2

Condition E applies to the Intermediate Range Neutron Flux trip Function when THERMAL POWER is above the P-6 setpoint (5E-11 amp as derived from a bistable circuit of the intermediate range channels) and below the P-10 setpoint (6% RTP as derived from a bistable circuit of the Power Range channels) and one channel is inoperable. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs a monitoring and protection function. With one NIS intermediate range channel inoperable, 2 hours is allowed to either reduce THERMAL POWER below the P-6 setpoint or increase THERMAL POWER above the P-10 setpoint. If THERMAL POWER is greater than the P-10 setpoint, the NIS power range detectors perform the monitoring and protection functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment above P-10 or below P-6 and take into account the redundant capability afforded by the redundant OPERABLE channel, and the low probability of its failure during this period. This action does not require the inoperable channel to be tripped because the Function uses one-out-

of-two logic. Tripping one channel would trip the reactor. Thus, the Required Actions specified in this Condition are only applicable when channel inoperability does not result in reactor trip.

Required Action E.2 is modified by a Note which states that the option to increase THERMAL POWER is not allowed if both intermediate range channels are inoperable or if THERMAL POWER is $< 5E-11$ amps. This prevents the plant from increasing THERMAL POWER when the trip capability of the Intermediate Range Neutron Flux trip Function is not available or if the plant has not yet entered this trip Function's MODE of Applicability.

F.1, F.2, and F.3

Condition F applies to the Source Range Neutron Flux trip Function when in MODE 2, below the P-6 setpoint. In this Condition, the NIS source range performs the monitoring and protection functions. With two channels inoperable, the RTBs and RTBBs must be opened immediately. With the RTBs and RTBBs opened, the core is in a more stable condition.

With one channel inoperable, operations involving positive reactivity additions shall be suspended immediately. This will preclude any power escalation since with only one source range channel OPERABLE, core protection is severely reduced. The inoperable channel must also be restored within 48 hours.

G.1

If the Required Actions of Condition D, E, or F cannot be met within the specified Completion Times, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

H.1, H.2, and H.3

Condition H applies to an inoperable source range channel in MODE 3, 4, or 5 with the CRD System capable of rod withdrawal or all rods not fully inserted. In this Condition, the NIS source range performs the monitoring and protection functions. With two channels inoperable, at least one channel must be restored to OPERABLE status within 1 hour. The Completion Time of 1 hour is reasonable considering the low probability of an event occurring during this interval.

With one of the source range channels inoperable, operations involving positive reactivity additions must be suspended immediately and 48 hours is allowed to restore it to OPERABLE status. The suspension of positive reactivity additions will preclude any power escalation.

I.1 and I.2

If the Source Range trip Function cannot be restored to OPERABLE status within the required Completion Time of Condition H, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, action must be immediately initiated to fully insert all rods. Additionally, the CRD System must be placed in a condition incapable of rod withdrawal within 1 hour. The Completion Time of 1 hour is sufficient to accomplish the Required Action, and takes into account the low probability of an event occurring during this interval.

J.1 and J.2

Condition J applies when the required Source Range Neutron Flux channel is inoperable in MODE 3, 4, or 5 with the CRD System not capable of rod withdrawal and all rods are fully inserted. In this Condition, the NIS source range performs the monitoring function. With no source range channels OPERABLE, operations involving positive reactivity additions shall be suspended immediately. This will preclude any power escalation.

Also, the SDM must be verified once within 12 hours and every 12 hours thereafter as per SR 3.1.1.1, SDM verification. With no source range channels OPERABLE, core protection is severely reduced. Verifying the SDM once per 12 hours allows sufficient time to perform the calculations and determine that the SDM requirements are met and to ensure that the core reactivity has not changed. Required Action J.1 precludes any positive reactivity additions; therefore, core reactivity should not be increasing, and a 12 hour Frequency is adequate. The Completion Time of once per 12 hours is based on operating experience in performing the Required Actions and the knowledge that plant conditions will change slowly.

K.1

Condition K applies to the following reactor trip Functions:

- Pressurizer Pressure-Low;
- Reactor Coolant Flow-Low (Two Loops);
- RCP Breaker Position (Two Loops);
- Undervoltage-Bus 11A and 11B; and
- Underfrequency-Bus 11A and 11B.

With one channel inoperable, the inoperable channel must be restored to OPERABLE status or placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one additional channel to initiate a reactor trip. The 6 hours allowed to place the channel in the tripped condition is consistent with Reference 9 if the inoperable channel cannot be restored to OPERABLE status.

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channel(s), and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with Condition K.

For the Reactor Coolant Flow-Low (Two Loops) Function, Condition K applies on a per loop basis. For the RCP Breaker Position (Two Loops) Function, Condition K applies on a per RCP basis. For Undervoltage-Bus 11A and 11B and underfrequency-Bus 11A and 11B, Condition K applies on a per bus basis. This allows one inoperable channel from each loop, RCP, or bus to be considered on a separate condition entry basis.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing surveillance testing of the other channels. The 4 hour time limit is consistent with Reference 9. The 4 hours is applied to each of the remaining OPERABLE channels.

L.1

If the Required Action and Completion Time of Condition K is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in MODE 1 < 8.5% RTP (P-7 setpoint) at which point the Function is no longer required. An alternative is not provided for increasing THERMAL POWER above the P-8 setpoint for the Reactor Coolant Flow-Low (Two Loops) and RCP Breaker Position (Two Loops) trip Functions since this places the plant in Condition M. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 1 < 8.5% RTP from full power conditions in an orderly manner and without challenging plant systems.

M.1

Condition M applies to the Reactor Coolant Flow-Low (Single Loop) reactor trip Function. Condition M applies on a per loop basis. With one channel per loop inoperable, the inoperable channel must be restored to OPERABLE status or placed in the tripped condition within 6 hours. The 6 hours allowed to restore the channel to OPERABLE status or place in trip is consistent with Reference 9.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing surveillance testing of the other channels. The 4 hours is applied to each of the two OPERABLE channels. The 4 hour time limit is consistent with Reference 9.

N.1

Condition N applies to the RCP Breaker Position (Single Loop) trip Function. Condition N applies on a per loop basis. There is one breaker position device per RCP breaker. With one channel per RCP inoperable, the inoperable channel must be restored to OPERABLE status within 6 hours. The 6 hours allowed to restore the channel to OPERABLE status is consistent with Reference 9.

O.1

If the Required Action and associated Completion Time of Condition M or N is not met, the plant must be placed in a MODE where the Functions are not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced to < 50% RTP (P-8 setpoint) within the next 6 hours. The Completion Time of 6 hours is consistent with Reference 9.

P.1

Condition P applies to Turbine Trip on Low Autostop Oil Pressure or on Turbine Stop Valve Closure in MODE 1 above the P-9 setpoint. With one channel inoperable, the inoperable channel must be restored to OPERABLE status or placed in the tripped condition within 6 hours. If placed in the tripped Condition, this results in a partial trip condition requiring only one additional channel to initiate a reactor trip. The 6 hours allowed to place the inoperable channel in the tripped condition is consistent with Reference 9.

The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing surveillance testing of the other channels. The 4 hours is applied to each remaining OPERABLE channel. The 4 hour time limit is consistent with Reference 9.

Q.1, Q.2.1, and Q.2.2

If the Required Action and Associated Completion Time of Condition P are not met, the plant must be placed in a MODE where the Turbine Trip Functions are no longer required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced to < 50% RTP (P-9 setpoint) within the next 6 hours. The Completion Time of 6 hours is consistent with Reference 9.

The Steam Dump system must also be verified OPERABLE within 7 hours or THERMAL POWER must be reduced to < 8% RTP. This ensures that either the secondary system or RCS is capable of handling the heat rejection following a reactor trip. The Completion Times are reasonable considering the need to perform the actions in an orderly manner and the low probability of an event occurring in this time.

R.1

Condition R applies to the SI Input from ESFAS reactor trip and the RTS Automatic Trip Logic in MODES 1 and 2. With one train inoperable, 6 hours is allowed to restore the train to OPERABLE status. The Completion Time of 6 hours to restore the train to OPERABLE status 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 during this interval.

The Required Action has been modified by a Note that allows bypassing one train up to 4 hours for surveillance testing, provided the other train is OPERABLE.

S.1 and S.2

Condition S applies to the P-6, P-7, P-8, P-9, and P-10 permissives. With one channel inoperable, the associated interlock must be verified to be in its required state for the existing plant condition within 1 hour or the associated RTS channel(s) must be declared inoperable. These actions are conservative for the case where power level is being raised. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions.

T.1

Condition T applies to the RTBs in MODES 1 and 2. With one train inoperable, 1 hour is allowed to restore the train to OPERABLE status. The 1 hour Completion Time is based on operating experience and the minimum amount of time allowed for manual operator actions.

The Required Action has been modified by two Notes. Note 1 allows one train to be bypassed for up to 2 hours for surveillance testing, provided the other train is OPERABLE. Note 2 allows one RTB to be bypassed for up to 6 hours for maintenance on undervoltage or shunt trip mechanisms if the other RTB train is OPERABLE. The 6 hours for maintenance is in addition to the 2 hours for surveillance testing (e.g., if a RTB fails 1 hour into its testing window, it must be restored within 6 additional hours (or 7 hours from start of test)).

U.1 and U.2

Condition U applies to the RTB Undervoltage and Shunt Trip Mechanisms (i.e., diverse trip features) in MODES 1 and 2. Condition U applies on a RTB basis. This allows one diverse trip feature to be inoperable on each RTB. However, with two diverse trip features inoperable (i.e., one on each of two different RTBs), at least one diverse trip feature must be restored to OPERABLE status within 1 hour. The Completion Time of 1 hour is reasonable considering the low probability of an event occurring during this time interval.

With one trip mechanism for one RTB inoperable, it must be restored to an OPERABLE status within 48 hours. The affected RTB shall not be bypassed while one of the diverse trip features is inoperable except for the time required to perform maintenance to one of the diverse trip features. The allowable time for performing maintenance of the diverse trip features is 6 hours for the reasons stated under Condition T. The Completion Time of 48 hours for Required Action U.2 is reasonable considering that in this Condition there is one remaining diverse trip feature for the affected RTB, and one OPERABLE RTB capable of performing the safety function and given the low probability of an event occurring during this interval.

V.1

If the Required Action and Associated Completion Time of Condition R, S, T, or U is not met, the plant must be placed in a MODE where the Functions are no longer required to be OPERABLE. To achieve this status, the plant must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner without challenging plant systems.

It should be noted that for inoperable channels of Functions 16a, 16b, 16c, and 16d, the MODE of Applicability will be exited before Required Action V.1 is completed. Therefore, the plant shutdown may be stopped upon exiting the MODE of Applicability per LCO 3.0.2.

W.1 and W.2

Condition W applies to the following reactor trip Functions in MODE 3, 4, or 5 with the CRD System capable of rod withdrawal or all rods not fully inserted:

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

With two trip mechanisms inoperable, at least one trip mechanism must be restored to OPERABLE status within 1 hour. The Completion Time of 1 hour is reasonable considering the low probability of an event occurring during this time interval.

With one trip mechanism or train inoperable, the inoperable trip mechanism or train must be restored to OPERABLE status within 48 hours. For the trip mechanisms, Condition W applies on a RTB basis. This allows one diverse trip feature to be inoperable on each RTB. However, with two diverse trip features inoperable (i.e., one on each of two different RTBs), at least one diverse trip feature must be restored to OPERABLE status within 1 hour.

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.

X.1 and X.2

If the Required Action and Associated Completion Time of Condition W is not met, the plant must be placed in a MODE where the Functions are no longer required. To achieve this status, action must be initiated immediately to fully insert all rods and the CRD System must be incapable of rod withdrawal within 1 hour. These Completion Times are reasonable, based on operating experience to exit the MODE of Applicability in an orderly manner.

SURVEILLANCE REQUIREMENTS

The SRs for each RTS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RTS Functions.

Note that each channel of process protection supplies both trains of the RTS. When testing Channel 1, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel 2, Channel 3, and Channel 4 (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies (Ref. 8).

SR 3.3.1.1

A CHANNEL CHECK is required for the following RTS trip functions:

- Power Range Neutron Flux-High;
- Power Range Neutron Flux-Low;

- Intermediate Range Neutron Flux;
- Source Range Neutron Flux;
- Overtemperature ΔT ;
- Overpower ΔT ;
- Pressurizer Pressure-Low;
- Pressurizer Pressure-High;
- Pressurizer Water Level-High;
- Reactor Coolant Flow-Low (Single Loop);
- Reactor Coolant Flow-Low (Two Loops); and
- SG Water Level-Low Low

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or of more serious instrument conditions. A CHANNEL CHECK will detect gross channel failure; thus, it is a verification that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Channel check acceptance criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

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

SR 3.3.1.2

This SR compares the calorimetric heat balance calculation to the NIS Power Range Neutron Flux-High channel output every 24 hours. If the calorimetric exceeds the NIS channel output by > 2% RTP, the NIS is still OPERABLE but must be adjusted. If the NIS channel output cannot be properly adjusted, the channel is then declared inoperable.

This SR is modified by a Note which states that this Surveillance is required to be performed within 12 hours after power is $\geq 50\%$ RTP. At lower power levels, calorimetric data are inaccurate.

The Frequency of every 24 hours is based on plant operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate the change in the absolute difference between NIS and heat balance calculated powers rarely exceeds 2% in any 24 hour period.

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

This SR compares the incore system to the NIS channel output every 31 effective full power days (EFPD). If the absolute difference is $\geq 3\%$, the NIS channel is still OPERABLE, but must be readjusted. If the NIS channel cannot be properly readjusted, the channel is then declared inoperable. This surveillance is performed to verify the $f(\Delta I)$ input to the Overttemperature ΔT Function.

This SR is modified by two Notes. Note 1 clarifies that the Surveillance is required to be performed within 7 days after THERMAL POWER is $\geq 50\%$ RTP but prior to exceeding 90% RTP following each refueling and if it has not been performed within the last 31 EFPD. Note 2 states that performance of SR 3.3.1.6 satisfies this SR since it is a more comprehensive test.

The Frequency of every 31 EFPD is based on plant operating experience, considering instrument reliability and operating history data for instrument drift. Also, the slow changes in neutron flux during the fuel cycle can be detected during this interval.

SR 3.3.1.4

This SR is the performance of a TADOT every 31 days on a STAGGERED TEST BASIS of the RTB, and the RTB Undervoltage and Shunt Trip Mechanisms. This test shall verify OPERABILITY by actuation of the end devices.

The test shall include separate verification of the undervoltage and shunt trip mechanisms except for the bypass breakers which do not require separate verification since no capability is provided for performing such a test at power. The independent test for bypass breakers is included in SR 3.3.1.11. However, the bypass breaker test shall include a local shunt trip. This test must be performed on the bypass breaker prior to placing it in service to take the place of a RTP.

The Frequency of every 31 days on a STAGGERED TEST BASIS is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.1.5

This SR is the performance of an ACTUATION LOGIC TEST on the RTS Automatic Trip Logic every 31 days on a STAGGERED TEST BASIS. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. All possible logic combinations, with and without applicable permissives, are tested for each protection function. The Frequency of every 31 days on a STAGGERED TEST BASIS is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.1.6

This SR is a calibration of the excore channels to the incore channels every 92 EFPD. If the measurements do not agree, the excore channels are still OPERABLE but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are then declared inoperable. This surveillance is performed to verify the $f(\Delta I)$ input to the Overtemperature ΔT Function.

This SR has been modified by a Note stating that this Surveillance is required to be performed within 7 days after THERMAL POWER is $\geq 50\%$ RTP but prior to exceeding 90% RTP following each refueling.

The Frequency of 92 EFPD is adequate based on industry operating experience, considering instrument reliability and operating history data for instrument drift.

SR 3.3.1.7

This SR is the performance of a COT every 92 days for the following RTS functions:

- Power Range Neutron Flux-High;
- Source Range Neutron Flux (in MODE 3; 4, or 5 with CRD System capable of rod withdrawal or all rods not fully inserted);
- Overtemperature ΔT ;

- Overpower ΔT ;
- Pressurizer Pressure-Low;
- Pressurizer Pressurizer-High;
- Pressurizer Water Level-High;
- Reactor Coolant Flow-Low (Single Loop);
- Reactor Coolant Flow-Low (Two Loops); and
- SG Water Level-Low Low

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be within the Trip Setpoint of Table 3.3.1-1. The "as left" values must be consistent with the drift allowance used in the setpoint methodology (Ref. 8).

This SR is modified by a Note that provides a 4 hour delay in the requirement to perform this surveillance for source range instrumentation when entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the plant is in MODE 3 with the RTBs closed for greater than 4 hours, this SR must be performed within 4 hours after entry into MODE 3.

The Frequency of 92 days is consistent with Reference 9.

SR 3.3.1.8

This SR is the performance of a COT as described in SR 3.3.1.7 for the Power Range Neutron Flux-Low, Intermediate Range Neutron Flux, and Source Range Neutron Flux (MODE 2), except that this test also includes verification that the P-6 and P-10 interlocks are in their required state for the existing plant condition. This SR is modified by two Notes that provide a 4 hour delay in the requirement to perform this surveillance. These Notes allow a normal shutdown to be completed and the plant removed from the MODE of Applicability for this surveillance without a delay to perform the testing required by this surveillance. The Frequency of every 92 days applies if the plant remains in the MODE of Applicability after the initial performances of prior to reactor startup and 4 hours after reducing power below P-10 or P-6.

The MODE of Applicability for this surveillance is < P-10 for the power range low and intermediate range channels and < P-6 for the Source range channels. Once the plant is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 or < P-6 for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the 4 hour limit. Four hours is a reasonable time to complete the required testing or place the plant in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range low channels are OPERABLE prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for periods > 4 hours.

SR 3.3.1.9

This SR is the performance of a TADOT for the Undervoltage-Bus 11A and 11B and Underfrequency-Bus 11A and 11B trip Functions. The Frequency of every 92 days is consistent with Reference 9.

This SR is modified by a Note that excludes verification of setpoints from the TADOT. Since this SR applies to Bus 11A and 11B undervoltage and underfrequency relays, setpoint verification requires elaborate bench calibration and is accomplished during the CHANNEL CALIBRATION required by SR 3.3.1.10.

SR 3.3.1.10

This SR is the performance of a CHANNEL CALIBRATION for the following RTS Functions:

- Power Range Neutron Flux-High;
- Power Range Neutron Flux-Low;
- Intermediate Range Neutron Flux;
- Source Range Neutron Flux;
- Overtemperature ΔT ;
- Overpower ΔT ;
- Pressurizer Pressure-Low;
- Pressurizer Pressure-High;
- Pressurizer Water Level-High;
- Reactor Coolant Flow-Low (Single Loop);
- Reactor Coolant Flow-Low (Two Loops);

- Undervoltage-Bus 11A and 11B;
- Underfrequency-Bus 11A and 11B;
- SG Water Level-Low Low;
- Turbine Trip-Low Autostop Oil Pressure; and
- Reactor Trip System Interlocks.

A CHANNEL CALIBRATION is performed every 24 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 plant specific setpoint methodology (Ref. 8). 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 24 months is based on the assumption of 24 month calibration intervals in the determination of the magnitude of equipment drift in the setpoint methodology.

With respect to RTDs, whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature detectors (RTD) sensors shall include an in-place qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. This is accomplished by an in-place cross calibration that compares the other sensing elements with the recently installed sensing element.

This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above 50% RTP. The CHANNEL CALIBRATION for the source range and intermediate range neutron detectors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. This Surveillance is not required for the NIS power range detectors for entry into MODE 2 or 1, and is not required for the NIS intermediate range detectors for entry into MODE 2, because the plant must be in at least MODE 2 to perform the test for the intermediate range detectors and MODE 1 for the power range detectors. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has

shown these components usually pass the Surveillance when performed on the 24 month Frequency.

SR 3.3.1.11

This SR is the performance of a TADOT of the Manual Reactor Trip, RCP Breaker Position, and the SI Input from ESFAS trip Functions. This TADOT is performed every 24 months. This test independently verifies the OPERABILITY of the undervoltage and shunt trip mechanisms for the Manual Reactor Trip Function for the Reactor Trip Breakers and Reactor Trip Bypass Breakers.

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.

SR 3.3.1.12

This SR is the performance of a TADOT for Turbine Trip Functions which is performed prior to reactor startup if it has not been performed within the last 31 days. This test shall verify OPERABILITY by actuation of the end devices.

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.

This SR is modified by a Note stating that 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 because portions of this test cannot be performed with the reactor at power.

SR 3.3.1.13

This SR is the performance of a COT of the RTS interlocks every 24 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.

REFERENCES

1. Atomic Industry Forum (AIF) GDC 14, Issued for comment July 10, 1967.
 2. 10 CFR 100.
 3. American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
 4. UFSAR, Chapter 7.
 5. UFSAR, Chapter 6.
 6. UFSAR, Chapter 15.
 7. IEEE-279-1971.
 8. RG&E Engineering Work Request (EWR) 5126, "Guidelines for Instrument Loop Performance Evaluation and Setpoint Verification," August 1992.
 9. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.
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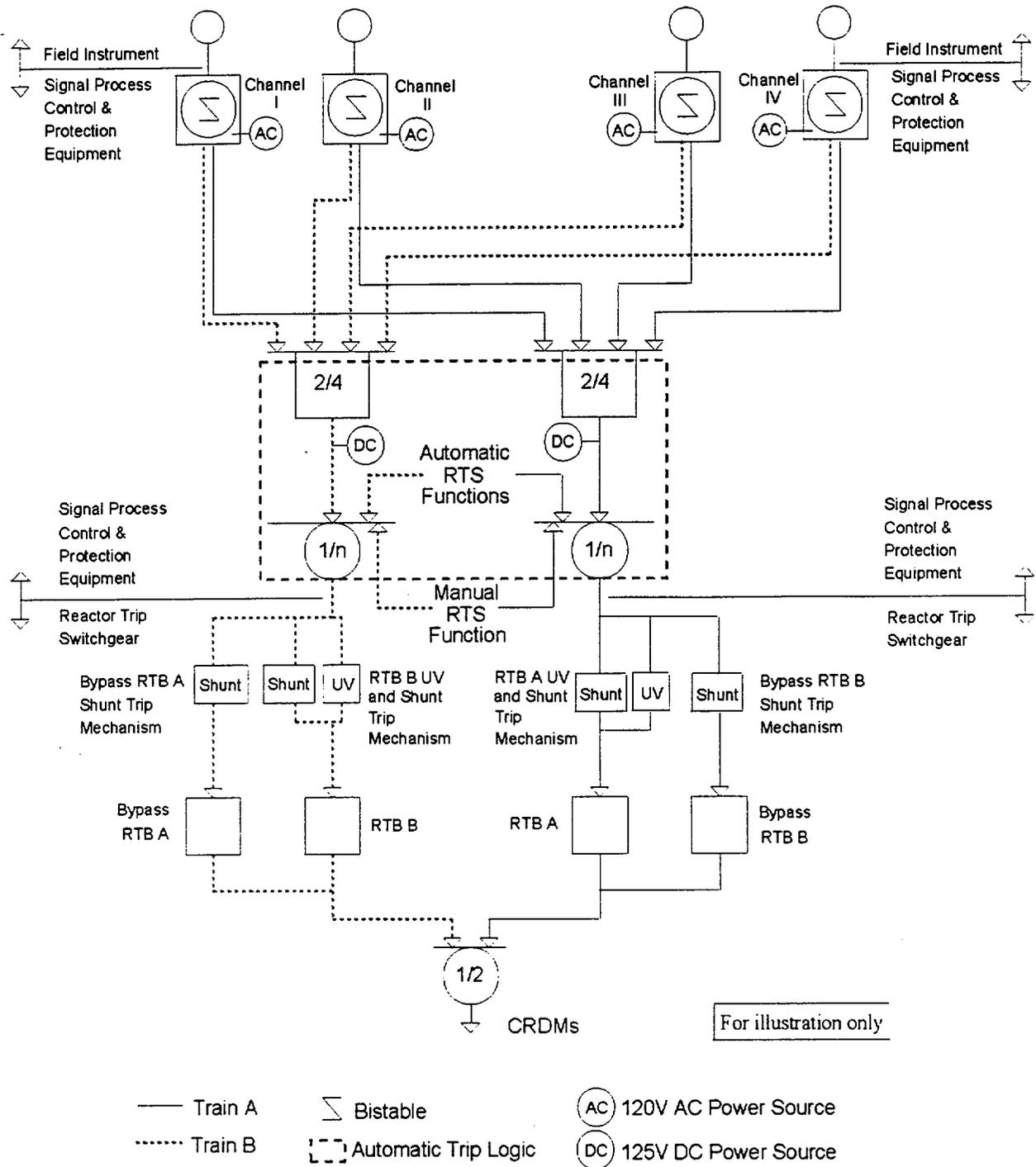


Figure B 3.3.1-1

B 3.3 INSTRUMENTATION

B 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

BASES

BACKGROUND Atomic Industrial Forum (AIF) GDC 15 (Ref. 1) requires that protection systems be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

The ESFAS initiates necessary safety systems, based on the values of selected plant parameters, to protect against violating core design limits and the Reactor Coolant System (RCS) pressure boundary, and to mitigate accidents.

The ESFAS instrumentation is segmented into two distinct but interconnected modules as described in UFSAR, Chapter 7 (Ref. 2):

- Field transmitters or process sensors; and
- Signal processing equipment.

These modules are discussed in more detail below.

Field Transmitters and Process Sensors

Field transmitters and process sensors provide a measurable electronic signal based on the physical characteristics of the parameter being measured. To meet the design demands for redundancy and reliability, two, three, and up to four field transmitters or sensors are used to measure required plant parameters. In many cases, field transmitters or sensors that input to the ESFAS are shared with the Reactor Trip System (RTS). To account for calibration tolerances and instrument drift, which is assumed to occur between calibrations, statistical allowances are provided. These statistical allowances provide the basis for determining acceptable "as left" and "as found" calibration values for each transmitter or sensor.

Signal Processing Equipment

The process control equipment provides signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with setpoints established by safety analyses. These setpoints are defined in UFSAR, Chapter 6 (Ref. 3), Chapter 7 (Ref. 2), and Chapter 15 (Ref. 4). If the measured value of a plant parameter exceeds the predetermined setpoint, an output from a bistable is forwarded to the logic relays.

Generally, three or four channels of process control equipment are used for the signal processing of plant parameters measured by the field transmitters and sensors. If a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are typically sufficient to provide the required reliability and redundancy. If one channel fails in a direction that would not result in a partial Function trip, the Function can still be accomplished with a two-out-of-two logic. If one channel fails in a direction that a partial Function trip occurs, a trip will not occur unless a second channel fails or trips in the remaining one-out-of-two logic.

If a parameter is used for input to the protection system and a control function, four channels with a two-out-of-four logic are typically sufficient to provide the required reliability and redundancy. This ensures that the circuit is able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Therefore, a single failure will neither cause nor prevent the protection function actuation. These requirements are described in IEEE-279-1971 (Ref. 5).

The actuation of ESF components is accomplished through master and slave relays. The protection system energizes the master relays appropriate for the condition of the plant. Each master relay then energizes one or more slave relays, which then cause actuation of the end devices.

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, SI-Pressurizer Pressure-Low is a primary actuation signal for small break 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 plant. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions may also serve as anticipatory actions to Functions that were credited in the accident analysis (Ref. 4).

This 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 LCO generally requires OPERABILITY of three or four 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. Two logic or manual initiation channels are required to ensure no single failure disables the ESFAS.

The LCO and Applicability of each ESFAS Function are provided in Table 3.3.2-1. Included on Table 3.3.2-1 are Allowable Values and Trip Setpoints for all applicable ESFAS Functions. Setpoints in accordance with the Allowable Value ensure that the consequences of Design Basis Accidents (DBAs) will be acceptable, providing the plant is operated within the LCOs, including any Required Actions that are in effect at the onset of the DBA and the equipment functions as designed.

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

The trip Setpoints used in the bistables are based on the analytical limits stated in References 2, 3, and 4. The selection of these trip Setpoints is such that adequate protection is provided when all sensor and processing time delays, calibration tolerances, instrumentation uncertainties, and instrument drift are taken into account. The Trip Setpoints specified in Table 3.3.2-1 are therefore conservatively adjusted with respect to The analytical limits (i.e., Allowable Values) used in the accident analysis. A detailed description of the methodology used to verify the adequacy of the existing Trip Setpoints, including their explicit uncertainties, is provided in Reference 6. If the measured setpoint exceeds the Trip Setpoint Value, the bistable is considered OPERABLE unless the Allowable Value as specified in plant procedures is exceeded. The Allowable Value specified in the plant procedures bounds that provided in Table 3.3.2-1 since the values in the table are typically those used in the accident analysis.

The Trip Setpoints and Allowable Values listed in Table 3.3.2-1 have been confirmed 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 processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

The required channels of ESFAS instrumentation provide plant protection in the event of any of the analyzed accidents. ESFAS protection functions provided in Table 3.3.2-1 are as follows:

1. Safety Injection

Safety Injection (SI) provides two primary functions:

1. Primary side water addition to ensure maintenance or recovery of reactor vessel water level (coverage of the active fuel for heat removal, clad integrity, and for limiting peak clad temperature to $< 2200^{\circ}\text{F}$); and
2. Boration to ensure recovery and maintenance of SDM ($k_{\text{eff}} < 1.0$).

These functions are necessary to mitigate the effects of high energy line breaks (HELBs) both inside and outside of containment. The SI signal is also used to initiate other Functions such as:

- Containment Isolation;
- Containment Ventilation Isolation;
- Reactor Trip;

- Feedwater Isolation; and
- Start of motor driven auxiliary feedwater (AFW) pumps.

These other functions ensure:

- Isolation of nonessential systems through containment penetrations;
- Trip of the reactor to limit power generation;
- Isolation of main feedwater (MFW) to limit secondary side mass losses; and
- Start of AFW to ensure secondary side cooling capability.

a. Safety Injection-Manual Initiation

This LCO requires one channel per train to be OPERABLE in MODES 1, 2, and 3. In these MODES, there is sufficient energy in the primary and secondary systems to warrant automatic initiation of ESF systems. The operator can initiate SI at any time by using either of two pushbuttons on the main control board. This action will cause actuation of all components with the exception of Containment Isolation and Containment Ventilation Isolation.

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

Each channel consists of one pushbutton and the interconnecting wiring to the actuation logic cabinet. Each pushbutton actuates both trains. This configuration does not allow testing at power.

This function is not required to be OPERABLE in MODES 4, 5, and 6 because there is adequate time for the operator to evaluate plant conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Plant pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of plant systems. Also, this Function is not required in MODE 4 since it does not actuate Containment Isolation or Containment Ventilation Isolation.

b. Safety Injection-Automatic Actuation Logic and Actuation Relays

This LCO requires two trains to be OPERABLE in MODES 1, 2, 3, and 4. In these MODES, there is sufficient energy in the primary and secondary systems to warrant automatic initiation of ESF systems. Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

This Function is not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate plant conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Plant pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of plant systems.

c. Safety Injection-Containment Pressure-High

This signal provides protection against the following accidents:

- SLB inside containment;
- LOCA; and
- Feed line break inside containment.

Containment Pressure-High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with a two-out-of-three logic. PT-945, PT-947, and PT-949 are the three channels required for this function. The transmitters and electronics are located outside of containment with the sensing lines passing through containment penetrations to sense the containment atmosphere in three different locations.

Thus, the high pressure Function will not experience any adverse environmental conditions and The Trip Setpoint reflects only steady state instrument uncertainties.

Containment Pressure-High must be OPERABLE in MODES 1, 2, 3, and 4 because there is sufficient energy in the primary and secondary systems to pressurize the containment following a pipe break. In MODES 5 and 6, Containment Pressure-High is not required to be OPERABLE because there is insufficient energy in the primary or secondary systems to pressurize the containment.

d. Safety Injection-Pressurizer Pressure-Low

This signal provides protection against the following accidents:

- Inadvertent opening of a steam generator (SG) atmospheric relief or safety valve;
- SLB;
- Rod cluster control assembly ejection accidents (rod ejection);
- Inadvertent opening of a pressurizer relief or safety valve;
- LOCAs; and
- SG Tube Rupture.

Since there are dedicated protection and control channels, only three protection channels are necessary to satisfy the protective requirements. PT-429, PT-430, and PT-431 are the three channels required for this function.

The transmitters are located inside containment, with the taps in the vapor space region of the pressurizer, and thus possibly experiencing adverse environmental conditions (LOCA, SLB inside containment, rod ejection). Therefore, the Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 (above the Pressurizer Pressure interlock) to mitigate the consequences of an HELB inside containment. This signal may be manually blocked by the operator below the interlock setpoint. Automatic SI actuation below this interlock setpoint is performed by the Containment Pressure-High signal.

This function is not required to be OPERABLE in MODE 3 below the Pressurizer Pressure interlock setpoint. Other ESF functions are used to detect accident conditions and actuate the ESF systems in this MODE. In MODES 4, 5, and 6, this Function is not needed for accident detection and mitigation.

e. Safety Injection-Steam Line Pressure-Low

Steam Line Pressure-Low provides protection against the following accidents:

- SLB;
- Feed line break; and
- Inadvertent opening of an SG atmospheric relief or an SG safety valve.

Steam line pressure transmitters provide control input, but the control function cannot initiate events that the Function acts to mitigate. Thus, three OPERABLE channels on each steam line are sufficient to satisfy the protective requirements with a two-out-of-three logic on each steam line. PT-468, PT-469, and PT-482 are the three channels required for steam line A. PT-478, PT-479, and PT-483 are the three channels required for steam line B. Each steam line is considered a separate function for the purpose of this LCO.

With the transmitters located in the Intermediate Building, it is possible for them to experience adverse environmental conditions during a secondary side break. Therefore, the Trip Setpoint reflects both steady state and adverse environmental instrument uncertainties.

Steam Line Pressure-Low must be OPERABLE in MODES 1, 2, and 3 (above The Pressurizer Pressure interlock) when a secondary side break or stuck open SG atmospheric relief or safety valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the interlock setpoint. Below the interlock setpoint, a feed line break is not a concern. This Function is not required to be OPERABLE in MODE 4, 5, or 6 because there is insufficient energy in the secondary side of the plant to cause an accident.

2. Containment Spray (CS)

CS provides three primary functions:

1. Lowers containment pressure and temperature after an HELB in containment;
2. Reduces the amount of radioactive iodine in the containment atmosphere; and
3. Adjusts the pH of the water in containment sump B after a large break LOCA.

These functions are necessary to:

- Ensure the pressure boundary integrity of the containment structure;
- Limit the release of radioactive iodine to the environment in the event of a failure of the containment structure; and
- Minimize corrosion of the components and systems inside containment following a LOCA.

CS is actuated manually or by Containment Pressure-High High. The CS actuation signal starts the CS pumps and aligns the discharge of the pumps to the CS nozzle headers in the upper levels of containment. Water is initially drawn from the RWST by the CS pumps and mixed with a sodium hydroxide solution from the spray additive tank. During the recirculation phase of accident recovery, the spray pump suctions are manually shifted to containment sump B if continued CS is required.

a. CS-Manual Initiation

The operator can initiate CS at any time from the control room by simultaneously depressing two CS actuation pushbuttons. Because an inadvertent actuation of CS could have serious consequences, two pushbuttons must be simultaneously depressed to initiate both trains of CS. Therefore, the inoperability of either pushbutton fails both trains of manual initiation.

Manual initiation of CS must be OPERABLE in MODES 1, 2, 3, and 4 because a DBA could cause a release of radioactive material to containment and an increase in containment temperature and pressure requiring the operation of the CS System.

In MODES 5 and 6, this function is not required to be OPERABLE because the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. In MODES 5 and 6, there is also adequate time for the operators to evaluate plant conditions and respond to mitigate the consequences of abnormal conditions by manually starting individual components.

b. CS-Automatic Actuation Logic and Actuation Relays

Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

Automatic initiation of CS must be OPERABLE in MODES 1, 2, 3, and 4 because a DBA could cause a release of radioactive material to containment and an increase in containment temperature and pressure requiring the operation of the CS System.

In MODES 5 and 6, this Function is not required to be OPERABLE because the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. In MODES 5 and 6, there is also adequate time for the operators to evaluate plant conditions and respond to mitigate the consequences of abnormal conditions by manually starting individual components.

c. CS-Containment Pressure-High High

This signal provides protection against a LOCA or an SLB inside containment. The transmitters are located outside of containment with the sensing lines passing through containment penetrations to sense the containment atmosphere in three different locations. The transmitters and electronics are located outside of containment. Thus, they will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

This is the only ESFAS Function that requires the bistable output to energize to perform its required action. It is not desirable to have a loss of power actuate CS, since the consequences of an inadvertent actuation of CS could be serious.

The Containment Pressure-High High instrument function consists of two sets with three channels in each set. One set is comprised of PT-945, PT-947, and PT-949. The second set is comprised of PT-946, PT-948, and PT-950. Each set is a two-out-of-three logic where the outputs are combined so that both sets tripped initiates CS. Each set is considered a separate function for the purposes of this LCO. Since containment pressure is not used for control, this arrangement exceeds the minimum redundancy requirements. Additional redundancy is warranted because this Function is energize to trip. Containment Pressure-High High must be OPERABLE in MODES 1, 2, 3 and 4 because a DBA could cause a release of radioactive material to containment and an increase in containment temperature and pressure requiring the operation of the CS System.

In MODES 5 and 6, this Function is not required to be OPERABLE because the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. In MODES 5 and 6, there is also adequate time for the operators to evaluate plant conditions and respond to mitigate the consequences of abnormal conditions by manually starting individual components.

3. Containment Isolation

Containment Isolation provides isolation of the containment atmosphere, and selected process systems that penetrate containment, from the environment. This Function is necessary to prevent or limit the release of radioactivity to the environment in the event of a LOCA.

Containment Isolation signals isolate all automatically isolatable process lines, except feedwater lines, main steam lines, and component cooling water (CCW). The main feedwater and steam lines are isolated by other functions since forced circulation cooling using the reactor coolant pumps (RCPs) and SGs is the preferred (but not required) method of decay heat removal. Since CCW is required to support RCP operation, not isolating CCW enhances plant safety by allowing operators to use forced RCS circulation to cool the plant. Isolating CCW may require the use of feed and bleed cooling, which could prove more difficult to control.

a. Containment Isolation-Manual Initiation

Manual Containment Isolation is actuated by either of two pushbuttons on the main control board. Either pushbutton actuates both trains. Manual initiation of Containment Isolation also actuates Containment Ventilation Isolation.

Manual initiation of Containment Isolation must be OPERABLE in MODES 1, 2, 3 and 4, because there is a potential for an accident to occur.

In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Containment Isolation. There also is adequate time for the operator to evaluate plant conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

b. Containment Isolation-Automatic Actuation Logic and Actuation Relays

Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

Automatic initiation of Containment Isolation must be OPERABLE in MODES 1, 2, 3 and 4, because there is a potential for an accident to occur.

In MODES 5 and 6, there is insufficient energy in the primary or secondary systems to pressurize the containment to require Containment Isolation. There also is adequate time for the operator to evaluate plant conditions and manually actuate individual isolation valves in response to abnormal or accident conditions.

c. Containment Isolation-Safety Injection

Containment Isolation is also initiated by all Functions that automatically initiate SI. The Containment Isolation requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all applicable initiating Functions and requirements.

4. Steam Line Isolation

Isolation of the main steam lines provides protection in the event of an SLB inside or outside containment. Closure of the main steam isolation valves (MSIVs) and their associated non-return check valves limits the accident to the blowdown from only the affected SG. For a SLB downstream of the MSIVs, closure of the MSIVs terminates the accident as soon as the steam lines depressurize. Steam Line Isolation also mitigates the effects of a feed line break and ensures a source of steam for the turbine driven AFW pump during a feed line break.

a. Steam Line Isolation-Manual Initiation

Manual initiation of Steam Line Isolation can be accomplished from the control room. There are two actuation devices (one pushbutton and one switch) on the main control board for each MSIV. Each device can initiate action to immediately close its respective MSIV. The LCO requires one channel (device) per loop to be OPERABLE. Each loop is not considered a separate function since there is only one required per loop.

Manual initiation of steam line isolation must be OPERABLE in MODES 1, 2, and 3 because a secondary side break or stuck open valve could result in rapid depressurization of the steam lines. This could result in the release of significant quantities of energy and cause a cooldown of the primary system. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless both MSIVs are closed and de-activated. In MODES 4, 5, and 6, the steam line isolation function is not required to be OPERABLE because there is insufficient energy in the RCS and SGs to experience an SLB or other accident releasing significant quantities of energy.

b. Steam Line Isolation-Automatic Actuation Logic and Actuation Relays

Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

Automatic initiation of steam line isolation must be OPERABLE in MODES 1, 2, and 3 because a secondary side break or stuck open valve could result in rapid depressurization of the steam lines. This could result in the release of significant quantities of energy and cause a cooldown of the primary system. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless both MSIVs are closed and de-activated. In MODES 4, 5, and 6, the steam line isolation function is not required to be OPERABLE because there is insufficient energy in the RCS and SGs to experience an SLB or other accident releasing significant quantities of energy.

c. Steam Line Isolation-Containment Pressure-High High

This Function actuates closure of both MSIVs in the event of a LOCA or an SLB inside containment to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. The transmitters are located outside containment with the sensing lines passing through containment penetrations to sense the containment atmosphere in three different locations. Thus, they will not experience any adverse environmental conditions, and the Trip Setpoint reflects only steady state instrument uncertainties. Containment Pressure-High High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy protective requirements with two-out-of-three logic. PT-946, PT-948, and PT-950 are the three channels required for this function.

Containment Pressure-High High must be OPERABLE in MODES 1, 2, and 3, because there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe break. This would cause a significant increase in the containment pressure, thus allowing detection and closure of the MSIVs. The steam line isolation Function must be OPERABLE in MODES 2 and 3 unless both MSIVs are closed and de-activated. In MODES 4, 5, and 6 the steam line isolation Function is not required to be OPERABLE because there is not enough energy in the primary and secondary sides to pressurize the containment to the Containment Pressure-High High setpoint.

d. Steam Line Isolation-High Steam Flow Coincident With Safety Injection and Coincident With T_{avg} -Low

This Function provides closure of the MSIVs during an SLB or inadvertent opening of multiple SG atmospheric relief or safety valves to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment.

The specified Allowable Value is based on steam line breaks occurring from no load conditions (1005 psig). Specifically, steam line breaks which result in a > 10% RTP step change (0.66E6 1bm/hr) are considered. The steam flow signal to this function's bistables are not pressure compensated (i.e., only the main control board indicators are compensated). However, the high steam flow bistable setpoint is determined from the expected flow transmitter differential pressure under steam conditions of 0.66E6 1bm/hr at 1005 psig. Steam breaks which result in higher flowrates or lower pressure generate larger differential pressures such that the high

steam flow bistables would be tripped. Steam line breaks which result in a < 10% RTP step change can be manually isolated by operators. The high steam flow bistables are OPERABLE if they are placed in the tripped condition since the specified Trip Setpoint and Allowable Value are met. However, all applicable surveillances related to the tripped channel must continue to be performed and met.

Two steam line flow channels per steam line are required to be OPERABLE for this Function. These are combined in a one-out-of-two logic to indicate high steam flow in one steam line. FT-464 and FT-465 are the two channels required for steam line A. FT-474 and FT-475 are the two channels required for steam line B. Each steam line is considered a separate function for the purpose of this LCO. The steam flow transmitters provide control inputs, but the control function cannot initiate events that the function acts to mitigate. Therefore, additional channels are not required to address control protection interaction issues. The one-out-of-two configuration allows online testing because trip of one high steam flow channel is not sufficient to cause initiation.

With the transmitters (d/p cells) located inside containment, it is possible for them to experience adverse environmental conditions during an SLB event. Therefore, the trip Setpoints reflect both steady state and adverse environmental instrument uncertainties.

The main steam line isolates only if the high steam flow signal occurs coincident with an SI and low RCS average temperature. The Main Steam Line Isolation Function requirements for the SI Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all applicable initiating functions and requirements.

Two channels of T_{avg} per loop are required to be OPERABLE for this Function. TC-401 and TC-402 are the two channels required for RCS loop A. TC-403 and TC-404 are the two channels required for RCS loop B. Each loop is considered a separate Function for the purpose of this LCO. The T_{avg} channels are combined in a logic such that any two of the four T_{avg} channels tripped in conjunction with SI and one of the two high steam line flow channels tripped causes isolation of the steam line associated with the tripped steam line flow channels. The accidents that this Function protects against cause reduction of T_{avg} in the entire primary system. Therefore, the provision of two OPERABLE channels per

loop in a two-out-of-four configuration ensures no single failure disables the T_{avg} -Low Function. The T_{avg} channels provide control inputs, but the control function cannot initiate events that the Function acts to mitigate. Therefore, additional channels are not required to address control protection interaction issues.

This Function must be OPERABLE in MODES 1, 2, and 3 when a secondary side break or stuck open valve could result in rapid depressurization of the steam lines. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless both MSIVs are closed and de-activated. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the plant to have an accident.

e. Steam Line Isolation-High High Steam Flow Coincident With Safety Injection

This Function provides closure of the MSIVs during a large steam line break to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment.

The specified allowable Value is based on steamline breaks occurring from full power steam conditions which result in $\geq 109\%$ RTP steam flow. The steam flow signal to this function's bistables are not pressure compensated (i.e., only the main control board indicators are compensated). However, the high-high steam flow bistable setpoint is determined from the expected flow transmitter differential pressure under steam conditions of $3.7E6$ lbm/hr at 755 psig. Steam breaks which result in higher flowrates or lower pressure generate larger differential pressures such that the high-high steam flow bistables would be tripped.

Two steam line flow channels per steam line are required to be OPERABLE for this Function. These are combined in a one-out-of-two logic to indicate high-high steam flow in one steam line. FT-464 and FT-465 are the two channels required for steam line A. FT-474 and FT-475 are the two channels required for steam line B. Each steam line is considered a separate function for the purpose of this LCO. The steam flow transmitters provide control inputs, but the control function cannot initiate events that the Function acts to mitigate. Therefore, additional channels are not required to address control protection interaction issues.

The main steam lines isolate only if the high-high steam flow signal occurs coincident with an SI signal. Steamline isolation occurs only for the steam line associated with the tripped steam flow channels. The Main Steam Line Isolation Function requirements for the SI Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all applicable initiating functions and requirements.

This Function must be OPERABLE in MODES 1, 2, and 3 because a secondary side break or stuck open valve could result in rapid depressurization of the steam lines. The Steam Line Isolation Function is required to be OPERABLE in MODES 2 and 3 unless both MSIV's are closed and de-activated. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the plant to have an accident.

5. Feedwater Isolation

The primary function of the Feedwater Isolation signals is to prevent and mitigate the effects of highwater level in the SGs which could cause carryover of water into the steam lines and result in excessive cooldown of the primary system. The SG high water level is due to excessive feedwater flows.

This Function is actuated by either a SG Water Level-High or an SI signal. The Function provides feedwater isolation by closing the Main Feedwater Regulating Valves (MFRVs) and the associated bypass valves. In addition, on an SI signal, the AFW System is automatically started, and the MFW pump breakers are opened which closes the MFW pump discharge valves. The SI signal was discussed previously.

a. Feedwater Isolation-Automatic Actuation Logic and Actuation Relays

Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

Automatic initiation must be OPERABLE in MODES 1, 2, and 3. The Feedwater Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all MFRVs and associated bypass valves are closed and de-activated or isolated by a closed manual valve. In MODES 4, 5, and 6, the MFW System and the turbine generator are not in service and this Function is not required to be OPERABLE.

b. Feedwater Isolation-Steam Generator Water Level-High

The Steam Generator Water Level-High Function must be OPERABLE in MODES 1, 2, and 3. The Feedwater Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all MFRVs and associated bypass valves are closed and de-activated or isolated by a closed manual valve. In MODES 4, 5, and 6, the MFW System and the turbine generator are not in service and this Function is not required to be OPERABLE.

This signal provides protection against excessive feedwater flow. The ESFAS SG water level instruments have dedicated protection and control channels, only three protection channels are necessary to satisfy the protective requirements. LT-461, LT-462, and LT-463 are the three channels required for SG A. LT-471, LT-472, and LT-473 are the three channels required for SG B. Each SG is considered a separate Function for the purpose of this LCO. The Allowable Value for SG Water Level-High is a percent of narrow range instrument span. The trip Setpoint is similarly calculated.

c. Feedwater Isolation-Safety Injection

The Safety Injection Function must be OPERABLE in MODES 1, 2, and 3. The Feedwater Isolation Function is required to be OPERABLE in MODES 2 and 3 unless all MFRVs and associated bypass valves are closed and de-activated or isolated by a closed manual valve. In MODES 4, 5, and 6, the MFW System and the turbine generator are not in service and this Function is not required to be OPERABLE.

Feedwater Isolation is also initiated by all Functions that initiate SI. The Feedwater Isolation Function requirements for these Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead Function 1, SI, is referenced for all initiating functions and requirements.

6. Auxiliary Feedwater

The AFW System is designed to provide a secondary side heat sink for the reactor in the event that the MFW System is not available. The preferred system has two motor driven pumps and a turbine driven pump, making it available during normal plant operation, during a loss of AC power, a loss of MFW, and during a Feedwater System pipe break (depending on break location). A Standby AFW (SAFW) System is also available in the event the preferred system is unavailable. The normal source of water for the AFW System is

the condensate storage tank (CST) which is not safety related. Upon a low level in the CST the operators can manually realign the pump suctions to the Service Water (SW) System which is the safety related water source. The SW System also is the safety related water source for the SAFW System. The AFW System is aligned so that upon a pump start, flow is initiated to the respective SGs immediately while the SAFW System is only manually initiated and aligned.

a. Auxiliary Feedwater-Manual Initiation

The operator can initiate AFW or SAFW at any time by using control switches on the Main Control board (one switch for each pump in each system). This action will cause actuation of their respective pump.

The LCO for the Manual Initiation Function ensures the proper amount of redundancy is maintained to ensure the operator has manual AFW and SAFW initiation capability.

The LCO requires one channel per pump in each system to be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. In MODE 4, AFW actuation is not required to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation. This Function is not required to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink.

b. Auxiliary Feedwater-Automatic Actuation Logic and Actuation Relays

Actuation logic consists of all circuitry housed within the actuation subsystems, including the initiating relay contacts responsible for actuating the ESF equipment.

Automatic initiation of Auxiliary Feedwater must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. In MODE 4, AFW actuation is not required to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation. This Function is not required to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink.

c. Auxiliary Feedwater-Steam Generator Water Level-Low Low

SG Water Level-Low Low must be OPERABLE in MODES 1, 2, and 3 to provide protection against a loss of heat sink. A feed line break, inside or outside of containment, or a loss of MFW, would result in a loss of SG water level. SG Water Level-Low Low in either SG will cause both motor driven AFW pumps to start. The system is aligned so that upon a start of the pump, water immediately begins to flow to the SGs. SG Water Level-Low Low in both SGs will cause the turbine driven pump to start. In MODE 4, AFW actuation is not required to be OPERABLE because either AFW or RHR will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation. This Function is not required to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink.

LT-461, LT-462, and LT-463 are the three channels required for SG A. LT-471, LT-472, and LT-473 are the three channels required for SG B. Each SG is considered a separate Function for the purpose of this LCO. The Allowable Value for SG Water Level - Low Low is a percent of narrow range instrument span. The Trip Setpoint is similarly calculated.

With the transmitters (d/p cells) located inside containment and thus possibly experiencing adverse environmental conditions (feed line break), the Trip Setpoint reflects the inclusion of both steady state and adverse environmental instrument uncertainties.

d. Auxiliary Feedwater-Safety Injection

The SI function must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. In MODE 4, AFW actuation is not required to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation. This Function is not required to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink.

An SI signal starts the motor driven and turbine driven AFW pumps. The AFW initiation functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all applicable initiating functions and requirements.

e. Auxiliary Feedwater-Undervoltage-Bus 11A and 11B

The Undervoltage-Bus 11A and 11B Function must be OPERABLE in MODES 1, 2, and 3 to ensure that the SGs remain the heat sink for the reactor. In MODE 4, AFW actuation is not required to be OPERABLE because either AFW or RHR will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation. This Function is not required to be OPERABLE in MODES 5 and 6 because there is not enough heat being generated in the reactor to require the SGs as a heat sink.

A loss of power to 4160 V Bus 11A and 11B will be accompanied by a loss of power to both MFW pumps and the subsequent need for some method of decay heat removal. The loss of offsite power is detected by a voltage drop on each bus. Loss of power to both buses will start the turbine driven AFW pump to ensure that at least one SG contains enough water to serve as the heat sink for reactor decay heat and sensible heat removal following the reactor trip. Each bus is considered a separate Function for the purpose of this LCO.

f. Auxiliary Feedwater-Trip Of Both Main Feedwater Pumps

A trip of both MFW pumps is an indication of a loss of MFW and the subsequent need for some method of decay heat and sensible heat removal. The MFW pumps are equipped with a breaker position sensing device. An open supply breaker indicates that the pump is not running. Two OPERABLE channels per MFW pump satisfy redundancy requirements with two-out-of-two logic. Each MFW pump is considered a Separate Function for the purpose of this LCO. A trip of both MFW pumps starts both motor driven AFW (MDAFW) pumps to ensure that at least one SG is available with water to act as the heat sink for the reactor. However, this actuation of the MDAFW pumps is not credited in the mitigation of any accident.

This Function must be OPERABLE in MODE 1. This ensures that at least one SG is provided with water to serve as the heat sink to remove reactor decay heat and sensible heat in the event of an accident. In MODES 2, 3, 4, 5, and 6 the MFW pumps may not be in operation, and thus pump trip is not indicative of a condition requiring automatic AFW initiation.

ACTIONS

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.2-1.

In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected. As shown on Figure B 3.3.2-1, the ESFAS is comprised of multiple interconnected modules and components. For the purpose of this LCO, a channel is defined as including all related components from the field instrument to the Automatic Actuation Logic. Therefore, a channel may be inoperable due to the failure of a field instrument, loss of 120 VAC instrument bus power or a bistable failure which affects one or both ESFAS trains. The only exception to this are the Manual ESFAS and Automatic Actuation Logic Functions which are defined strictly on a train basis. The Automatic Actuation Logic consists of all circuitry housed within the actuation subsystem, including the master relays, slave relays, and initiating relay contacts responsible for activating the ESF equipment.

A.1

Condition A applies to all ESFAS protection functions.

Condition A addresses the situation where one channel or train for one or more Functions are inoperable. The Required Action is to refer to Table 3.3.2-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

When the number of inoperable channels in an ESFAS Function exceed those specified in all related Conditions associated with an ESFAS Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if the ESFAS function is applicable in the current MODE of operation.

B.1

Condition B applies to the AFW-Trip of Both MFW Pumps ESFAS Function. If a channel is inoperable, 48 hours is allowed to return it to OPERABLE status. The specified Completion Time of 48 hours is reasonable considering the nature of this Function, the available redundancy, and the low probability of an event occurring during this interval.

C.1

If the Required Action and Completion Time of Condition B is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. The allowed Completion time is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

D.1

Condition D applies to the following ESFAS Functions:

- Manual Initiation of SI;
- Manual Initiation of Steam Line Isolation; and
- AFW-Undervoltage-Bus 11A and 11B.

If a channel is inoperable, 48 hours is allowed to restore it to OPERABLE status. The specified Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE for each manual initiation Function, additional AFW actuation channels available besides the Undervoltage-Bus 11A and 11B AFW Initiation Function, and the low probability of an event occurring during this interval.

E.1

Condition E applies to the automatic actuation logic and actuation relays for the following ESFAS Functions:

- Steam Line Isolation;
- Feedwater Isolation; and
- AFW.

Condition E addresses the train orientation of the protection system and the master and slave relays. If one train is inoperable, a Completion Time of 6 hours is allowed to restore the train to OPERABLE status. This Completion Time is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this time interval. The Completion Time of 6 hours is consistent with Reference 7.

F.1

Condition F applies to the following Functions:

- Steam Line Isolation-Containment Pressure-High High;
- Steam Line Isolation-High Steam Flow Coincident With Safety Injection and Coincident With T_{avg} -Low;
- Steam Line Isolation-High-High Steam Flow Coincident With Safety Injection;
- Feedwater Isolation-SG Water Level-High; and
- AFW-SG Water Level-Low Low.

Condition F applies to Functions that typically operate on two-out-of-three logic. Therefore, failure of one channel places the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-two configuration that satisfies redundancy requirements.

If one channel is inoperable, a Completion Time of 6 hours is allowed to restore the channel to OPERABLE status or to place it in the tripped condition. Placing the channel in the Tripped condition conservatively compensates for the inoperability, restores capability to accommodate a single failure, and allows operation to continue.

The Required Actions are modified by a Note that allows the inoperable channel to be bypassed for up to 4 hours for surveillance testing of other channels. This 4 hours applies to each of the remaining OPERABLE channels.

The Completion Time of 6 hours allowed to restore the channel to OPERABLE status or to place the inoperable channel in the tripped condition, and the 4 hours allowed for testing, are justified in Reference 7.

G.1

If the Required Actions and Completion Times of Conditions D, E, or F are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

H.1

Condition H applies to the following ESFAS functions:

- Manual Initiation of CS; and
- Manual Initiation of Containment Isolation.

If a channel is inoperable, 48 hours is allowed to restore it to OPERABLE status. The specified Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE for each Function (except for CS) and the low probability of an event occurring during this interval.

I.1

Condition I applies to the automatic actuation logic and actuation relays for the following Functions:

- SI;
- CS; and
- Containment Isolation.

Condition I addresses the train orientation of the protection system and the master and slave relays. If one train is inoperable, a Completion Time of 6 hours is allowed to restore the train to OPERABLE status. This Completion Time is reasonable considering that there is another train OPERABLE, and the low probability of an event occurring during this interval. The Completion Time of 6 hours is consistent with Reference 7.

J.1

Condition J applies to the following Functions:

- SI-Containment Pressure-High; and
- CS-Containment Pressure-High High.

Condition J applies to Functions that operate on a two-out-of-three logic (for CS-Containment Pressure-High High there are two sets of this logic). Therefore, failure of one channel places the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-two configuration that satisfies redundancy requirements.

If one channel is inoperable, a Completion Time of 6 hours is allowed to restore the channel to OPERABLE status or place it in the tripped condition. Placing the channel in the tripped condition conservatively compensates for the inoperability, restores capability to accommodate a single failure, and allows operation to continue.

The Required Action is modified by a Note that allows the inoperable channel to be bypassed for up to 4 hours for surveillance testing of other channels. The 4 hours applies to each of the remaining OPERABLE channels.

The Completion Time of 6 hours to restore the inoperable channel or place it in trip, and the 4 hours allowed for surveillance testing is justified in Reference 7.

K.1

If the Required Actions and Completion Times of Conditions H, I, or J are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant 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 plant conditions from full power conditions in an orderly manner and without challenging plant systems.

L.1

Condition L applies to the following Functions:

- SI-Pressurizer Pressure-Low; and
- SI-Steam Line Pressure-Low.

Condition L applies to Functions that operate on a two-out-of-three logic. Therefore, failure of one channel places the Function in a two-out-of-two configuration. One channel must be tripped to place the Function in a one-out-of-two configuration that satisfies redundancy requirements.

If one channel is inoperable, a Completion Time of 6 hours is allowed to restore the channel to OPERABLE status or place it in the tripped condition. Placing the channel in the tripped condition conservatively compensates for the inoperability, restores capability to accommodate a single failure, and allows operation to continue.

The Required Action is modified by a Note that allows the inoperable channel to be bypassed for up to 4 hours for surveillance testing of other channels. The 4 hours applies to each of the remaining OPERABLE channels.

The Completion Time of 6 hours to restore the inoperable channel or place it in trip, and the 4 hours allowed for surveillance testing is justified in Reference 7.

M.1

If the Required Actions and Completion Times of Condition L are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and pressurizer pressure reduced to < 2000 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

N.1

Condition N applies if an AFW Manual Initiation channel is inoperable. If a manual initiation switch is inoperable, the associated AFW or SAFW pump must be declared inoperable and the applicable Conditions of LCO 3.7.5, "Auxiliary Feedwater (AFW) System" must be entered immediately. Each AFW manual initiation switch controls one AFW or SAFW pump. Declaring the associated pump inoperable ensures that appropriate action is taken in LCO 3.7.5 based on the number and type of pumps involved.

**SURVEILLANCE
REQUIREMENTS**

The SRs for each ESFAS Function are identified by the SRs column of Table 3.3.2-1. Each channel of process protection supplies both trains of the ESFAS. When testing Channel 1, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel 2, Channel 3, and Channel 4 (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

A Note has been added to the SR Table to clarify that Table 3.3.2-1 determines which SRs apply to which ESFAS Functions.

SR 3.3.2.1

This SR is the performance of a CHANNEL CHECK for the following ESFAS Functions:

- SI-Containment Pressure-High;
- SI-Pressurizer Pressure-Low;
- SI-Steam Line Pressure-Low;

- CS-Containment Pressure-High High;
- Steam Line Isolation-Containment Pressure-High High;
- Steam Line Isolation-High Steam Flow Coincident with SI and T_{avg} -Low;
- Steam Line Isolation-High-High Steam Flow Coincident with SI;
- Feedwater Isolation-SG Water Level-High; and
- AFW-SG Water Level-Low Low.

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 instrument channels could be an indication of excessive instrument drift in one of the channels or of more serious instrument conditions. A CHANNEL CHECK will detect gross channel failure; thus, it is a verification the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

CHANNEL CHECK acceptance criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

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

SR 3.3.2.2

This SR is the performance of a COT every 92 days for the following ESFAS functions:

- SI-Containment Pressure-High;
- SI-Pressurizer Pressure-Low;
- SI-Steam Line Pressure-Low;
- CS-Containment Pressure-High High;

- Steam Line Isolation-Containment Pressure-High High;
- Steam Line Isolation-High Steam Flow Coincident with SI and T_{avg} -Low;
- Steam Line Isolation-High-High Steam Flow Coincident with SI;
- Feedwater Isolation-SG Water Level-High; and
- AFW-SG Water Level-Low Low.

A COT is performed on each required channel to ensure the entire channel will perform the intended Function. Setpoints must be found to be within the Allowable Values specified in Table 3.3.2-1 and established plant procedures. The "as left" values must be consistent with the drift allowance used in the setpoint methodology.

The Frequency of 92 days is consistent with in Reference 7. The Frequency is adequate based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.2.3

This SR is the performance of a TADOT every 92 days. This test is a check of the AFW-Undervoltage-Bus 11A and 11B Function.

The test includes trip devices that provide actuation signals directly to the protection system. The SR is modified by a Note that excludes verification of setpoints for relays. Relay setpoints require elaborate bench calibration and are verified during CHANNEL CALIBRATION. The Frequency of 92 days is adequate based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.2.4

This SR is the performance of a TADOT every 24 months. This test is a check of the SI, CS, Containment Isolation, Steam Line Isolation, and AFW Manual Initiations, and the AFW-Trip of Both MFW Pumps Functions. Each Function is tested up to, and including, the master transfer relay coils. The Frequency of 24 months is based on industry operating experience and is consistent with the typical refueling cycle. The SR is modified by a Note that excludes verification of setpoints during the TADOT. The Manual Initiations, and AFW-Trip of Both MFW Pumps Functions have no associated setpoints.

SR 3.3.2.5

This SR is the performance of a CHANNEL CALIBRATION every 24 months of the following ESFAS Functions:

- SI-Containment Pressure-High;
- SI-Pressurizer Pressure-Low;
- SI-Steam Line Pressure-Low;
- CS-Containment Pressure-High High;
- Steam Line Isolation-Containment Pressure-High High;
- Steam Line Isolation-High Steam Flow Coincident with SI and T_{avg} Low;
- Steam Line Isolation-High-High Steam Flow Coincident with SI;
- Feedwater Isolation-SG Water Level-High;
- AFW-SG Water Level-Low Low; and
- AFW-Undervoltage-Bus 11A and 11B.

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 plant specific setpoint methodology. The "as left" values must be consistent with the drift allowance used in the setpoint methodology.

The Frequency of 24 months is based on the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

SR 3.3.2.6

This SR ensures the SI-Pressurizer Pressure-Low and SI-Steam Line Pressure-Low Functions are not bypassed when pressurizer pressure > 2000 psig while in MODES 1, 2, and 3. Periodic testing of the pressurizer pressure channels is required to verify the setpoint to be less than or equal to the limit.

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 (Ref. 6). The setpoint shall be left set consistent with the assumptions of the current plant specific setpoint methodology.

If the pressurizer pressure interlock setpoint is nonconservative, then the Pressurizer Pressure-Low and Steam Line Pressure-Low Functions are considered inoperable. Alternatively, the pressurizer pressure interlock can be placed in the conservative condition (nonbypassed). If placed in the nonbypassed condition, the SR is met and the Pressurizer Pressure-Low and Steam Line Pressure-Low Functions would not be considered inoperable.

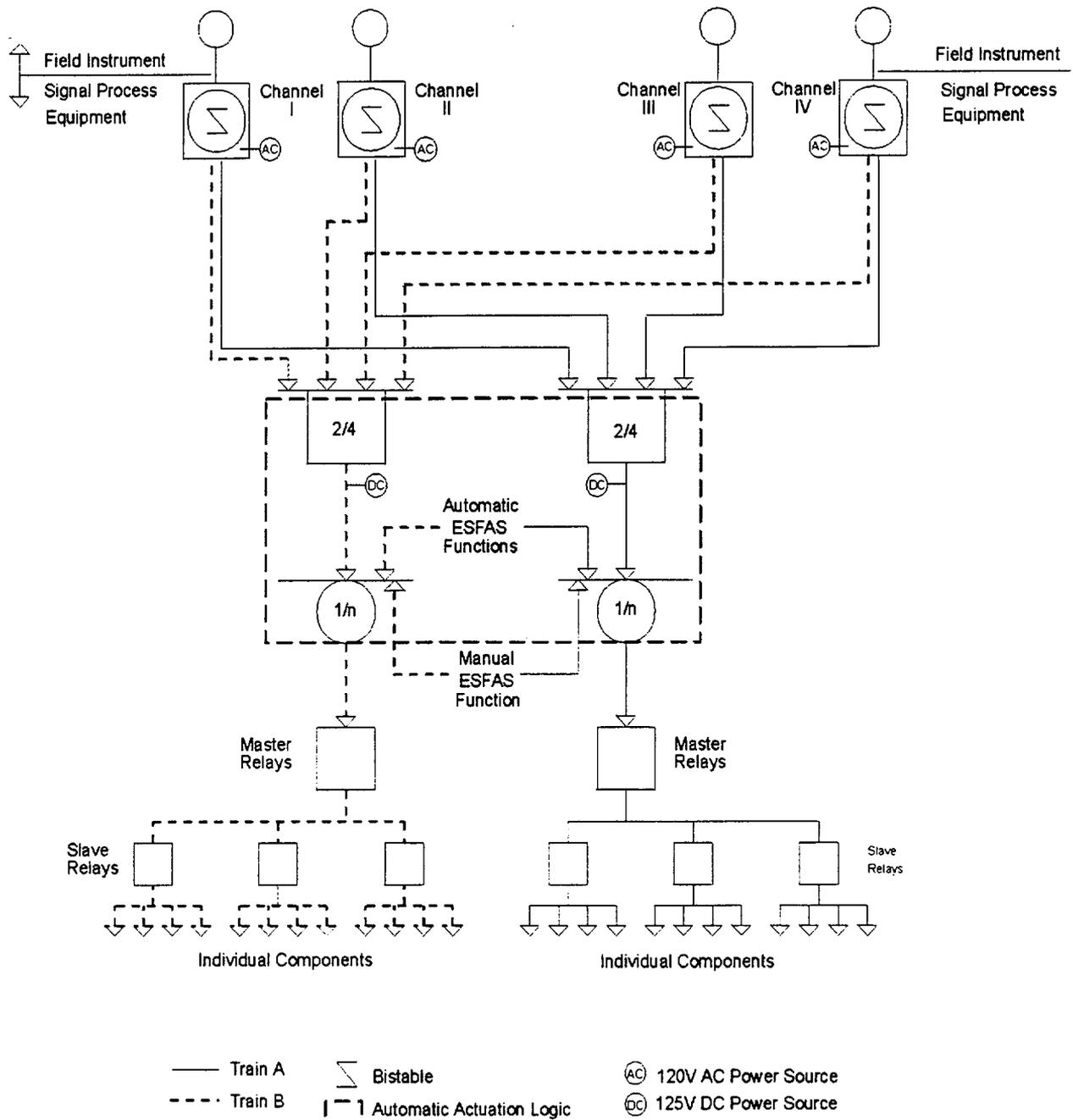
SR 3.3.2.7

This SR is the performance of an ACTUATION LOGIC TEST on all ESFAS Automatic Actuation Logic and Actuation Relays Functions every 24 months. This test includes the application of various simulated or actual input combinations in conjunction with each possible interlock state and verification of the required logic output. Relay and contact operation is verified by a continuance check or actuation of the end device.

The Frequency of 24 months is based on operating experience and the need to perform this testing during a plant shutdown to prevent a reactor trip from occurring.

REFERENCES

1. Atomic Industrial Forum (AIF) GDC 15, Issued for Comment July 10, 1967.
 2. UFSAR, Chapter 7.
 3. UFSAR, Chapter 6.
 4. UFSAR, Chapter 15.
 5. IEEE-279-1971.
 6. EWR-5126, "Guidelines For Instrument Loop Performance Evaluation and Setpoint Verification," August 1992.
 7. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.
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For illustration only

Figure B 3.3.2-1

B 3.3 INSTRUMENTATION

B 3.3.3 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM instrumentation is to display plant variables that provide information required by the control room operators during accident conditions. This instrumentation provides the necessary support for the operator to take required manual actions, verify that automatic and required manual safety functions have been completed, and to determine if fission product barriers have been breached following a Design Basis Accident (DBA).

The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected plant parameters to monitor and assess plant status and behavior during an accident.

The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. These essential instruments are identified in Reference 1 addressing the recommendations of Regulatory Guide 1.97 (Ref. 2) as required by Supplement 1 to NUREG-0737 (Ref. 3).

The instrument channels required to be OPERABLE by this LCO provide information for key parameters identified during implementation of Regulatory Guide 1.97 as Category I variables. Category I variables are organized into four types and are the key variables deemed risk significant because they are needed to:

- a. Provide the primary information required for the control room operator to take specific manually controlled actions for which no automatic control is provided, and that are required for safety systems to accomplish their safety functions for DBAs (Type A).
- b. Provide the primary information required for the control room operator to verify that required automatic and manually controlled functions have been accomplished (Type B);
- c. Provide information to the control room operators that will enable them to determine the likelihood of a gross breach of the barriers to radioactivity release (Type C); and
- d. Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public, and to estimate the magnitude of any impending threat (Type E).

All Type A and key Type B, C, and E parameters have been identified as Category I variables in Reference 1 which also provides justification for deviating from the NRC proposed list of Category I variables.

The specific instrument Functions listed in Table 3.3.3-1 are discussed in the LCO section.

APPLICABLE
SAFETY
ANALYSES

The PAM instrumentation ensures the availability of Regulatory Guide 1.97 Category I variables so that the control room operating staff can:

- Perform the diagnosis specified in the emergency operating procedures for the primary success path of DBAs (e.g., loss of coolant accident (LOCA));
- Take the specified, pre-planned, manually controlled actions, for which no automatic control is provided, and that are required for safety systems to accomplish their safety function;
- Determine whether required automatic and manual safety functions have been accomplished;
- Determine the likelihood of a gross breach of the barriers to radioactivity release;
- Determine if a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

PAM instrumentation that meets the definition of Type A in Regulatory Guide 1.97 satisfies Criterion 3 of the NRC Policy Statement. Category I, non-Type A, instrumentation must be retained in TS because it is intended to assist operators in minimizing the consequences of accidents. Therefore, Category I, non-Type A, variables are important for reducing public risk and satisfy Criterion 4.

LCO

The PAM instrumentation LCO provides OPERABILITY requirements for Regulatory Guide 1.97 Type A monitors, which provide information required by the control room operators to perform certain manual actions specified in the plant Emergency Operating Procedures. These manual actions ensure that a system can accomplish its safety function, and are credited in the safety analyses. Additionally, this LCO addresses Regulatory Guide 1.97 instruments that have been designated Category I, non-Type A.

The OPERABILITY of the PAM instrumentation ensures there is sufficient information available on selected plant parameters to monitor and assess plant status following an accident.

This LCO requires two OPERABLE channels for most Functions. Two OPERABLE channels ensure no single failure prevents operators from obtaining the information necessary to determine the safety status of the plant, and to bring the plant to and maintain it in a safe condition following an accident.

Furthermore, OPERABILITY of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information. More than two channels may be required if failure of one accident monitoring channel results in information ambiguity (that is, the redundant displays disagree) that could lead operators to defeat or fail to accomplish a required safety function.

Table 3.3.3-1 lists all Category I variables identified by Reference 1.

Category I variables are considered OPERABLE when they are capable of providing immediately accessible display and continuous readout in the control room. The Hydrogen Monitors are considered OPERABLE when continuous readout is available in the Control Room or in the relay room. Each channel must also be supplied by separate electrical trains except as noted below. In addition, in accordance with LCO 3.0.6, it is not required to declare a supported system inoperable due to the inoperability of the support system (e.g., electric power). Since the inoperability of Instrument Bus D does not have any associated Required Actions, the loss of this power source may affect the OPERABILITY of the Pressurizer Pressure and SG Water Level (Narrow Range) Functions.

Listed below are discussions of the specified instrument Functions listed in Table 3.3.3-1.

1. Pressurizer Pressure

Pressurizer Pressure is a Type A variable used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Pressurizer pressure is also used to verify the plant conditions necessary to establish natural circulation in the RCS and to verify that the plant is maintained in a safe shutdown condition. Any of the following combinations of pressure transmitters comprise the two channels required for this function:

- PT-429 and PT-431;
- PT-430 and PT-431;
- PT-429 and PT-449;

- PT-430 and PT-449; or
- PT-431 and PT-449

The loss of Instrument Bus D requires declaring PT-449 inoperable.

2. Pressurizer Level

Pressurizer Level is a Type A variable used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Pressurizer water level is also used to verify that the plant is maintained in a safe shutdown condition. Any of the following combinations of level transmitters comprise the two channels required for this function:

- LT-426 and LT-428; or
- LT-427 and LT-428.

3. Reactor Coolant System (RCS) Hot Leg Temperature

RCS Hot and Cold Leg Temperatures are Category I variables (RCS Cold Leg Temperature is also a Type A variable) provided for verification of core cooling and long term surveillance of RCS integrity.

RCS hot and cold leg temperatures are used to determine RCS subcooling margin. RCS subcooling margin will allow termination of SI, if still in progress, or reinitiation of SI if it has been stopped. RCS subcooling margin is also used for plant stabilization and cooldown control.

In addition, RCS cold leg temperature is used in conjunction with RCS hot leg temperature to verify natural circulation in the RCS.

Temperature inputs are provided by two independent temperature sensor resistance elements and associated transmitters in each loop. Temperature elements TE-409B-1 and TE-410B-1 provide the required RCS cold leg temperature input for RCS Loops A and B, respectively. Temperature elements TE-409A-1 and TE-410A-1 provide the required RCS hot leg temperature input for RCS Loops A and B, respectively.

4. RCS Cold Leg Temperature

Refer to description of Function number 3 above.

5. RCS Pressure (Wide Range)

RCS wide range pressure is a Type A variable provided for verification of core cooling and the long term surveillance of RCS integrity.

RCS pressure is used to verify delivery of SI flow to the RCS from at least one train when the RCS pressure is below the SI pump shutoff head. RCS pressure is also used to verify closure of manually closed pressurizer spray line valves and pressurizer power operated relief valves (PORVs) and for determining RCS subcooling margin.

RCS pressure can also be used:

- to determine whether to terminate actuated SI or to reinitiate stopped SI;
- to determine when to reset SI and stop the residual heat removal pumps (RHR);
- to manually restart the RHR pumps;
- as reactor coolant pump (RCP) trip criteria;
- to make a determination on the nature of the accident in progress and where to go next in the emergency operating procedure; and
- to determine whether to operate the pressurizer heaters.

RCS pressure is also related to three decisions about depressurization. They are:

- to determine whether to proceed with primary system depressurization;
- to verify termination of depressurization; and
- to determine whether to close accumulator isolation valves during a controlled cooldown/depressurization.

RCS pressure is a Type A variable because the operator uses this indication to monitor the cooldown of the RCS following a steam generator tube rupture (SGTR) or small break LOCA. Operator actions to maintain a controlled cooldown, such as adjusting steam generator (SG) pressure or level, would use this indication.

RCS pressure transmitters PT-420 and PT-420A provide the two required channels for this function.

6. RCS Subcooling Monitor

RCS Subcooling Monitor is a Type A variable provided for verification of core cooling and long term surveillance of RCS integrity. The RCS Subcooling Monitor is used to provide information to the operator, derived from RCS hot leg temperature and RCS pressure, on subcooling. RCS subcooling margin is used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. RCS subcooling margin is also used for plant stabilization and cooldown control.

The emergency operating procedures determine RCS subcooling margin based on the core exit thermocouples (CETs) and RCS pressure. Therefore, any of the following combination of parameters comprise the two required channels for this function:

- TI-409A and TI-410A; or
- One pressurizer pressure transmitter and two CETs in each of the four quadrants supplied by electrical train A and train B (i.e., total of two pressurizer pressure transmitters and 16 CETs).

7. Reactor Vessel Water Level

Reactor Vessel Water Level is a Type A variable provided for verification and long term surveillance of core cooling. It is also used for accident diagnosis and to determine reactor coolant inventory adequacy.

When both RCPs are stopped, the Reactor Vessel Water Level Indication System (RVLIS) provides a direct measurement of the collapsed liquid level above the fuel alignment plate. The collapsed level represents the amount of liquid mass that is in the reactor vessel above the core. When the RCPs are operating, RVLIS indicates the fluid fraction of the RCS. Measurement of the collapsed water level or fluid fraction is selected because it is a direct indication of the water inventory.

Level transmitters LT-490A and LT-490B provide the two required channels for this function.

8. Containment Sump B Water Level

Containment Sump B Water Level is a Type A variable provided for verification and long term surveillance of RCS integrity.

Containment Sump B Water Level is used to determine:

- containment sump level for accident diagnosis;
- when to begin the recirculation procedure; and
- whether to terminate SI, if still in progress.

Level transmitters LT-942 and LT-943, each with five discrete level switches, provide the two required channels for this function.

9. Containment Pressure (Wide Range)

Containment Pressure (Wide Range) is a Type A variable provided for verification of RCS and containment OPERABILITY.

Containment Pressure (Wide Range) is used to determine the type of accident in progress and when, and if, to use emergency operating procedure containment adverse values.

Any of the following combinations of pressure transmitters comprise the two required channels for this function:

- PT-946 and PT-948; or
- PT-950 and PT-948.

10. Containment Area Radiation (High Range)

Containment Area Radiation (High Range) is a Type E Category I variable provided to monitor for the potential of significant radiation releases into containment and to provide release assessment for use by operators in determining the need to invoke site emergency plans.

Containment radiation level is used to determine the type of accident in progress (e.g., LOCA), and when, or if, to use emergency operating procedure containment adverse values.

Radiation monitors R-29 and R-30 are used to provide the two required channels for this function.

11. Hydrogen Monitors

Hydrogen Concentration is a Type C Category I variable provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion. This variable is also important in verifying the adequacy of mitigating actions.

Hydrogen monitors HMSLCPA and HMSLCPB provide the two required channels for this function. In addition, the Post Accident Sampling System may take the place of one of these monitors provided that monitor HMSLCPA or HMSLCPB remains OPERABLE. The PASS system Hydrogen Function is not required to provide continuous readout in the control room or relay room for OPERABILITY. The use of the PASS system is allowed due to the long time period following the accident before hydrogen monitoring would be required (Ref. 4). Since the PASS system does not receive power from any safeguards buses, there are no support system LCOs related to its OPERABILITY (i.e., LCO 3.0.6 is not relevant). Instead, OPERABILITY of the PASS system is addressed by the Post Accident Sampling Program (Specification 5.5.3).

12. Condensate Storage Tank Level

Condensate Storage Tank (CST) Level is a Type A variable provided to ensure a water supply is available for the preferred Auxiliary Feedwater (AFW) System. The CST consists of two identical tanks connected by a common outlet header.

CST level is used to determine:

- if sufficient CST inventory is available immediately following a loss of normal feedwater or small break LOCA; and
- when to manually replenish the CST or align the safety related source of water (service water) to the preferred AFW system.

Level transmitters LT-2022A and LT-2022B provide the two required channels for this function. However, only the level transmitter associated with the CST(s) required by LCO 3.7.6, "Condensate Storage Tank(s)" are required for this LCO.

13. Refueling Water Storage Tank Level

Refueling Water Storage Tank (RWST) Level is a Type A variable provided for verifying a water source to the SI, RHR, and Containment Spray (CS) Systems.

The RWST level accuracy is established to allow an adequate supply of water to the SI, RHR, and CS pumps during the switchover to the recirculation phase of an accident. A high degree of accuracy is required to maximize the time available to the operator to complete the switchover to the sump recirculation phase and ensure sufficient water is available to maintain adequate net positive suction head (NPSH) to operating pumps.

Level transmitters LT-920 and LT-921 provide the two required channels for this function.

14. Residual Heat Removal Flow

Residual Heat Removal (RHR) Flow is a Type A variable provided for verifying low pressure safety injection to the reactor vessel and to the CS and SI pumps.

RHR flow is used to determine when to stop the RHR pumps and if sufficient flow is available to the CS and SI pumps during recirculation.

Since different flow transmitters are used to verify injection to the reactor vessel and to verify flow to the CS and SI pumps, FT-626 and FT-931A comprise one required channel and FT-689 and FT-931B comprise a second required channel.

15. Core Exit Temperature - QUADRANT 1

Core Exit Temperature is a Type A variable provided for verification and long term surveillance of core cooling.

An evaluation was made of the minimum number of valid CETs necessary for measuring core cooling. The evaluation determined the necessary complement of CETs required to detect initial core recovery and trend the ensuing core heatup. The evaluation accounted for core nonuniformities, including incore effects of the radial decay power distribution, excore effects of reflux in the hot legs, and nonuniform inlet temperatures. Based on these evaluations, adequate core cooling is ensured with two valid Core Exit Temperature channels per quadrant with two CETs per required channel.

Core Exit Temperature is used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Core Exit Temperature is also used for plant stabilization and cooldown control.

Two OPERABLE channels of Core Exit Temperature are required in each quadrant to provide indication of radial distribution of the coolant temperature rise across representative regions of the core. Because of the small core size, two randomly selected thermocouples are sufficient to meet the two thermocouples per channel requirement in any quadrant. However, a CET which lies directly on the dividing line between two quadrants can only be used to satisfy the minimum required channels for one quadrant.

A CET is considered OPERABLE when it is within ± 35 °F of the average CET reading except for the CETs associated with peripheral assemblies. These CETs (A7, B5, C3, C.11, D2, D12, H13, I2, K3, K11, L10, and M6) are considered OPERABLE when they are within ± 43 °F of the average CET reading. At least two CETs from each of the following trains must be OPERABLE in each of the four quadrants:

Train A		Train B	
<u>CET</u>	<u>Location</u>	<u>CET</u>	<u>Location</u>
T2	M6	T1	I4
T5	J3	T3	L7
T6	I2	T4	K3
T7	J6	T10	J9
T8	L10	T11	I7
T9	J8	T13	K11
T12	H6	T14	D12
T15	H9	T16	H10
T18	F8	T17	E10
T21	C11	T19	G7
T22	H11	T20	C8
T23	H13	T24 ^(a)	F12
T26 ^(b)	I10	T25	G12
T28	D5	T27	E6
T33	D2	T29	E4
T34	C3	T30 ^(b)	G4
T36	B7	T31 ^(a)	G2
T38	B5	T32 ^(b)	G1
		T35	A7
		T37	C6

(a) Removed from scan.

(b) These thermocouples are in the reactor vessel head and cannot be credited with respect to this LCO.

16. Core Exit Temperature - QUADRANT 2

Refer to description of Function number 15 above.

17. Core Exit Temperature - QUADRANT 3

Refer to description of Function number 15 above.

18. Core Exit Temperature - QUADRANT 4

Refer to description of Function number 15 above.

19. Auxilliary Feedwater (AFW) Flow to Steam Generator (SG) A

Auxilliary Feedwater (AFW) Flow is a Type A variable provided to monitor operation of the preferred AFW system.

The AFW System provides decay heat removal via the SGs and is comprised of the preferred AFW System and the Standby AFW (SAFW) System. The use of the preferred AFW or SAFW System to provide this decay heat removal function is dependent upon the type of accident. AFW flow indication is required from the three pump trains which comprise the preferred AFW System since these pumps automatically start on various actuation signals. The failure of the preferred AFW System (e.g., due to a high energy line break (HELB) in the Intermediate Building) is detected by AFW flow indication. At this point, the SAFW System is manually aligned to provide the decay heat removal function.

SAFW flow can also be used to verify that AFW flow is being delivered to the SGs. However, the primary indication of this is provided by SG water level. Therefore, flow indication from the SAFW pumps is not required.

Each of the three preferred AFW pump trains has two redundant transmitters; however, only the flow transmitter supplied power from the same electrical train as the AFW pump is required for this LCO. Therefore, flow transmitters FT-2001 (MCB indicator FI-2021A) and FT-2006 (MCB indicator FI-2023A) comprise the two required channels for SG A and FT-2002 (MCB indicator FI-2022A) and FT-2007 (MCB indicator FI-2024A) comprise the two required channels for SG B.

20. AFW Flow to SG B

Refer to description of Function number 19 above.

21. SG A Water Level (Narrow Range)

Steam Generator (SG) Water Level is a Type A variable provided to monitor operation of decay heat removal via the SGs. For the narrow range level, the signals from the transmitters are independently indicated on the main control board as 0% to 100%. This corresponds to approximately above the top of the tube bundles to the top of the swirl vane separators (span of 143 inches). For the wide range level, signals from the transmitters are indicated as 0 to 520 inches (0% to 100%) on the main control board.

SG Water Level (Narrow and Wide Range) is used to:

- identify the faulted SG following a tube rupture;
- verify that the intact SGs are an adequate heat sink for the reactor;
- determine the nature of the accident in progress (e.g., verify an SGTR); and
- verify plant conditions for termination of SI during secondary plant HELBs outside containment.

Redundant monitoring capability is provided by two trains of instrumentation per SG.

SG Water Level (Narrow Range) requires 2 channels of indication per SG. This can be met using any of the following combinations of level transmitters for SG A:

- LT-461 and LT-462;
- LT-462 and LT-463; or
- LT-461 and LT-463;

For SG B, any of the following combinations of level transmitters can be used:

- LT-471 and LT-473;
- LT-471 and LT-472; or
- LT-472 and LT-473.

The loss of Instrument Bus D requires declaring LT-463 and LT-471 inoperable.

SG Water Level (Wide Range) requires 2 channels of indication per SG. Two channels per SG are required since the loss of one channel with no backup available may result in the complete loss of information required by the operators to accomplish necessary safety functions. Level transmitters LT-504 and LT-505 comprise the two required channels for SG A and LT-506 and LT-507 comprise the two required channels for SG B.

22. SG B Water Level (Narrow Range)

Refer to description of Function number 21 above.

23. SG A Water Level (Wide Range)

Refer to description of Function number 21 above.

24. SG B Water Level (Wide Range)

Refer to description of Function number 21 above.

25. SG A Pressure

Steam Generator (SG) Pressure is a Type A variable provided to monitor operation of decay heat removal via the SGs. The signals from the transmitters are calibrated for a range of 0 psig to 1400 psig. Redundant monitoring capability is provided by three available trains of instrumentation.

Any of the following combinations of pressure transmitters comprise the two required channels for SG A:

- PT-468 and PT-482; or
- PT-469 and PT-482.

Any of the following combinations of pressure transmitters comprise the two required channels for SG B:

- PT-479 and PT-478; or
- PT-478 and PT-483.

26. SG B Pressure

Refer to description of Function number 25 above.

APPLICABILITY The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables are related to the diagnosis and pre-planned actions required to mitigate DBAs. The applicable DBAs are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, the PAM instrumentation is not required to be OPERABLE because plant conditions are such that the likelihood of an event that would require PAM instrumentation is low.

ACTIONS The ACTIONS are modified by two Notes.

Note 1 has been added to exclude the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the passive function of the instruments, the operator's ability to respond to an accident using alternate instruments and methods, and the low probability of an event requiring these instruments.

Note 2 has been added to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed on Table 3.3.3-1. The Completion Time(s) of 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 when one or more Functions have one required channel that is inoperable. Required Action A.1 requires restoring the inoperable channel to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel, the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM Instrumentation during this interval.

Condition A is modified by a Note which states that the Condition is not applicable to Table 3.3.3-1 Functions 3 and 4. These Functions are addressed by Condition C which provides the necessary required actions for these single channel Functions.

B.1

Condition B applies when the Required Action and associated Completion Time for Condition A is not met. This Condition requires the immediate initiation of actions to prepare and submit a special report to the NRC. This report shall be submitted within the following 14 days from the time the Condition is entered. This report shall discuss the results of the root cause evaluation of the inoperability and identify proposed restorative actions or alternate means of providing the required function. This action is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability, and given the likelihood of plant conditions that would require information provided by this instrumentation. If alternate means are to be used, they must be developed and tested prior to submittal of the special report.

C.1

Condition C applies when a Function has one inoperable required channel and no diverse channel OPERABLE (i.e., loss of RCS Hot Leg Temperature or RCS Cold Leg Temperature Functions). This Condition requires restoring the inoperable channel in the affected Function to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information. Continuous operation with a complete loss of function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of the inoperable channel limits the risk that the PAM Function will be in a degraded condition should an accident occur.

Condition C is modified by a Note which states that this Condition is only applicable to Table 3.3.3-1 Functions 3 and 4. All remaining Functions are addressed by Condition A with one channel inoperable.

D.1

Condition D applies when one or more Functions have two inoperable required channels (i.e., two channels inoperable in the same Function). Required Action D.1 requires restoring one channel in the Function(s) to OPERABLE status within 7 days. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrument operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur. Condition D is modified by a Note that excludes Function 11 since the inoperability of two hydrogen monitor channels is addressed by Condition E.

E.1

Condition E applies when two hydrogen monitor channels are inoperable. This Condition requires restoring one hydrogen monitor channel to OPERABLE status within 72 hours. The 72 hour Completion Time is reasonable based on the backup capability of the Post Accident Sampling System to monitor the hydrogen concentration for evaluation of core damage and to provide information for operator decisions. Also, it is unlikely that a LOCA which would potentially require use of the hydrogen recombiners would occur during this time.

F.1

Condition F applies when the Required Action and associated Completion Time of Condition C, D, or E are not met. Required Action F.1 requires entering the appropriate Condition referenced in Table 3.3.3-1 for the channel immediately. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met any Required Action of Condition C, D, or E, and the associated Completion Time has expired, Condition F is entered for that channel and provides for transfer to the appropriate subsequent Condition.

G.1 and G.2

If one channel for Function 3 and 4 cannot be restored to OPERABLE status within the required Completion Time for Condition C, if one channel for Function 1, 2, 3, 4, 5, 6, 8, 9, 12, 13, 14, 15, 16, 17, 18, 19, 20, 21, 22, 23, 24, 25, or 26 cannot be restored to OPERABLE status within the required Completion Time of Condition D, or if one channel for Function 11 cannot be restored to OPERABLE status within the required Completion Time of Condition E, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

H.1

If one channel for Function 7 or 10 cannot be restored to OPERABLE status within the required Completion Time of Condition D, the plant must take immediate action to prepare and submit a special report to the NRC. This report shall be submitted within the following 14 days from the time the action is required. This report discusses the alternate means of monitoring Reactor Vessel Water Level and Containment Area Radiation, the degree to which the alternate means are equivalent to the installed PAM channels, the areas in which they are not equivalent, and a schedule for restoring the normal PAM channels.

These alternate means must have been developed and tested and may be temporarily installed if the normal PAM channel(s) cannot be restored to OPERABLE status within the allotted time.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation Function in Table 3.3.3-1.

SR 3.3.3.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross instrumentation failure 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 more serious instrument conditions. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each

CHANNEL CALIBRATION. The high radiation instrumentation should be compared to similar plant instruments located throughout the plant.

Channel check acceptance criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

As specified in the SR, a CHANNEL CHECK is only required for those channels that are normally energized.

The Frequency of 31 days is based on operating experience that demonstrates that channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.3.2

A CHANNEL CALIBRATION is performed every 24 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 the measured parameter with the necessary range and accuracy. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the Core Exit thermocouple sensors shall include an in-place qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. This is accomplished by an in-place cross calibration that compares the other sensing elements with the recently installed sensing element. The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. UFSAR, Section 7.5.2.
 2. Regulatory Guide 1.97, Rev. 3.
 3. NUREG-0737, Supplement 1, "TMI Action Items."
 4. UFSAR, Section 6.2.5.
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B 3.3 INSTRUMENTATION

B 3.3.4 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation

BASES

BACKGROUND The DGs provide a source of emergency power when offsite power is either unavailable or is insufficiently stable to allow safe plant operation. The LOP DG start instrumentation consists of two channels on each of safeguards Buses 14, 16, 17, and 18 (Ref. 1). Each channel contains one loss of voltage relay and one degraded voltage relay (see Figure B 3.3.4-1). A one-out-of-two logic in both channels will cause the following actions on the associated safeguards bus:

- a. trip of the normal feed breaker from offsite power;
- b. trip of the bus-tie breaker to the opposite electrical train (if closed);
- c. shed of all bus loads except the CS pump, component cooling water pump (if no safety injection signal is present), and safety related motor control centers; and
- d. start of the associated DG.

The degraded voltage logic is provided on each 480 V safeguards bus to protect Engineered Safety Features (ESF) components from exposure to long periods of reduced voltage conditions which can result in degraded performance and to ensure that required motors can start. The loss of voltage logic is provided on each 480 V safeguards bus to ensure the DG is started within the time limits assumed in the accident analysis to provide the required electrical power if offsite power is lost.

The degraded voltage relays have time delays which have inverse operating characteristics such that the lower the bus voltage, the faster the operating time. The loss of voltage relays have definite time delays which are not related to the rate of the loss of bus voltage. These time delays are set to permit voltage transients during worst case motor starting conditions.

APPLICABLE SAFETY ANALYSES

The LOP DG start instrumentation is required for the ESF Systems to function in any accident with a loss of offsite power. Its design basis is that of the ESF Actuation System (ESFAS). Undervoltage conditions which occur independent of any accident conditions result in the start and bus connection of the associated DG, but no automatic loading occurs.

Accident analyses credit the loading of the DG based on the loss of offsite power during a Design Basis Accident (DBA). The most limiting DBA of concern is the large break loss of coolant accident (LOCA) which requires ESF Systems in order to maintain containment integrity and protect fuel contained within the reactor vessel (Ref. 2). The detection and processing of an undervoltage condition, and subsequent DG loading, has been included in the delay time assumed for each ESF component requiring DG supplied power following a DBA and loss of offsite power.

The loss of offsite power has been assumed to occur either coincident with the DBA or at a later period (40 to 90 seconds following the reactor trip) due to a grid disturbance caused by the turbine generator trip. If the loss of offsite power occurs at the same time as the safety injection (SI) signal parameters are reached, the accident analyses assumes the SI signal will actuate the DG within 2 seconds and that the DG will connect to the affected safeguards bus within an additional 10 seconds (12 seconds total time). If the loss of offsite power occurs before the SI signal parameters are reached, the accident analyses assumes the LOP DG start instrumentation will actuate the DG within 2.75 seconds and that the DG will connect to the affected safeguards bus within an additional 10 seconds (12.75 seconds total time). If the loss of offsite power occurs after the SI signal parameters are reached (grid disturbance), the accident analyses assumes the DG will connect to the bus within 1.5 seconds after the feeder breaker to the bus is opened (DG was actuated by SI signal). The grid disturbance has been evaluated based on a 140 °F peak clad temperature penalty during a LOCA and demonstrated to result in acceptable consequences.

The degraded voltage and undervoltage setpoints are based on the minimum voltage required for continued operation of ESF Systems assuming worst case loading conditions (i.e., maximum loading upon DG sequencing). The Trip Setpoint for the loss of voltage relays, and associated time delays, have been chosen based on the following considerations:

- a. Actuate the associated DG within 2.75 seconds as assumed in the accident analysis; and
- b. Prevent DG actuation on momentary voltage drops associated with starting of ESF components during an accident with offsite power available and during normal operation due to minor system disturbances. Therefore, the time delay setting must be greater than the time between the largest assumed voltage drop below the voltage setting and the reset value of the trip function.

The Trip Setpoint for the degraded voltage channels, and associated time delays, have been chosen based on the following considerations;

- a. Prevent motors supplied by the 480 V bus from operating at reduced voltage conditions for long periods of time; and
- b. Prevent DG actuation on momentary voltage drops associated with starting of ESF components during an accident with offsite power available, and during normal operation due to minor system disturbances. Therefore, the time delay setting must be greater than the time between the largest voltage drop below the maximum voltage setting and the reset value of the trip function.

The LOP DG start instrumentation channels satisfy Criterion 3 of the NRC Policy Statement.

LCO

This LCO requires that each 480 V safeguards bus have two OPERABLE channels of the LOP DG start instrumentation in MODES 1, 2, 3, and 4 when the associated DG supports safety systems associated with the ESFAS. In MODES 5 and 6, the LOP DG start instrumentation channels for each 480 V safeguards bus must be OPERABLE whenever the associated DG is required to be OPERABLE to ensure that the automatic start of the DG is available when needed. Loss of the LOP DG Start Instrumentation Function could result in the delay of safety systems initiation when required. This could lead to unacceptable consequences during accidents.

The LOP start instrumentation is considered OPERABLE when two channels, each comprised of one degraded voltage and one loss of voltage relays are available for each 480 V safeguards bus (i.e., Bus 14, 16, 17, and 18). Each of the LOP channels must be capable of detecting undervoltage conditions within the voltage limits and time delays assumed in the accident analysis.

The Allowable Values and Trip Setpoints for the degraded voltage and loss of voltage Functions are specified in SR 3.3.4.2. The Allowable Values specified in SR 3.3.4.2 are those setpoints which ensure that the associated DG will actuate within 2.75 seconds on undervoltage conditions, and that the DG will not actuate on momentary voltage drops which could affect ESF actuation times as assumed in the accident analysis. The Trip Setpoints specified in SR 3.3.4.2 are the nominal setpoints selected to ensure that the setpoint measured by the Surveillance does not exceed the Allowable Value accounting for maximum instrument uncertainties between scheduled surveillances. Therefore, LOP start instrumentation channels are OPERABLE when the CHANNEL CALIBRATION "as left" value is within the Trip Setpoint limits and the CHANNEL CALIBRATION and TADOT "as found" value is within

the Allowed Value setpoints. The basis for all setpoints is contained in Reference 3.

APPLICABILITY The LOP DG Start Instrumentation Functions are required in MODES 1, 2, 3, and 4 because ESF Functions are designed to provide protection in these MODES. Actuation in MODE 5 or 6 is required whenever the required DG must be OPERABLE so that it can perform its function on an LOP or degraded power to the 480 V safeguards buses.

ACTIONS In the event a relay's Trip Setpoint is found to be nonconservative with respect to the Allowable Value, or the channel is found to be inoperable, then the channel must be declared inoperable and the LCO Condition entered as applicable.

A Note has been added in the ACTIONS to clarify the application of Completion Time rules. This Note states that separate Condition entry is allowed for each 480 V safeguards bus.

A.1

With one or more 480 V bus(es) with one channel inoperable, Required Action A.1 requires the inoperable channel(s) to be placed in trip within 6 hours. With an undervoltage channel in the tripped condition, the LOP DG start instrumentation channels are configured to provide a one-out-of-one logic to initiate a trip of the incoming offsite power for the respective bus. The remaining OPERABLE channel is comprised of one-out-of-two logic from the degraded and loss of voltage relays. Any additional failure of either of these two OPERABLE relays requires entry into Condition B.

B.1

Condition B applies to the LOP DG start Function when the Required Action and associated Completion Time for Condition A are not met or with one or more 480 V bus(es) with two channels of LOP start instrumentation inoperable.

Condition B requires immediate entry into the Applicable Conditions specified in LCO 3.8.1, "AC Sources - MODES 1, 2, 3, and 4," or LCO 3.8.2, "AC Sources - MODES 5 and 6," for the DG made inoperable by failure of the LOP DG start instrumentation. The actions of those LCOs provide for adequate compensatory actions to assure plant safety.

**SURVEILLANCE
REQUIREMENTS**

The Surveillances are modified by a Note to indicate that, when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 4 hours, provided the second channel maintains trip capability. Upon completion of the Surveillance, or expiration of the 4 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the assumption that 4 hours is the average time required to perform channel surveillance. Based on engineering judgement, the 4 hour testing allowance does not significantly reduce the probability that the LOP DG start instrumentation will trip when necessary.

SR 3.3.4.1

This SR is the performance of a TADOT every 31 days. This test checks trip devices that provide actuation signals directly. For these tests, the relay Trip Setpoints are verified and adjusted as necessary to ensure Allowable Values can still be met. The 31 day Frequency is based on the known reliability of the relays and controls and has been shown to be acceptable through operating experience.

SR 3.3.4.2

This SR is the performance of a CHANNEL CALIBRATION every 24 months, or approximately at every refueling of the LOP DG start instrumentation for each 480 V bus.

The voltage setpoint verification, as well as the time response to a loss of voltage and a degraded voltage test, shall include a single point verification that the trip occurs within the required time delay.

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.

The Frequency of 24 months is based on operating experience consistent with the typical industry refueling cycle and is justified by the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. UFSAR, Section 8.3.
 2. UFSAR, Chapter 15.
 3. RG&E Design Analysis DA-EE-93-006-08, "480 Volt Undervoltage Relay Settings and Test Acceptance Criteria."
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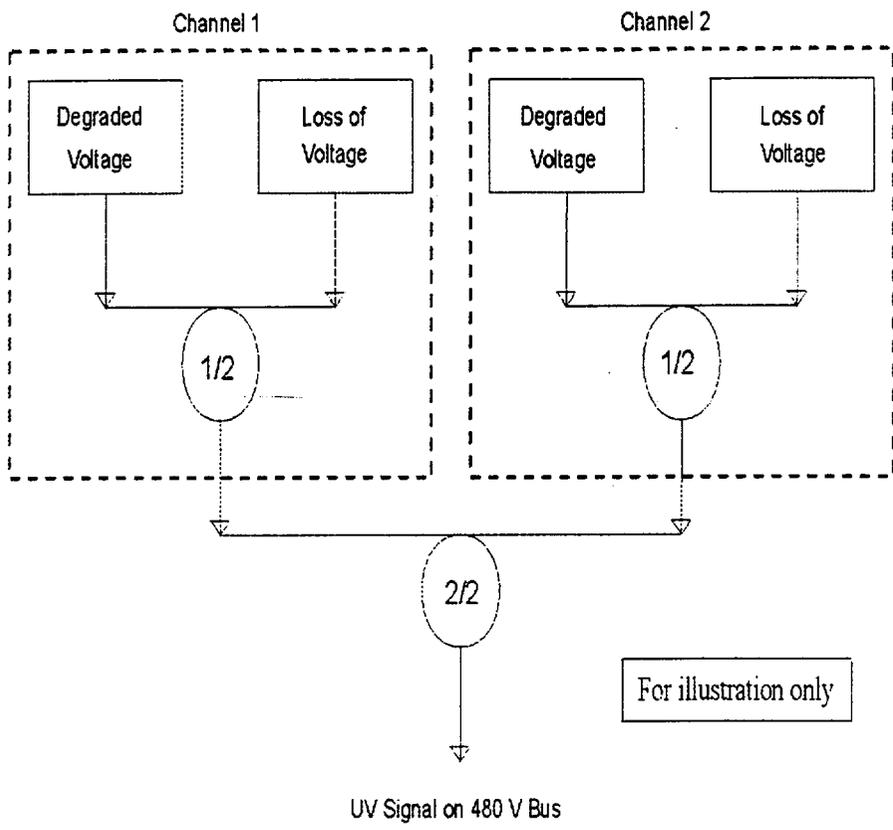


Figure B 3.3.4-1
DG LOP Instrumentation

B 3.3 INSTRUMENTATION

B 3.3.5 Containment Ventilation Isolation Instrumentation

BASES

BACKGROUND

Containment ventilation isolation instrumentation closes the containment isolation valves in the Mini-Purge System and the Shutdown Purge System. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The Mini-Purge System may be used in all MODES while the Shutdown Purge System may only be used with the reactor shutdown.

Containment ventilation isolation initiates on a containment isolation signal, containment radiation signal, or by manual actuation of containment spray (CS). The Bases for LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss the containment isolation and manual containment spray modes of initiation.

Two containment radiation monitoring channels are provided as input to the containment ventilation isolation. The two radiation detectors are of different types: gaseous (R-12), and particulate (R-11). Both detectors will respond to most events that release radiation to containment. However, analyses have not been conducted to demonstrate that all credible events will be detected by more than one monitor. Therefore, for the purposes of this LCO the two channels are not considered redundant. Instead, they are treated as two one-out-of-one Functions. Since the radiation monitors constitute a sampling system, various components such as sample line valves, sample line heaters, sample pumps, and filter motors are required to support monitor OPERABILITY.

The Mini-Purge System has inner and outer containment isolation valves in its supply and exhaust ducts while the Shutdown Purge System only has one valve located outside containment since the inside valve was replaced by a blind flange that is used during MODES 1, 2, 3, and 4. A high radiation signal from any one of the two channels initiates containment ventilation isolation, which closes all isolation valves in the Mini-Purge System and the Shutdown Purge System. These systems are described in the Bases for LCO 3.6.3, "Containment Isolation Boundaries."

APPLICABLE
SAFETY
ANALYSES

The safety analyses assume that the containment remains intact with penetrations unnecessary for accident mitigation functions isolated early in the event, within approximately 60 seconds. The isolation of the purge valves has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed. The containment ventilation isolation radiation monitors act as backup to the containment isolation signal to ensure closing of the ventilation valves. They are also the primary means for automatically isolating containment in the event of a fuel handling accident during shutdown even though containment isolation is not specifically credited for this event. Containment isolation in turn ensures meeting the containment leakage rate assumptions of the safety analyses, and ensures that the calculated accident 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 3.3.5-1, is OPERABLE.

1. Automatic Actuation Logic and Actuation Relays

The LCO requires two trains 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 2.b, Containment Spray-Manual Initiation, and ESFAS Function 3, Containment Isolation. The applicable MODES and specified conditions for the containment ventilation isolation portion of these Functions are different and less restrictive than those for their respective CS and ESFAS roles. If one or more of the CS or containment isolation Functions becomes inoperable in such a manner that only the Containment Ventilation Isolation Function is affected, the Conditions applicable to their respective isolation Functions in LCO 3.3.2 need not be entered. The less restrictive Actions specified for inoperability of the Containment Ventilation Isolation Functions specify sufficient compensatory measures for this case.

2. Containment Radiation

The LCO specifies two required channels of radiation monitors (R-11 and R-12) to ensure that the radiation monitoring instrumentation necessary to initiate Containment Ventilation Isolation remains OPERABLE.

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.

3. Containment Isolation

Refer to LCO 3.3.2, Function 3, for all initiating Functions and requirements.

4. Containment Spray-Manual Initiation

Refer to LCO 3.3.2, Function 2.a, for all initiating Functions and requirements. This Function provides the manual initiation capability for containment ventilation isolation.

APPLICABILITY

The Automatic Actuation Logic and Actuation Relays, Containment Isolation, Containment Spray-Manual Initiation, and Containment Radiation Functions are required to be OPERABLE in MODES 1, 2, 3, and 4, and during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. 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.

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 plant 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 3.3.5-1. The Completion Time(s) of the inoperable channel(s)/train(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 containment ventilation isolation radiation monitor channel. Since the two containment radiation monitors measure different parameters, failure of a single channel may result in loss of the radiation monitoring Function for certain events. Consequently, the failed channel must be restored to OPERABLE status. The 4 hour allowed to restore the affected channel is justified by 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 system 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 train is inoperable, multiple channels are inoperable, 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 mini-purge isolation valve made inoperable by failure of isolation instrumentation. For example, if R-11 and R-12 were both inoperable, then all four mini-purge isolation valves must be declared inoperable. If CVI Train A were inoperable, then the two mini-purge valves which receive a Train A isolation signal must be declared inoperable.

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 applies to all Containment Ventilation Isolation Functions and addresses the train orientation of the system 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 train is inoperable, multiple channels are inoperable, 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 each mini-purge isolation valve in its closed position or the applicable Conditions of LCO 3.9.3, "Containment Penetrations," are met for each mini-purge isolation valve made inoperable by failure of isolation instrumentation. 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 3.3.5-1 determines which SRs apply to which Containment Ventilation Isolation Functions.

SR 3.3.5.1

Performance of the CHANNEL CHECK once every 24 hours ensures that a gross failure of instrumentation has not occurred and the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The CHANNEL CHECK agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.5.2

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). This test verifies the capability of the instrumentation to provide the containment ventilation system isolation. The setpoint shall be left consistent with the current plant specific calibration procedure tolerance.

SR 3.3.5.3

This SR is the performance of an ACTUATION LOGIC TEST. All possible logic combinations, with and without applicable permissives, are tested for each protection function. In addition, the master relay is tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. This test is performed every 24 months. The Surveillance interval is acceptable based on instrument reliability and industry operating experience.

SR 3.3.5.4

A CHANNEL CALIBRATION is performed every 24 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.

The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. 10 CFR 100.11.
 2. NUREG-1366.
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B 3.3 INSTRUMENTATION

B 3.3.6 Control Room Emergency Air Treatment System (CREATS) Actuation Instrumentation

BASES

BACKGROUND The CREATS provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity. This system is described in the Bases for LCO 3.7.9, "Control Room Emergency Air Treatment System (CREATS)." This LCO only addresses the actuation instrumentation for the high radiation state CREATS Mode F.

The high radiation state CREATS Mode F actuation instrumentation consists of noble gas (R-36), particulate (R-37), and iodine (R-38) radiation monitors. These detectors are located on the operating level on the Turbine Building and utilize a common air supply pump. A high radiation signal from any of these detectors will initiate the CREATS filtration train and isolate each air supply path with two dampers. The control room operator can also initiate the CREATS filtration train and isolate the air supply paths by using a manual pushbutton in the control room.

APPLICABLE SAFETY ANALYSES

The location of components and CREATS related ducting within the control room emergency zone envelope ensures an adequate supply of filtered air to all areas requiring access. The CREATS provides airborne radiological protection for the control room operators in MODES 1, 2, 3, and 4, as demonstrated by the control room accident dose analyses for the most limiting design basis loss of coolant accident and steam generator tube rupture (Ref. 1). This analysis shows that with credit for the CREATS, or with credit for instantaneous isolation of the control room coincident with the accident initiator and no CREATS filtration train available, the dose rates to control room personnel remain within GDC 19 limits.

During movement of irradiated fuel assemblies or during CORE ALTERATIONS, the CREATS ensures control room habitability in the event of a fuel handling accident. It has been demonstrated that the CREATS is not required in the event of a waste gas decay tank rupture (Ref. 2).

The CREATS Actuation Instrumentation satisfies Criterion 3 of the NRC Policy Statement.

LCO

The LCO requirements ensure that instrumentation necessary to initiate the CREATS is OPERABLE.

1. Manual Initiation

The LCO requires one train to be OPERABLE. The train consists of one pushbutton and the interconnecting wiring to the actuation logic. The operator can initiate the CREATS Filtration train at any time by using a pushbutton in the control room. This action will cause actuation of all components in the same manner as any of the automatic actuation signals required by this LCO.

2. Automatic Actuation Logic and Actuation Relays

The LCO requires one train of Actuation Logic and Actuation Relays to be OPERABLE. Actuation logic consists of all circuitry associated with manual initiation and Control Room Intake Monitors within the actuation system, including the initiation relay contacts responsible for actuating the CREATS.

3. Control Room Radiation Intake Monitor

The LCO specifies single channels of iodine (R-38), noble gas (R-36), and particulate (R-37) of the Control Room Intake Monitors to ensure that the radiation monitoring instrumentation necessary to initiate the CREATS filtration train and isolation dampers remains OPERABLE.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CREATS actuation instrumentation must be OPERABLE to control operator exposure during and following a Design Basis Accident.

During movement of irradiated fuel assemblies or during CORE ALTERATIONS, the CREATS actuation instrumentation must be OPERABLE to cope with the release from a fuel handling accident.

ACTIONS

The most common cause of channel inoperability is failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the plant 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. The "as left" Trip Setpoint must be within the tolerance specified by the calibration procedure. If the "as found" Trip Setpoint exceeds the limits specified in Table 3.3.6-1, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each Function. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.6-1 in the accompanying LCO. The Completion Time(s) of the inoperable channel/train 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 one or more Functions with one channel of the CREATS actuation instrumentation inoperable.

If one or more radiation monitor channels, the manual initiation train, or the automatic actuation logic train is inoperable, action must be taken to restore OPERABLE status within 1 hour or isolate the control room from outside air. In this Condition for the manual initiation train inoperable or a radiation monitor channel inoperable, the remaining CREATS actuation instrumentation is adequate to perform the control room protection function but the actuation time or responsiveness of the CREATS may be affected. In this Condition for the automatic actuation logic train inoperable or all radiation monitor channels inoperable, the CREATS is not capable of performing its intended automatic function. This is considered a loss of safety function. The CREATS, however, may still be capable of being placed in CREATS Mode F by manual operator actions. The 1 hour Completion Time is based on the low probability of a DBA occurring during this time frame, and the ability of the CREATS dampers to automatically isolate the control room or be manually isolated by the operator.

The Required Action for Condition A is modified by a Note which allows the control room to be unisolated for ≤ 1 hour every 24 hours. This allows fresh air makeup to improve the working environment within the control room and is acceptable based on the low probability of a DBA occurring during this makeup period.

B.1 and B.2

Condition B applies when the Required Action and associated Completion Time of Condition A has not been met and the plant is in MODE 1, 2, 3, or 4. The plant must be brought to a MODE that minimizes accident risk. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

Condition C applies when the Required Action and associated Completion Time of Condition A has not been met during movement of irradiated fuel assemblies or during CORE ALTERATIONS. Movement of irradiated fuel assemblies and CORE ALTERATIONS must be suspended immediately to reduce the risk of accidents that would require CREATS actuation. This places the plant in a condition that minimizes risk. This does not preclude movement of fuel or other components to a safe position.

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 CREATS Actuation Functions.

SR 3.3.6.1

This SR is the performance of a COT once every 92 days on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide the automatic CREATS actuation. The setpoints shall be left consistent with the plant specific calibration procedure tolerance. The Frequency of 92 days is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.

SR 3.3.6.2

This SR is the performance of a TADOT of the Manual Initiation Function every 24 months. The Manual Initiation Function is tested up to, and including, the master relay coils.

The Frequency of 24 months is based on the known reliability of the Function and the redundancy available, and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints because the Manual Initiation Function has no setpoints.

SR 3.3.6.3

This SR is the performance of a CHANNEL CALIBRATION every 24 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.

The Frequency of 24 months is based on operating experience and is consistent with the typical industry refueling cycle.

SR 3.3.6.4

This SR is the performance of an ACTUATION LOGIC TEST. All possible logic combinations are tested for the CREATS actuation instrumentation. In addition, the master relay is tested for continuity. This verifies that the logic modules are OPERABLE and there is an intact voltage signal path to the master relay coils. This test is acceptable based on instrument reliability and operating experience.

REFERENCES

1. UFSAR, Section 6.4.
 2. Letter from Robert C. Mecredy, RG&E, to Guy S. Vissing, NRC, Subject: Application for Amendment to Facility Operating License Control Room Emergency Air Treatment System CREATS Applicability Change (LCO 3.3.6 and LCO 3.7.9), dated July 21, 2000.
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BASES

BACKGROUND
(continued)

Alternative means also exist to monitor RCS LEAKAGE inside containment. These include humidity detectors, air temperature and pressure monitoring, and condensate flow rate from the air coolers. The capability of these systems to detect RCS leakage is influenced by several factors including containment free volume and detector location. These systems are most useful as alarms or indirect indicating devices available to the operators and are not required by this LCO (Ref. 2).

The leakage detection systems are also used to support identification of leakage from open systems found in containment. This includes service water and fire service water systems. Leakage from these systems is required to be monitored in response to IE Bulletin No. 80-24 (Ref. 3).

APPLICABLE
SAFETY ANALYSES

The asymmetric loads produced by the postulated breaks are the result of an assumed pressure imbalance, both internal and external to the RCS. The internal asymmetric loads result from a rapid decompression that cause large transient pressure differentials across the core barrel and fuel assemblies. The external asymmetric loads result from the rapid depressurization of annulus regions, such as the annulus between the reactor vessel and the shield wall, and cause large transient pressure differentials to act on the vessel. These asymmetric loads could damage RCS supports, core cooling equipment or core internals. This concern was first identified as Multiplant Action (MPA) D-10 and subsequently as Unresolved Safety Issue (USI) 2, "Asymmetric LOCA Loads" (Ref. 4).

The resolution of USI-2 for Westinghouse PWRs was use of fracture mechanics technology for RCS piping > 10 inches diameter (Ref. 5). This technology became known as leak-before-break (LBB). Included within the LBB methodology was the requirement to have leakage detection systems capable of detecting a 1.0 gpm leak within four hours. This leakage rate is designed to ensure that adequate margins exist to detect leaks in a timely manner during normal operating conditions. The use of LBB for the Ginna Station RCS is documented in Reference 6.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

The LBB methodology was further expanded to include the portions of the residual heat removal (RHR) system piping from its connection to the RCS hot and cold leg piping to the first motor operated isolation valve. The specific application of LBB to this piping was reviewed by the NRC and the staff concluded that, because the appropriate margins on leakage and crack size have been met given the Ginna leakage detection system capability of 0.25 gpm, it has been demonstrated that these sections of piping will exhibit LBB behavior.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the plant and the public. Required corrective actions are provided in LCO 3.4.13, RCS Operational LEAKAGE. The capability of the leakage detection systems was evaluated by the NRC in References 7 and 8.

RCS leakage detection instrumentation satisfies Criterion 1 of the NRC Policy Statement.

LCO

One method of protecting against large RCS LEAKAGE derives from the ability of instruments to rapidly detect extremely small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition, when RCS LEAKAGE indicates possible RCPB degradation.

(continued)

BASES

LCO
(continued)

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment sump A monitor (level or pump actuation from either sump A pump), in combination with a gaseous (R-12) and particulate (R-11) radioactivity monitor provides an acceptable minimum. Alternatively, the plant vent gaseous (R-14) or particulate (R-13) monitors may be used in place of R-12 and R-11, respectively, provided that a flowpath through normally closed valve 1590 is available and R-14A is OPERABLE.

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature is $\leq 200^{\circ}\text{F}$ and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

A.1.1, A.1.2, and A.2

With the required containment sump A monitor inoperable, no other form of sampling can provide the equivalent information; however, the containment atmosphere radioactivity monitors will provide indications of changes in leakage. In addition to OPERABLE gaseous and particulate atmosphere monitors, the containment air cooler condensate collection system must be verified to be OPERABLE within 24 hours, or the periodic surveillance for RCS water inventory balance, SR 3.4.13.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage.

(continued)

BASES

ACTIONS

A.1.1, A.1.2, and A.2 (continued)

The containment air cooler condensate collection system is OPERABLE if the flow paths from all four containment air coolers to their respective collection tanks are available and a CHANNEL CALIBRATION of the monitor has been performed within the last 24 months. The containment air cooler condensate collection system is provided as an option for detecting RCS leakage since SR 3.4.13.1 is not performed until after 12 hours of steady state operation. Therefore, this collection system can be used during MODE changes if the containment sump monitor is inoperable to meet the LCO.

Restoration of the required sump monitor to OPERABLE status within a Completion Time of 30 days is required to regain the function after the monitor's failure. This time is acceptable, considering the Frequency and adequacy of the RCS water inventory balance required by Required Action A.1.

Required Actions A.1.1, A.1.2, and A.2 are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the containment sump monitor is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

B.1

With the gaseous (R-12) containment atmosphere radioactivity monitoring instrumentation channel inoperable (and its alternate R-14), a verification that the particulate (R-11) containment atmosphere radioactivity monitor is OPERABLE is required. The 1 hour Completion Time is based on the low probability of a RCS leak occurring during this time frame.

Required Action B.1 is modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the gaseous containment atmosphere radioactivity monitor is inoperable. This allowance is provided because other instrumentation is available to monitor for RCS LEAKAGE.

(continued)

BASES

ACTIONS
(continued)

C.1 and C.2

With the particulate (R-11) containment atmosphere radioactivity monitoring instrumentation channel inoperable (and its alternate R-13) or Required Action B.1 not met, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information as the gaseous (R-12) containment atmosphere radioactivity monitor can only measure between a 2.0 and 10.0 gpm leak within one hour and the containment sump monitor can only measure a 2.0 gpm leak within one hour.

The 12 hour interval provides periodic information that is adequate to detect leakage and recognizes that at least one other form of leakage detection is available.

Required Actions C.1 and C.2 are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the particulate containment atmosphere radioactivity monitor is inoperable. This allowance is provided because other instrumentation is available to monitor for RCS LEAKAGE.

D.1 and D.2

With the gaseous (R-12) and the particulate (R-11) containment atmosphere radioactivity monitor inoperable, the only installed means of detecting leakage is the containment sump monitor. This condition does not provide a diverse means of leakage detection. Also, the sump monitor can only measure a 2.0 gpm leak within 1 hour.

In addition to the Required Actions of Conditions B and C, restoration of either of the inoperable monitors to OPERABLE status within 30 days is required to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a reduced configuration for a lengthy period of time.

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

Required Actions D.1 and D.2 are modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the gaseous and particulate containment atmosphere radioactivity monitors are inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

E.1 and E.2

If a Required Action of Condition A, C, or D cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant 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 plant conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1

With all required monitors inoperable, no automatic means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE
REQUIREMENTS

SR 3.4.15.1

This SR requires the performance of a CHANNEL CHECK of the containment atmosphere radioactivity monitors. The check gives reasonable confidence that the channels are operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS
(continued)

SR 3.4.15.2

This SR requires the performance of a CHANNEL OPERATIONAL TEST (COT) on the containment atmosphere radioactivity monitors. The test ensures that the monitors can perform their function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

SR 3.4.15.3 and SR 3.4.15.4

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 24 months considers channel reliability and operating experience has proven that this Frequency is acceptable.

REFERENCES

1. Atomic Industry Forum (AIF) GDC 16 and 34, Issued for comment July 10, 1967.
2. Regulatory Guide 1.45.
3. IE Bulletin No. 80-24, "Prevention of Damage Due to Water Leakage Inside Containment."
4. NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems," 1981.
5. Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops."
6. Letter from D. C. DiIanni, NRC, to R. W. Kober, RG&E, Subject: "Generic Letter 84-04," dated September 9, 1985.
7. NUREG-0821, "Integrated Plant Safety Assessment, Systematic Evaluation Program, R. E. Nuclear Power Plant," December 1982.

(continued)

BASES

REFERENCES

(continued)

8. Letter from Guy S. Vissing (NRC) to Robert C. Mecredy (RG&E), "Staff Review of the Submittal by Rochester Gas and Electric Company to Apply Leak-Before-Break Status to Portions of the R.E. Ginna Nuclear Power Plant Residual Heat Removal System Piping", dated February 25, 1999.
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B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred (but non safety related) heat sink, provided by the condenser and circulating water system, is not available.

Four MSSVs are located on each main steam header, outside containment in the Intermediate Building, upstream of the main steam isolation valves (Ref. 1). MSSVs 3509, 3511, 3513, and 3515 are located on the steam generator (SG) A main steam header while MSSVs 3508, 3510, 3512 and 3514 are located on the SG B main steam header. The MSSVs are designed to limit the secondary system to $\leq 110\%$ of design pressure when passing 100% of design flow. The MSSV design includes staggered setpoints so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine/ reactor trip.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs is to limit the secondary system pressure to $\leq 110\%$ of design pressure when passing 100% of design steam flow. This design basis is sufficient to cope with any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis. The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased RCS heat removal events (Ref. 2). Of these, the full power loss of external load event is the limiting AOO. This event also results in the loss of normal feedwater flow to the SGs.

The transient response for a loss of external load event without a direct reactor trip (i.e., loss of load when $< 50\%$ RTP) presents no hazard to the integrity of the RCS or the Main Steam System. For transients at power levels $> 50\%$, the effect on RCS safety limits is evaluated with no credit taken for the pressure relieving capability of pressurizer spray, the steam dump system, and the SG atmospheric relief valves. The reactor is tripped on high pressurizer pressure with the pressurizer safety valves and MSSVs required to be opened to maintain the RCS and Main Steam System within 110% of their design values.

The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening (as an initiating event only), and failure to reclose once opened. The passive failure mode is failure to open upon demand which is not considered in the accident analyses.

The MSSVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

The accident analysis requires four MSSVs per steam generator to provide overpressure protection for design basis transients occurring at 102% RTP. The OPERABILITY of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve SG overpressure, and reseal when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

The lift settings, according to SR 3.7.1.1 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB or secondary system.

APPLICABILITY

In MODES 1, 2, and 3, four MSSVs per SG are required to be OPERABLE to ensure that the RCS remains within its pressure safety limit and that the secondary system, from the SGs to the main steam isolation valves, is limited to $\leq 110\%$ of design pressure for all DBAs.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The SGs are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1

With one or more MSSVs inoperable, the assumptions used in the accident analysis for loss of external load may no longer be valid and the safety valve(s) must be restored to OPERABLE status within 4 hours. This Condition specifically addresses the appropriate ACTIONS to be taken in the event that a non-significant discrepancy related to the MSSVs is discovered with the plant operating in MODES 1, 2, or 3. Examples of this type of discrepancy include administrative (e.g., documentation of inspection results) or similar deviations which do not result in a loss of MSSV capability to relieve steam. The 4 hour Completion Time allows a reasonable period of time for correction of administrative only problems or for the plant to contact the NRC to discuss appropriate action. The 4 hour Completion time is based on engineering judgement.

This Condition is not applicable to a situation in which the ability of a MSSV to open or reclose is questionable. In this event, this Condition is no longer applicable and Condition B of this LCO should be entered immediately since no corrective actions can be implemented during MODES 1, 2, and 3.

B.1 and B.2

If the MSSV(s) cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 3), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1987 (Ref. 4). According to Reference 4, the following tests are required:

- a. Visual examination;
- b. Seat tightness determination;
- c. Setpoint pressure determination (lift setting);
- d. Compliance with owner's seat tightness criteria; and
- e. Verification of the balancing device integrity on balanced valves.

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. This SR allows a +1% and -3% setpoint tolerance for OPERABILITY; however, the valves are reset to $\pm 1\%$ during the Surveillance to allow for drift.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. UFSAR, Section 10.3.2.4.
 2. UFSAR, Section 15.2.
 3. ASME, Boiler and Pressure Vessel Code, Section XI.
 4. ANSI/ASME OM-1-1987.
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B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs) and Non-Return Check Valves

BASES

BACKGROUND

The MSIVs (3516 and 3517) isolate steam flow from the secondary side of the steam generators (SGs) following a Design Basis Accident (DBA). MSIV closure is necessary to isolate a SG affected by a steam generator tube rupture (SGTR) event or a steam line break (SLB) to stop the loss of SG inventory and to protect the integrity of the unaffected SG for decay heat removal. The MSIVs are air operated swing disk check valves that are held open by an air operator against spring pressure. The MSIVs are installed to use steam flow to assist in the closure of the valve (Ref. 1).

A MSIV is located in each main steam line header outside containment in the Intermediate Building. The MSIVs are downstream from the main steam safety valves (MSSVs) and turbine driven auxiliary feedwater (AFW) pump steam supply, to assure the MSSVs prevent overpressure on the secondary side and assure steam is available to the AFW system following MSIV closure. Closing the MSIVs isolates each SG from the other, and isolates the turbine, steam dump system, and other auxiliary steam supplies from the SGs.

The MSIVs close on a main steam isolation signal generated by either high containment pressure, high steam flow coincident with low T_{avg} and safety injection (SI), or high-high steam flow coincident with SI.

The MSIVs are designed to work with non-return check valves (3518 and 3519) located immediately downstream of each MSIV to preclude the blowdown of more than one SG following a SLB. The MSIVs fail closed on loss of control or actuation power and loss of instrument air once the air is bled off from the supply line. The MSIVs may also be actuated manually.

Each MSIV has a normally closed manual MSIV bypass valve.

APPLICABLE
SAFETY
ANALYSES

The design basis of the MSIVs and non-return check valves is established by the large SLB (Ref. 2). The SLB is evaluated for two cases, one with respect to reactor core response and the second with respect to containment integrity. The SLB for reactor core response is evaluated assuming initial conditions and single failures which have the highest potential for power peaking or departure from nucleate boiling (DNB). The most limiting single failure for this evaluation is the loss of a safety injection pump which reduces the rate of boron injection into the Reactor Coolant System (RCS) delaying the return to subcriticality. The MSIV on the intact SG for this case is assumed to close to prevent excessive cooldown of the RCS which could result in a lower DNB ratio.

The SLB for containment integrity is evaluated assuming initial conditions and single failures which result in the addition of the largest amount of mass and energy into containment. For this scenario, offsite power is assumed to be available and reactor power is below 100% RTP. With offsite power available, the reactor coolant pumps continue to circulate coolant maximizing the RCS cooldown. At lower power levels, the SG inventory and temperature are at their greatest, which maximizes the analyzed mass and energy release to containment. Due to the non-return check valve on the faulted SG, reverse flow from the steam headers downstream of the MSIV and from the intact SG is prevented from contributing to the energy and mass released inside containment by the SLB. This check valve is a passive device which is not assumed to fail.

SLBs outside of containment can occur in the Intermediate Building and downstream of the MSIVs in the Turbine Building. A SLB in piping > 6 inches diameter in the Intermediate Building is not required to be considered due to an augmented piping inspection program (Ref. 3). For a SLB in the Turbine building, the MSIVs on both SGs must close to isolate the break and terminate the event.

The MSIVs are also credited in a SGTR to manually isolate the SG with the ruptured tube. In addition to minimizing the radiological releases, this assists the operator in isolating the RCS flow through the ruptured SG by preventing the SG from continuing to depressurize and creating a higher pressure difference between the secondary system and the primary system.

The MSIVs are also considered in other DBAs such as the feedwater line break in which closure of the MSIV on the intact SG maximizes the effect of the break since the energy removal capability of the intact SG would be reduced with respect to long term heat removal.

In addition to providing isolation of a faulted SG during a SLB, feedwater line break, or a SGTR, the MSIVs also serve as a containment isolation boundary. The MSIVs are the second containment isolation boundary for the main steam line penetrations which use the steam lines and SGs inside containment as the first boundary. The MSIVs do not receive an automatic containment isolation signal since a spurious signal could result in a significant plant transient.

The MSIVs and non-return check valves satisfy Criterion 3 of the NRC Policy Statement.

LCO

This LCO requires that two MSIVs and the non-return check valves in the steam lines be OPERABLE. The MSIVs are considered OPERABLE when their isolation times are within limits and they can close on an isolation actuation signal. A MSIV must also be capable of isolating a SG for containment isolation purposes. The non-return check valves are considered OPERABLE when they are capable of closing.

This LCO provides assurance that the MSIVs and non-return check valves will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. 4) limits.

APPLICABILITY

The MSIVs and non-return check valves must be OPERABLE in MODES 1, 2, and 3 when there is significant mass and energy in the RCS and SGs to challenge the integrity of containment, or allow a transient to approach DNBR limits. When the MSIVs are closed and de-activated in MODES 2 and 3, they are already performing their safety function and the MSIVs and their associated non-return check valves are not required to be OPERABLE per this LCO.

In MODE 4, the MSIVs and non-return check valves are normally closed, and the RCS and SG energy is low. In MODE 5 or 6, the SGs do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs and non-return check valves are not required for isolation of potential main steam pipe breaks in these MODES.

ACTIONS

A.1

With one or more valves inoperable in flow path from a SG in MODE 1, ACTION must be taken to restore OPERABLE status within 8 hours. Some repairs to these valves can be made with the plant under hot conditions. The 8 hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs and non-return check valves and the ability to isolate the affected SG by turbine stop valves.

The 8 hour Completion Time is greater than that normally allowed for containment isolation boundaries because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from most other containment isolation boundaries in that the closed system provides an additional means for containment isolation. Failure of this closed system can only result from a SGTR which is not postulated to occur with any other DBA (e.g., LOCA).

B.1

If the MSIV and/or non-return check valve from a SG cannot be restored to OPERABLE status within 8 hours, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in MODE 2 within 6 hours and Condition C would be entered. The Completion Time is reasonable, based on operating experience, to reach MODE 2 in an orderly manner without challenging plant systems.

C.1 and C.2

Since the MSIVs and non-return check valve are required to be OPERABLE in MODES 2 and 3, the inoperable valve(s) may either be restored to OPERABLE status or the associated MSIV closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis and the non-return check valve is no longer required.

The 8 hour Completion Time is consistent with that allowed in Condition A.

For inoperable valves that cannot be restored to OPERABLE status within the specified Completion Time, but the associated MSIV is closed, the MSIV must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgement, in view of MSIV status indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

D.1 and D.2

If the MSIVs and/or non-return check valve cannot be restored to OPERABLE status or the associated MSIV is not closed within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from MODE 2 conditions in an orderly manner without challenging plant systems.

E.1

If one or more valves in the flow path from each SG are inoperable, the plant is in a condition outside of the accident analyses; therefore, LCO 3.0.3 must be entered immediately. This Condition must be entered when any combination of MSIVs and non-return check valves are inoperable such that at least one valve is inoperable in each of the two main steam flow paths.

SURVEILLANCE
REQUIREMENTS

SR 3.7.2.1

This SR verifies that MSIV closure time is ≤ 5 seconds under no flow and no load conditions. The MSIVs are swing-disk check valves that are held open by their air operators against spring pressure. Once the MSIVs begin to close during hot conditions, the steam flow will assist the valve closure such that testing under no flow and no load conditions is conservative. The 5 second closure time is consistent with the expected response time for instrumentation associated with the MSIV and the accident analysis assumptions.

As the MSIVs are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 5), requirements during operation in MODE 1, 2, or 3. The Frequency is in accordance with the Inservice Testing Program.

SR 3.7.2.2

This SR verifies that each main steam non-return check valve can close. As the non-return check valves are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 5), requirements during operation in MODE 1, 2, or 3. The Frequency is in accordance with the Inservice Testing Program.

SR 3.7.2.3

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. The MSIVs should not be tested at power, since even a partial stroke exercise increases the risk of a valve closure and plant transient when the plant is above MODE 4. As the MSIVs are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 5), requirements during operation in MODES 1, 2 and 3.

The frequency of MSIV testing is every 24 months. The 24 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 5.4.4.
 2. UFSAR, Section 15.1.5.
 3. UFSAR, Section 3.6.2.5.1.
 4. 10 CFR 100.11.
 5. ASME, Boiler and Pressure Vessel Code, Section XI.
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B 3.7 PLANT SYSTEMS

B 3.7.3 Main Feedwater Regulating Valves (MFRVs), Associated Bypass Valves, and Main Feedwater Pump Discharge Valves (MFPDVs)

BASES

BACKGROUND

The MFRVs (4269 and 4270) and their associated bypass valves (4271 and 4272), and MFPDVs (3977 and 3976) isolate main feedwater (MFW) flow to the secondary side of the steam generators (SGs) following a Design Basis Accident (DBA). The safety related function of the MFRVs, associated bypass valves, and MFPDVs is to provide for isolation of MFW flow to the secondary side of the SGs terminating the DBA for line breaks occurring downstream of the valves. Closure effectively terminates the addition of feedwater to an affected SG, limiting the mass and energy release for steam line breaks (SLBs) or feedwater line breaks (FWLBs) inside containment, and reducing the cooldown effects for SLBs.

The MFRVs, associated bypass valves, and MFPDVs in conjunction with check valves located downstream of the isolation valves also provide a pressure boundary for the controlled addition of auxiliary feedwater (AFW) to the intact SG (see Figure B 3.7.3-1).

One MFPDV is located in the Turbine Building on the discharge line of each MFW pump (Ref. 1). One MFRV and associated bypass valve is located on each MFW line to its respective SG, outside containment in the Turbine Building. The MFRVs, associated bypass valves, and MFPVs are located upstream of the AFW injection point so that AFW may be supplied to the SGs following closure of the MFRVs and bypass valves. The piping volume from these valves to the SGs is accounted for in calculating mass and energy releases, and must be refilled prior to AFW reaching the SG following either an SLB or FWLB.

The MFPDV closes on the opening of the MFW pump breaker which occurs on receipt of a safety injection signal or any other signal which trips the pump breaker. The MFRVs and bypass valves close on receipt of a safety injection signal, a SG high level signal, or on a reactor trip with $T_{avg} < 554^{\circ}\text{F}$ with the associated MFRV in auto. All valves may also be actuated manually. In addition to the MFRVs, associated bypass valves and MFPDVs, a check valve located outside containment for each feedwater line is available. The check valve isolates the feedwater line penetrating containment providing a containment isolation boundary.

APPLICABLE
SAFETY
ANALYSES

The design basis of the MFRVs, associated bypass valves, and MFPDVs is established by the analyses for the SLB. The SLB is evaluated for two cases, one with respect to reactor core response and the second with respect to containment integrity (Ref. 2). The SLB for reactor core response is evaluated assuming initial conditions and single failures which have the highest potential for power peaking or departure from nucleate boiling (DNB). The most limiting single failure for this evaluation is the loss of a safety injection pump which reduces the rate of boron injection into the Reactor Coolant System (RCS) delaying the return to subcriticality. The MFRV and bypass valve on the intact SG for this case are assumed to close on a safety injection signal to prevent excessive cooldown of the RCS which could result in a lower DNB ratio. The failure of either of these valves is bounded by the eventual coastdown of the MFW pumps, which have their breakers opened by a SI signal, and the MFPDV which close on opening of the MFW pump breakers.

The SLB for containment integrity is evaluated assuming initial conditions and single failures which result in the addition of the largest amount of mass and energy into containment. For this scenario, offsite power is assumed to be available and reactor power is below 100% RTP. With offsite power available, the reactor coolant pumps continue to circulate coolant, maximizing the RCS cooldown. At lower power levels, the SG inventory and temperature are at their greatest, which maximizes the analyzed mass and energy release to containment. The MFRV and bypass valve on the faulted SG are assumed to close on a safety injection signal to prevent continued contribution to the energy and mass released inside containment by the SLB. The failure of either of these valves is bounded by the eventual coastdown of the MFW pumps and closure of the MFPDVs.

The MFRVs and bypass valves are also credited for isolation in the feedwater transient analyses (e.g., increase in feedwater flow). These valves close on either a safety injection or high SG level signal depending on the scenario. The valves also must close on a FWLB to limit the amount of additional mass and energy delivered to the SGs and containment.

The failure of the MFRVs to control flow is also considered as an initiating event. This includes consideration of a valve failure coincident with an atmospheric relief valve failure since a single component in the Advanced Digital Feedwater Control System (ADFCS) controls both components (Ref. 3). This combined valve failure accident scenario is evaluated with respect to DNB since a large RCS cooldown is possible with this combination of failures. However, this scenario is bounded by the SLB accident.

The MFRVs, associated bypass valves, and MFPDVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

This LCO ensures that the MFRVs, associated bypass valves, and MFPDVs will isolate MFW flow to the SGs, following a FWLB or SLB.

This LCO requires that two MFPDVs, two MFRVs, and two MFRV bypass valves be OPERABLE. The MFRVs, associated bypass valves, and MFPDVs are considered OPERABLE when isolation times are within limits and they can close on an isolation actuation signal.

Failure to meet the LCO requirements can result in additional mass and energy being released to containment following an SLB or FWLB inside containment. It may also result in the introduction of water into the main steam lines for an excess feedwater flow event.

APPLICABILITY

The MFRVs, associated bypass valves, and MFPDVs valves must be OPERABLE whenever there is significant mass and energy in the RCS and SGs. This ensures that, in the event of a DBA, the accident analysis assumptions are maintained. In MODES 1, 2, and 3, the MFRVs, associated bypass valves, and MFPDVs are required to be OPERABLE to limit the amount of available fluid that could be added to containment in the case of a secondary system pipe break inside containment. When the valves are closed and de-activated or isolated by a closed manual valve such that both SGs are isolated from both MFW pumps, they are already performing their safety function and no longer required to be OPERABLE.

In MODE 4, the MFRVs, associated bypass valves, and MFPDVs are normally closed since AFW is providing decay heat removal due to the low SG energy level. In MODE 5 or 6, the SGs do not contain much energy because their temperature is below the boiling point of water; therefore, the MFRVs, associated bypass valves, and MFPDVs are not required for isolation of potential pipe breaks in these MODES.

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1

With one or more MFPDV(s) inoperable, action must be taken to restore the affected valve to OPERABLE status, or close the inoperable valve within 24 hours. The 24 hour Completion Time takes into account the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 24 hour Completion Time is reasonable, based on operating experience.

An inoperable MFPDV that is closed must be verified on a periodic basis that it remains closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion time is reasonable, based on engineering judgement, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed.

B.1 and B.2

With one or more MFRV(s) inoperable, action must be taken to restore the affected valve to OPERABLE status, or to close or isolate the inoperable valve within 24 hours. The 24 hour Completion Time takes into account the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 24 hour Completion Time is reasonable, based on operating experience.

An inoperable MFRV that is closed or isolated must be verified on a periodic basis that it remains closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion time is reasonable, based on engineering judgement, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

C.1 and C.2

With one or more MFRV bypass valve(s) inoperable, action must be taken to restore the affected valve to OPERABLE status, or to close or isolate the inoperable valve within 24 hours. The 24 hour Completion Time takes into account the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The 24 hour Completion Time is reasonable, based on operating experience.

An inoperable MFRV bypass valve that is closed or isolated must be verified on a periodic basis that it remains closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

D.1 and D.2

If the MFRV, associated bypass valve, or MFPDV cannot be restored to OPERABLE status or closed within 24 hours or cannot be verified closed once per 31 days, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

If one or more MFPDV(s) and one or more MFRV(s), or one or more MFPDV(s) and one or more MFRV bypass valve(s) are inoperable, the plant is in a condition outside of the accident analyses; therefore, LCO 3.0.3 must be entered immediately. This Condition must be entered when any combination of MFRVs, associated bypass valves, or MFPDVs are inoperable such that a MFW pump, condensate pump, or condensate booster pump can provide unisolable flow to one or both SGs (see Figure B 3.7.3-1).

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each MFPDV is ≤ 80 seconds from the full open position on an actual or simulated actuation signal (i.e., from opening of MFW pump breakers). The valve closure times are assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. These valves should not be tested at power since even a partial stroke exercise increases the risk of a valve closure with the plant generating power. As these valves are not tested at power, they are exempt from the ASME Code, Section XI, (Ref. 4) requirements during operation in MODES 1, 2, and 3.

The Frequency for this SR is in accordance with the Inservice Testing Program.

SR 3.7.3.2

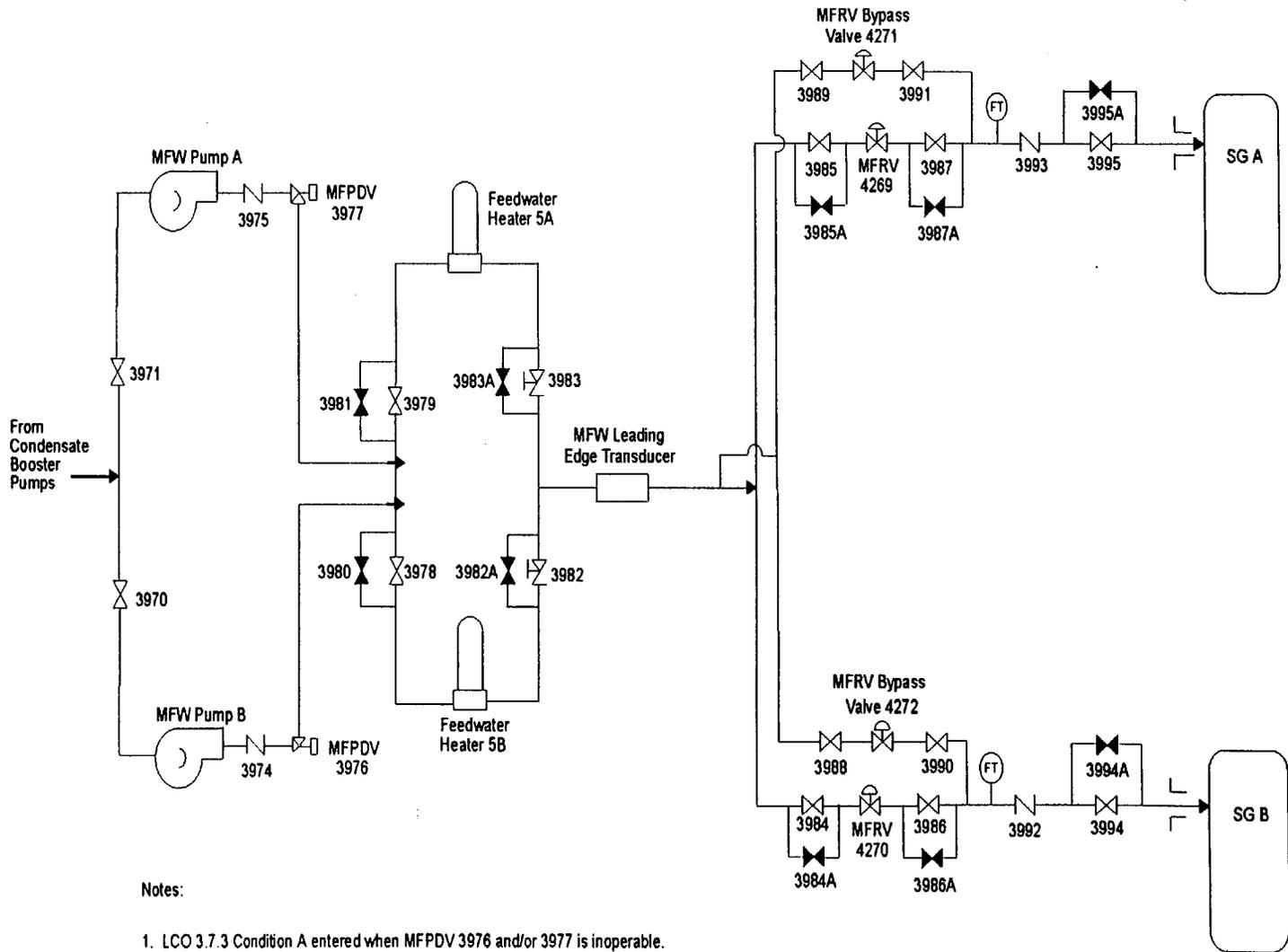
This SR verifies that the closure time of each MFRV and associated bypass valve is ≤ 10 seconds from the full open position on an actual or simulated actuation signal. The valve closure times are assumed in the accident and containment analyses. This Surveillance is normally performed upon returning the plant to operation following a refueling outage. These valves should not be tested at power since even a partial stroke exercise increases the risk of a valve closure with the plant generating power. As these valves are not tested at power, they are exempt from the ASME Code, Section XI (Ref. 4), requirements during operation in MODES 1, 2, and 3.

The Frequency for this SR is in accordance with the Inservice Testing Program.

REFERENCES

1. UFSAR, Section 10.4.5.3.
 2. UFSAR, Section 15.1.5.
 3. UFSAR, Section 15.1.6.
 4. ASME, Boiler and Pressure Vessel Code, Section XI.
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Figure B 3.7.3-1
MFRVs, Associated Bypass Valves and MFPDVs



Notes:

1. LCO 3.7.3 Condition A entered when MFPDV 3976 and/or 3977 is inoperable.
2. LCO 3.7.3 Condition B entered when MFRV 4269 and/or 4270 is inoperable.
3. LCO 3.7.3 Condition C entered when MFRV Bypass Valve 4271 and/or 4272 is inoperable.
4. LCO 3.7.3 Condition E entered when any combination of valve inoperabilities results in an unisolable flowpath from the condensate booster pumps to one or more SGs.

For illustration only

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Relief Valves (ARVs)

BASES

BACKGROUND

There is an ARV (3410 and 3411) located on the main steam header from each steam generator (SG). The ARVs are provided with upstream block valves to permit their being tested at power, and to provide an alternate means of isolation. The ARVs have two functions (Ref. 1):

- a. provide secondary system overpressure protection below the setpoint of the main steam safety valves (MSSVs); and
- b. provide a method for cooling the plant should the preferred heat sink via the steam dump system to the condenser not be available.

The accident analyses do not credit either of these functions since the ARVs do not have a safety-related source of motive air and the accident analyses do not typically require cooldown to the residual heat removal entry conditions since the plant was originally designed to maintain Hot Shutdown conditions indefinitely. The only exception is with respect to steam generator tube rupture (SGTR) events which require the use of at least one ARV to provide heat removal from the Reactor Coolant System (RCS) to prevent saturation conditions from developing.

The ARVs are air operated valves located in the Intermediate Building with a relief capacity of 329,000 lbm/hr each (approximately 5% of RTP). The ARVs are normally closed, fail closed valves which receive motive air from the instrument air system. The valves can also receive motive air from a non-seismic backup nitrogen bottle bank system. The valves are equipped with pneumatic controllers to permit control of the cooldown rate. The ARVs are normally controlled by the Advanced Digital Feedwater Control System (ADFCS) but can also be remote manually operated and opened locally by use of handwheels located on the valves.

**APPLICABLE
SAFETY
ANALYSES**

The design basis for the ARVs is established by the SGTR event (Ref. 2). For this accident scenario, the operator is required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured SG. Following a SGTR, the MSSVs will maintain the secondary system pressure at approximately 1085 psig which could result in the loss of subcooling margin since the RCS average temperature is attempting to stabilize at approximately 547°F. The ARVs are used during the first 30 to 60 minutes of the SGTR to continue the RCS cooldown in an effort to reduce, and eventually terminate, the primary to secondary system flow in the ruptured SG. The inability to cooldown could result in inadequate subcooling margin which would delay the termination of the leakage through the ruptured tube.

The opening of the ARVs is also considered coincident with a failure of a main feedwater regulating valve (Ref. 3) since a single component in the ADFCS controls both components. This combined valve failure accident scenario is evaluated with respect to departure from nucleate boiling since a large RCS cooldown is possible with this combination of failures. However, this scenario is bounded by the steam line break accident.

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

The ARVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

Two ARVs and their associated manual block valves are required to be OPERABLE. The ARVs are required for manual operation either locally (using the handwheel) or remotely to relieve main steam pressure. The ARV block valves must be OPERABLE to isolate a failed open ARV. A closed block valve does not render it or its ARV line inoperable if operator ACTION time to open the block valve can be accomplished within the time frames specified below. Failure to meet the LCO can result in the inability to cool the plant following a SGTR event in which the condenser is unavailable for use with the steam dump system.

An ARV line is considered OPERABLE when it is capable of being manually opened within 20 minutes of determining the need to utilize the ARV following a SGTR. The ARV line must also be capable of closing within 15 minutes in the event the ARV spuriously opens on the SG with the ruptured tube. Finally, the ARV line must be capable of closing within 5 minutes in the event that the ARV on the intact SG fails to close following initiation of a cooldown. For the closure requirements, either the ARV or its associated block valve may be credited for OPERABILITY.

APPLICABILITY In MODES 1 and 2, and in MODE 3 with RCS average temperature $\geq 500^{\circ}\text{F}$, the ARV lines are required to be OPERABLE.

In MODE 3 with RCS average temperature $< 500^{\circ}\text{F}$, and in MODE 4, the ARVs are not required since the saturation pressure of the reactor coolant is below the lift settings of the MSSVs. In MODE 5 or 6, an SGTR is not a credible event since the water in the SGs is below the boiling point and RCS pressure is low.

ACTIONS

A.1

With one ARV line inoperable, ACTION must be taken to restore the valve to OPERABLE status within 7 days. The 7 day Completion Time allows for the redundant capability afforded by the remaining OPERABLE ARV line and a nonsafety grade backup in the steam dump system.

Required Action A.1 is modified by a Note indicating that LCO 3.0.4 does not apply since the steam dump system would normally be in service during lower MODES of operation and can provide an acceptable alternative to the inoperable ARV line.

B.1

If the ARV line cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 with RCS average temperature $< 500^{\circ}\text{F}$ within 8 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

If both ARV lines are inoperable, the plant is in a condition outside of the accident analyses for a SGTR event; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE
REQUIREMENTS

SR 3.7.4.1

To perform a cooldown of the RCS, the ARVs must be able to be opened either remotely or locally. 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 plant cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. The Frequency is acceptable from a reliability standpoint.

SR 3.7.4.2

The function of the block valve is to isolate a failed open ARV. Cycling the block valve both closed and open demonstrates its capability to perform this function. Performance of inservice testing or use of the block valve during plant cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. The Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 10.3.2.5.
 2. UFSAR, Section 15.6.3.
 3. UFSAR, Section 15.1.6.
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B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System supplies feedwater to the steam generators (SGs) to remove decay heat from the Reactor Coolant System (RCS) upon the loss of normal feedwater supply. The SGs function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the SGs via the main steam safety valves (MSSVs) or atmospheric relief valves (ARVs). If the main condenser is available, steam may be released via the steam dump valves. The AFW System is comprised of two separate systems, a preferred AFW System and a Standby AFW (SAFW) System (Ref. 1).

AFW System

The preferred AFW System consists of two motor driven AFW (MDAFW) pumps and one turbine driven AFW (TDAFW) pump configured into three separate trains which are all located in the Intermediate Building (see Figure B 3.7.5-1). Each MDAFW train provides 100% of AFW flow capacity, and the TDAFW pump provides 200% of the required capacity to the SGs, as assumed in the accident analysis. The pumps are equipped with independent recirculation lines to the condensate storage tanks (CSTs). Each MDAFW train is powered from an independent Class 1E power supply and feeds one SG, although each pump has the capability to be realigned from the control room to feed the other SG via cross-tie lines containing normally closed motor operated valves (4000A and 4000B). The two MDAFW trains will actuate automatically on a low-low level signal in either SG, opening of the main feedwater (MFW) pump breakers, a safety injection (SI) signal, or the ATWS mitigation system actuation circuitry (AMSAC). The pumps can also be manually started from the control room.

The TDAFW pump receives steam from each main steam line upstream of the two main steam isolation valves. Either of the steam lines will supply 100% of the requirements of the TDAFW pump. The TDAFW pump supplies a common header capable of feeding both SGs by use of fail-open, air-operated control valves (4297 and 4298). The TDAFW pump will actuate automatically on a low-low level signal in both SGs, loss of voltage on 4160 V Buses 11A and 11B, or the ATWS mitigation system actuation circuitry (AMSAC). The pump can also be manually started from the control room.

The normal source of water for the AFW System is the CSTs which are located in the non-seismic Service Building. The Service Water (SW) System (LCO 3.7.8) can also be used to supply a safety-related source of water through normally closed motor operated valves (4013, 4027, and 4028) which supply each AFW train.

SAFW System

The SAFW System consists of two motor driven pumps configured into two separate trains (see Figure B 3.7.5-2). Each motor driven SAFW train provides 100% of the AFW flow capacity as assumed in the accident analyses and supplies one SG through the use of a normally open motor-operated stop check valve. Each pump has the capability to be realigned from the control room to feed the other SG via normally closed motor operated valves (9703A and 9703B). Each pump is powered from an independent Class 1E power supply and can be powered from the diesel generators provided that the breaker for the associated MDAFW pump is opened. The safety-related source of water for the SAFW System is the SW System through two normally closed motor operated valves (9629A and 9629B). Condensate can also be supplied by a 10,000 gallon condensate test tank and the yard fire hydrant yard loop.

The SAFW System is manually actuated in the event that the preferred AFW System has failed due to a high energy line break (HELB) in the Intermediate Building, a seismic or fire event. The SAFW trains are located in the SAFW Pump Building located adjacent to the Auxiliary Building.

The SAFW Pump Building environment is controlled by room coolers which are supplied by the same SW header as the pump trains. These coolers are required when the outside air temperature is $\geq 80^{\circ}\text{F}$ to ensure the SAFW Pump Building remains $\leq 120^{\circ}\text{F}$ during accident conditions.

The AFW System is designed to supply sufficient water to the SG(s) to remove decay heat with SG pressure at the lowest MSSV set pressure plus 1%. Subsequently, the AFW System supplies sufficient water to cool the plant to RHR entry conditions, with steam released through the ARVs.

APPLICABLE
SAFETY
ANALYSES

The design basis of the AFW System is to supply water to the SG(s) to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the SGs at pressures corresponding to the lowest MSSV set pressure plus 1%.

The AFW System mitigates the consequences of any event with the loss of normal feedwater. The limiting Design Basis Accidents (DBAs) and transients for the AFW System are as follows (Ref. 2):

- a. Feedwater Line Break (FWLB);
- b. Loss of MFW (with and without offsite power);
- c. Steam Line Break (SLB);
- d. Small break loss of coolant accident (LOCA);
- e. Steam generator tube rupture (SGTR); and
- f. External events (tornados and seismic events).

AFW is also used to mitigate the effects of an ATWS event which is a beyond design basis event not addressed by this LCO.

The AFW System design is such that any of the above DBAs can be mitigated using the preferred AFW System or SAFW System. For the FWLB, SLB, and external events DBAs (items a, c, and f), the worst case scenario is the loss of all three preferred AFW trains due to a HELB in the Intermediate or Turbine Building, or a failure of the Intermediate Building block walls. For these three events, the use of the SAFW System within 10 minutes is assumed by the accident analyses. Since a single failure must also be assumed in addition to the HELB or external event, the capability of the SAFW System to supply flow to an intact SG could be compromised if the SAFW cross-tie is not available. For HELBs within containment, use of either the SAFW System or the AFW System to the intact SG is assumed within 10 minutes.

For the SGTR events (item e), the accident analyses assume that one AFW train is available upon a SI signal or low-low SG level signal. Additional inventory is being added to the ruptured SG as a result of the SGTR such that AFW flow is not a critical feature for this DBA.

The loss of MFW (item b) is a Condition 2 event (Ref. 3) which places limits on the response of the RCS from the transient (e.g., no challenge to the pressurizer power operated relief valves is allowed). This analysis has been performed assuming no AFW flow is available until 10 minutes with acceptable results. The most limiting small break LOCA (item d) analysis has also been performed assuming no AFW flow with no adverse impact on peak cladding temperature.

In summary, all limiting DBAs and transients have been analyzed assuming a 10 minute delay for actuation of flow.

In addition to its accident mitigation function, the energy and mass addition capability of the AFW System is also considered with respect to HELBs within containment. For SLBs and FWLBs within containment, maximum pump flow from all three AFW pumps is assumed for 10 minutes until operations can isolate the flow by tripping the AFW pumps or by closing the respective pump discharge flow path(s). Therefore, the motor operated discharge isolation valves for the motor MDAFW pump trains (4007 and 4008) are designed to limit flow to ≤ 230 gpm to limit the energy and mass addition so that containment remains within design limits for items a and c. The TDAFW train is assumed to be at runout conditions (i.e., 600 gpm).

The AFW System satisfies the requirements of Criterion 3 of the NRC Policy Statement.

LCO

This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of accidents that could result in overpressurization of the reactor coolant pressure boundary or containment.

The AFW System is comprised of two systems which are configured into five trains. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the SGs are OPERABLE (see Figures B 3.7.5-1 and B 3.7.5-2). This requires that the following be OPERABLE:

- a. Two MDAFW trains taking suction from the CSTs as required by LCO 3.7.6 (and capable of taking suction from the SW system within 10 minutes), and capable of supplying their respective SG with ≥ 200 gpm within 10 minutes and ≤ 230 gpm upon AFW actuation (on a per pump basis);
- b. The TDAFW train taking suction from the CSTs as required by LCO 3.7.6 (and capable of taking suction from the SW system within 10 minutes), provided steam is available from both main steam lines upstream of the MSIVs, and capable of supplying both SGs with ≥ 200 gpm each within 10 minutes; and
- c. Two motor driven SAFW trains capable of being initiated either locally or from the control room within 10 minutes, taking suction from the SW System, and supplying their respective SG and the opposite SG through the SAFW cross-tie line with ≥ 200 gpm.

The piping, valves, instrumentation, and controls in the required flow paths are also required to be OPERABLE. The TDAFW train is comprised of a common pump and two flow paths. A TDAFW train flow path is defined as the steam supply line and the SG injection line from/to the same SG. The failure of the pump or both flow paths renders the TDAFW train inoperable.

The cross-tie line for the preferred MDAFW pumps is not required for this LCO. However, since the accident analyses have been performed assuming a 10 minute delay for AFW, and there are two separate systems, the use of this cross-tie line is allowed in MODES 1, 2, and 3. Also, provided that the AFW and SAFW discharge valves are set to provide the minimum required flow, the recirculation lines for the preferred AFW system and SAFW system pumps are not credited in the accident analysis. The recirculation lines are also not required to be OPERABLE for this LCO since the MSSVs maintain the SG pressure below the pump's shutoff head.

The SAFW Pump Building room coolers are required to be OPERABLE when the outside air temperature is $\geq 80^{\circ}\text{F}$. If one room cooler is inoperable, the associated SAFW train is inoperable.

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE in the event that it is called upon to function when the MFW System is lost. In addition, the AFW System is required to supply enough makeup water to replace the lost SG secondary inventory as the plant cools to MODE 4 conditions.

In MODE 4, 5, or 6, the SGs are not normally used for heat removal, and the AFW System is not required.

ACTIONS

A.1

If one of the TDAFW train flow paths is inoperable, action must be taken to restore the flow path to OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

- a. The redundant OPERABLE turbine driven AFW pump flow path;
- b. The availability of redundant OPERABLE MDAFW and SAFW pumps; and
- c. The low probability of an event occurring that requires the inoperable TDAFW pump flow path.

A TDAFW train flow path is defined as the steam supply line and SG injection line from/to the same SG.

B.1

If one MDAFW train is inoperable, action must be taken to restore the train to OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

- a. The redundant OPERABLE MDAFW train;
- b. The availability of redundant OPERABLE TDAFW and SAFW pumps; and
- c. The low probability of an event occurring that requires the inoperable MDAFW train.

C.1

With the TDAFW train inoperable, or both MDAFW trains inoperable, or one TDAFW train flow path and one MDAFW train inoperable to opposite SGs, action must be taken to restore OPERABLE status within 72 hours. If the inoperable MDAFW train supplies the same SG as the inoperable TDAFW flow path, Condition D must be entered.

The combination of failures which requires entry into this Condition all result in the loss of one train (or one flow path) of preferred AFW cooling to each SG such that redundancy is lost. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the SAFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

D.1

With all AFW trains to one or both SGs inoperable, action must be taken to restore at least one train or TDAFW flow path to each affected SG to OPERABLE status within 4 hours.

The combination of failures which require entry into this Condition all result in the loss of preferred AFW cooling to at least one SG. If a SGTR were to occur in this condition, preferred AFW is potentially unavailable to the unaffected SG. If AFW is unavailable to both SGs, the accident analyses for small break LOCAs and loss of MFW would not be met.

The two MDAFW trains of the preferred AFW System are normally used for decay heat removal during low power operations since air operated bypass control valves are installed in each train to better control SG level (see Figure B 3.7.5-1). Since a feedwater transient is more likely during reduced power conditions, 4 hours is provided to restore at least one train of additional preferred AFW before requiring a controlled cooldown. This will also provide time to find a condensate source other than the SW System for the SAFW System if all three AFW trains are inoperable. The 4 hour Completion Time is reasonable, based on redundant capabilities afforded by the SAFW System, time needed for repairs, and the low probability of a DBA occurring during this time period.

E.1

With one SAFW train inoperable, action must be taken to restore OPERABLE status within 14 days. This Condition includes the inoperability of one of the two SAFW cross-tie valves which requires declaring the associated SAFW train inoperable (e.g., failure of 9703B would result in declaring SAFW train D inoperable). The 14 day Completion Time is reasonable, based on redundant capabilities afforded by the AFW System, time needed for repairs, and the low probability of a HELB or other event which would require the use of the SAFW System during this time period.

F.1

With both SAFW trains inoperable, action must be taken to restore at least one SAFW train to OPERABLE status within 7 days. This Condition includes the inoperability of the SAFW cross-tie. The 7 day Completion Time is reasonable, based on redundant capabilities afforded by the AFW System, time needed for repairs, and the low probability of a HELB or other event which would require the use of the SAFW System during this time period.

G.1 and G.2

When Required Action A.1, B.1, C.1, D.1, E.1, or F.1 cannot be completed within the required Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant condition from full power conditions in an orderly manner and without challenging plant systems.

H.1

If all three preferred AFW trains and both SAFW trains are inoperable the plant is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety related equipment. In such a condition, the plant should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one MDAFW, TDAFW, or SAFW train to OPERABLE status. For the purposes of this Required Action, only one TDAFW train flow path and the pump must be restored to exit this Condition.

Required Action H.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one MDAFW, TDAFW, or SAFW train is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the plant into a less safe condition.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW and SAFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification, through a system walkdown, that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

Periodically comparing the reference differential pressure and flow of each AFW pump in accordance with the inservice testing requirements of ASME, Section XI (Ref. 4) detects trends that might be indicative of an incipient failure. The Frequency of this surveillance is specified in the Inservice Testing Program, which encompasses Section XI of the ASME code. Section XI of the ASME code provides the activities and Frequencies necessary to satisfy this requirement.

This SR is modified by a Note indicating that the SR is only required to be met prior to entering MODE 1 for the TDAFW pump since suitable test conditions have not been established. This deferral is required because there is insufficient steam pressure to perform the test.

SR 3.7.5.3

Periodically comparing the reference differential pressure and flow of each SAFW pump in accordance with the inservice testing requirements of ASME, Section XI (Ref. 4) detects trends that might be indicative of an incipient failure. Because it is undesirable to introduce SW into the SGs while they are operating, this testing is performed using the test condensate tank. The Frequency of this surveillance is specified in the Inservice Testing Program, which encompasses Section XI of the ASME code. Section XI of the ASME code provides the activities and Frequencies necessary to satisfy this requirement.

SR 3.7.5.4

This SR verifies that each AFW and SAFW motor operated suction valve from the SW System (4013, 4027, 4028, 9629A, and 9629B), each AFW and SAFW discharge motor operated valve (4007, 4008, 9704A, 9704B, and 9746), and each SAFW cross-tie motor operated valve (9703A and 9703B) can be operated when required. The Frequency of this Surveillance is specified in the Inservice Test Program and is consistent with ASME Code, Section XI (Ref. 4).

SR 3.7.5.5

This SR verifies that AFW can be delivered to the appropriate SG in the event of any accident or transient that generates an actuation signal, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

SR 3.7.5.6

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an actuation signal by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal. The 24 month Frequency is based on the potential need to perform this Surveillance under the conditions that apply during a plant outage.

This SR is modified by a Note indicating that the SR is only required to be met prior to entering MODE 1 for the TDAFW pump since suitable test conditions may have not been established. This deferral is required because there is insufficient steam pressure to perform the test.

SR 3.7.5.7

This SR verifies that the SAFW System can be actuated and controlled from the control room. The SAFW System is assumed to be manually initiated within 10 minutes in the event that the preferred AFW System is inoperable. The Frequency of 24 months 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 at power.

REFERENCES

1. UFSAR, Section 10.5.
 2. UFSAR Chapter 15.
 3. American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
 4. ASME, Boiler and Pressure Vessel Code, Section XI.
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Figure B 3.7.5-1
Preferred AFW System

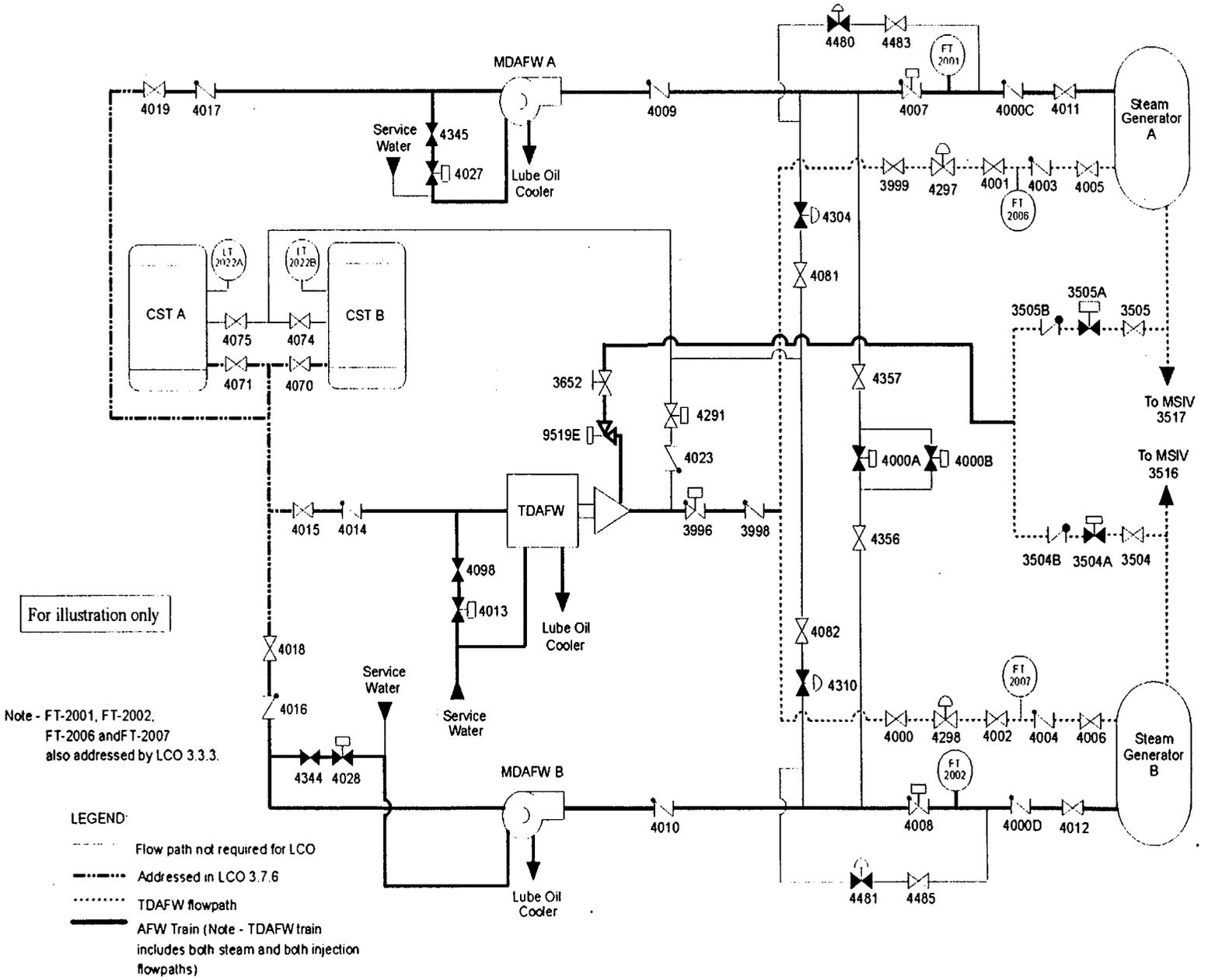
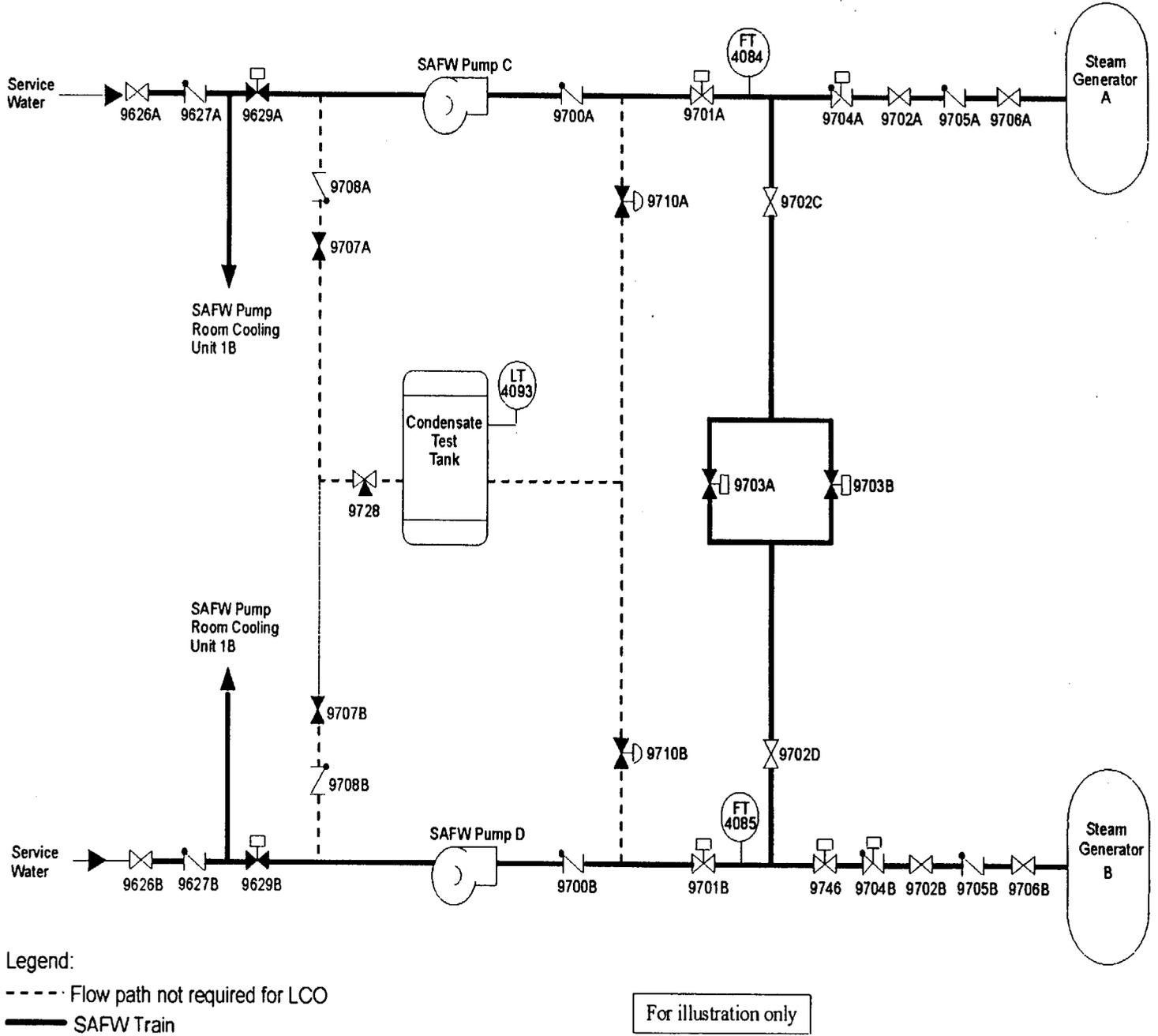


Figure B 3.7.5-2
Standby AFW System



B 3.7 PLANT SYSTEMS

B 3.7.6 Condensate Storage Tanks (CSTs)

BASES

BACKGROUND The CSTs provide a source of water to the steam generators (SGs) for removing decay and sensible heat from the Reactor Coolant System (RCS). The CST provides a passive flow of water, by gravity, to the preferred Auxiliary Feedwater (AFW) System (LCO 3.7.5) (see Figure B 3.7.5-1). The resulting steam produced in the SGs is released to the atmosphere by the main steam safety valves or the atmospheric relief valves.

When the main steam isolation valves are open, the preferred means of heat removal from the RCS is to discharge steam to the condenser by the nonsafety grade path of the steam dump valves. The condensed steam is then returned to the SGs by the main feedwater system. This has the advantage of conserving condensate while minimizing releases to the environment.

There are two 30,000 gallon CSTs located in the non-seismic Service Building (Ref. 1). The CSTs are not considered safety related components since the tanks are not protected against earthquakes or other natural phenomena, including missiles. The safety related source of condensate for the AFW and Standby AFW Systems is the Service Water (SW) System (LCO 3.7.8). The CSTs are connected by a common header which leads to the suction of all three AFW pumps. A single level transmitter is provided for each CST (LT-2022A and LT-2022B). The CSTs can be refilled from the condenser hotwell or the all-volatile-treatment condensate storage tank.

APPLICABLE SAFETY ANALYSES

The CSTs provide cooling water to remove decay heat and to cooldown the plant following all events in the accident analysis (Ref. 2) which assumes that the preferred AFW System is available immediately following an accident. For any event in which AFW is not required for at least 10 minutes following the accident, the SW System provides the source of cooling water to remove decay heat.

The accident analyses have been verified to be within acceptance limits if AFW is not available until 10 minutes. However, the preferred AFW pumps receive various automatic actuation signals. Assuming that all three AFW pumps initiate at their maximum flowrate, the CSTs provide sufficient inventory for at least 20 minutes (at greater than required flowrates) before operator ACTION to refill the CSTs or transfer suction to the SW System is required.

A nonlimiting event considered in CST inventory determinations is a main feedwater line break inside containment. This break has the potential for dumping condensate until terminated by operator ACTION after 10 minutes since there is no automatic re-configuration of the AFW System. Following termination of the AFW flow to the affected SG by closing the AFW train discharge valves or stopping a pump, flow from the remaining AFW train or the SAFW System is directed to the intact SG for decay heat removal. This loss of condensate is partially compensated for by the retention of inventory in the intact SG.

For cooldowns following loss of all onsite and offsite AC electrical power, the CSTs contain sufficient inventory to provide a minimum of 2 hours of decay heat removal via the turbine-driven AFW pump >as required by NUREG-0737 (Ref. 4), item II.E.1.1. This beyond DBA requirement provides more limiting criteria for CST inventory.

The CSTs satisfy Criterion 3 of the NRC Policy Statement.

LCO

To satisfy accident analysis assumptions, the CST must contain sufficient inventory to support operation of the preferred AFW system for at least 10 minutes. After this time period, the accident analyses assume that AFW pump suction can be transferred to the safety related suction source (i.e., the SW System).

However, the required CST water volume is $\geq 22,500$ gallons, which is based on the need to provide at least 2 hours of decay heat removal via the turbine-driven AFW pump following loss of all AC electrical power (i.e., a beyond design basis event). The CSTs are considered OPERABLE when at least 22,500 gallons of water is available. The 22,500 gal minimum volume is met if one CST is ≥ 21.5 ft or if both CSTs are ≥ 12.5 ft. Since the CSTs are 30,000 gallon tanks, only one CST is required to meet the minimum required water volume for this LCO.

The OPERABILITY of the CSTs is determined by maintaining the tank level at or above the minimum required water volume.

APPLICABILITY

In MODES 1, 2, and 3, the CSTs are required to be OPERABLE to support the AFW System requirements.

In MODE 4, 5, or 6, the CST is not required because the AFW System is not required to be OPERABLE.

ACTIONS

A.1 and A.2

If the CST water volume is not within limits, the OPERABILITY of the backup supply should be verified by administrative means within 4 hours. OPERABILITY of the backup feedwater supply must include verification that the flow paths from the backup water supply to the preferred AFW pumps are OPERABLE and immediately available upon AFW initiation, and that the backup supply has the required volume of water available. Alternate sources of water include, but is not limited to, the SW System and the all-volatile-treatment condensate tank. In addition, the CSTs must be restored to OPERABLE status within 7 days, because the backup supply may be performing this function in addition to its normal functions. Continued verification of the backup supply is not required due to the large volume of water typically available from these alternate sources. The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period requiring the CSTs.

B.1 and B.2

If the backup supply cannot be verified or the CSTs cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.7.6.1

This SR verifies that the CSTs contain the required volume of cooling water. The 22,500 gal minimum volume is met if one CST is ≥ 21.5 ft or if both CSTs are ≥ 12.5 ft. The 12 hour Frequency is based on operating experience and the need for operator awareness of plant evolutions that may affect the CST inventory between checks. Also, the 12 hour Frequency is considered adequate in view of other indications in the control room, including alarms, to alert the operator to abnormal deviations in the CST level.

REFERENCES

1. UFSAR, Section 10.7.4.
 2. UFSAR, Chapter 15.
 3. American National Standard, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," N18.2-1973.
 4. NUREG-0737, "Clarification of TMI ACTION Plan Requirements," November 1980.
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B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, and a normal shutdown, the CCW System also provides this function for various safety related and nonsafety related components. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Service Water (SW) System, and thus to the environment. The safety related functions of the CCW system are covered by this LCO.

The CCW System consists of a single loop header supplied by two separate, 100% capacity, safety related pump and heat exchanger trains (Ref. 1) (see Figure B 3.7.7-1). Each CCW train consists of a manual suction and discharge valve, a pump, and a discharge check valve. The trains discharge to a common header which then supplies two heat exchangers, either of which can supply the safety related and non-safety related components cooled by CCW. The CCW loop header begins at the common piping at the discharge of the two parallel heat exchangers, and continues up to the first isolation valve for each component supplied by the CCW System. The CCW loop header then continues from the last isolation valve on the discharge of each supplied load to the common piping at the suction of the CCW pumps. Each pump is powered from a separate Class 1E electrical bus. An open surge tank in the system provides for thermal expansion and contraction of the CCW system and ensures that sufficient net positive suction head is available to the pumps. The CCW System is also provided with a radiation detector (R-17) to isolate the surge tank from the Auxiliary Building environment and to provide indication of a leak of radioactive water into the CCW System.

The CCW System is normally maintained below 100°F by the use of one pump train in conjunction with one heat exchanger. The standby CCW pump will automatically start if the system pressure falls to 50 psig.

The principal safety related function of the CCW System is the removal of decay heat from the reactor via the Residual Heat Removal (RHR) System. Since the removal of decay heat via the RHR System is only performed during the recirculation phase of an accident, the CCW pumps do not receive an automatic start signal. Following the generation of a safety injection signal, the normally operating CCW pump will remain in service unless an undervoltage signal is present on either Class 1E electrical Bus 14 or Bus 16 at which time the pump is stripped from its respective bus. A CCW pump can then be manually placed into service prior to switching to recirculation operations which would not be required until a minimum of 22.4 minutes following an accident.

APPLICABLE
SAFETY
ANALYSES

The design basis of the CCW System is for one CCW train and one CCW heat exchanger to remove the loss of coolant accident (LOCA) heat load from the containment sump during the recirculation phase. The Emergency Core Cooling System (ECCS) and containment models for a LOCA each consider the minimum performance of the CCW System. The normal temperature of the CCW is $\leq 100^{\circ}\text{F}$, and, during LOCA conditions, a maximum temperature of 120°F is assumed. This prevents the CCW System from exceeding its design temperature limit of 200°F , and provides for a gradual reduction in the temperature of containment sump fluid as it is recirculated to the Reactor Coolant System (RCS) by the ECCS pumps. The CCW System is designed to perform its function with a single failure of any active component, assuming a coincident loss of offsite power.

The CCW trains, heat exchangers, and loop headers are manually placed into service prior to the recirculation phase of an accident (i.e., 22.4 minutes following a large break LOCA).

The CCW System can also function to cool the plant from RHR entry conditions ($T_{\text{avg}} < 350^{\circ}\text{F}$), to MODE 5 ($T_{\text{avg}} < 200^{\circ}\text{F}$), during normal cooldown operations. The time required to cool from 350°F to 200°F is a function of the number of CCW and RHR trains operating. Since CCW is comprised of a large loop header, a passive failure can be postulated during this cooldown period which results in draining the CCW System within a short period of time. The CCW System is also vulnerable to external events such as tornados. The plant has been evaluated for the loss of CCW under these conditions with the use of alternate cooling mechanisms (e.g., providing for natural circulation using the atmospheric relief valves and the Auxiliary Feedwater System) with acceptable results (Ref. 1). Leaks within the CCW System during post accident conditions can be mitigated by the available makeup water sources.

The CCW System satisfies Criterion 3 of the NRC Policy Statement.

LCO

In the event of a DBA, one CCW train, one heat exchanger, and the loop header is required to provide the minimum heat removal capability assumed in the safety analysis for the systems to which it supplies cooling water (see Figure B 3.7.7-1). To ensure this requirement is met, two trains of CCW, two heat exchangers, and the loop header must be OPERABLE. At least one CCW train will operate assuming the worst case single active failure occurs coincident with a loss of offsite power.

A CCW train is considered OPERABLE when the pump is OPERABLE and capable of providing cooling water to the loop header. The automatic start logic associated with low CCW system pressure is not required for this LCO. In addition, if a CCW pump fails an Inservice Testing Program surveillance (e.g., pump developed head) the pump is only declared inoperable when the flowrate to required components is below that required to provide the heat removal capability assumed in the accident analyses.

The CCW loop header is considered OPERABLE when the associated piping, valves, surge tank, and the instrumentation and controls required to provide cooling water to the following safety related components are available and capable of performing their safety related function:

- a. Two RHR heat exchangers;
- b. Two RHR pump mechanical seal coolers and bearing water jackets;
- c. Three safety injection pump mechanical seal coolers; and
- d. Two containment spray pump mechanical seal coolers.

The CCW loop header temperature must also be $\leq 120^{\circ}\text{F}$ prior to the CCW cooling water reaching the first isolation valve supplying these components.

The CCW loop header begins at the common piping at the discharge of the CCW heat exchangers and continues up to the first isolation valve for each of the above components. The CCW loop header then continues from the last isolation valve on the discharge of each of the above components to the common piping at the suction of the CCW pumps.

The portion of CCW piping, valves, instrumentation and controls between the isolation valves to components a through d above is addressed by the following LCOs:

- a. LCO 3.4.6, "RCS Loops - MODE 4,"
- b. LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled,"

- c. LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled,"
- d. LCO 3.5.2, "ECCS - MODES 1, 2, and 3,"
- e. LCO 3.5.3, "ECCS - MODE 4,"
- f. LCO 3.9.4, "RHR and Coolant Circulation - Water Level \geq 23 Ft,"
and
- g. LCO 3.9.5, "RHR and Coolant Circulation - Water Level $<$ 23 Ft."

The CCW piping inside containment for the reactor coolant pumps (RCPs) and the reactor support coolers also serves as a containment isolation boundary. This is addressed by LCO 3.6.3, "Containment Isolation Boundaries."

The CCW system radiation detector (R-17) is not required to be OPERABLE for this LCO since the CCW system outside containment is not required to be a closed system.

The isolation of CCW from other components or systems not required for safety may render those components or systems inoperable but does not affect the OPERABILITY of the CCW System.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CCW System is a normally operating system, which must be capable to perform its post accident safety functions. The failure to perform this safety function could result in the loss of reactor core cooling and containment integrity during the recirculation phase following a LOCA.

In MODE 5 or 6, the OPERABILITY requirements of the CCW System are determined by LCO 3.4.7, LCO 3.4.8, LCO 3.9.4, and LCO 3.9.5.

ACTIONS

A.1

If one CCW train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE CCW train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE CCW train could result in loss of CCW function. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this period.

B.1

If one CCW heat exchanger is inoperable, action must be taken to restore OPERABLE status within 31 days. In this Condition, the remaining OPERABLE heat exchanger is adequate to perform the heat removal function. However, the overall reliability is reduced because a passive failure in the OPERABLE CCW heat exchanger could result in a loss of CCW function. The 31 day Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a passive failure of the remaining heat exchanger.

C.1 and C.2

If the CCW train or CCW heat exchanger cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1, D.2, and D.3

With both CCW trains, both CCW heat exchangers, or the loop header inoperable, action must be immediately initiated to restore OPERABLE status to one CCW train, one CCW heat exchanger, and the loop header. In this Condition, there is no OPERABLE CCW System available to provide necessary cooling water which is a loss of a safety function. Also, the plant must be placed in a MODE in which the consequences of a loss of CCW coincident with an accident are reduced. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The plant is not required to exit the Applicability for this LCO (i.e., enter MODE 5) until at least one CCW train, one CCW heat exchanger, and the loop header is restored to OPERABLE status to support RHR operation.

Required Actions D.1, D.2, and D.3 are modified by a Note indicating that all required MODE changes or power reductions required by other LCOs are suspended until one CCW train, one CCW heat exchanger, and the loop header are restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the plant into a less safe condition.

SURVEILLANCE
REQUIREMENTS

SR 3.7.7.1

Verifying the correct alignment for manual and power operated valves in the CCW flow path provides assurance that the proper flow paths exist for CCW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification, through a system walkdown, that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

This SR is modified by a Note indicating that the isolation of the CCW flow to individual components may render those components inoperable but does not affect the OPERABILITY of the CCW loop header.

SR 3.7.7.2

This SR verifies that the two motor operated isolation valves to the RHR heat exchangers (738A and 738B) can be operated when required since the valves are normally maintained closed. The Frequency of this Surveillance is specified in the Inservice Test Program and is consistent with ASME Code, Section XI (Ref. 2).

REFERENCES

1. UFSAR, Section 9.2.2.
 2. ASME, Boiler and Pressure Vessel Code, Section XI.
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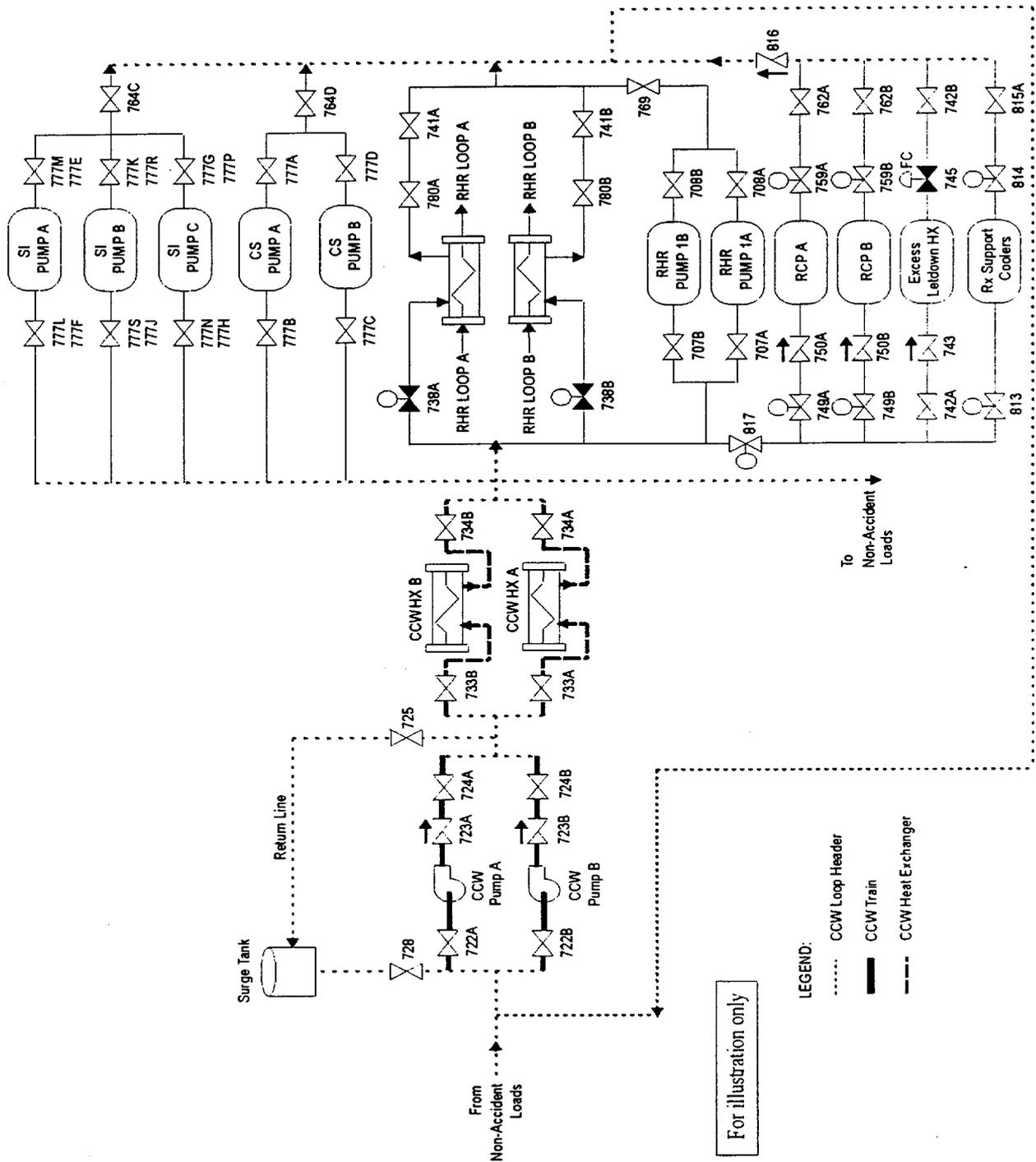


Figure B 3.7.7-1
CCW System

B 3.7 PLANT SYSTEMS

B 3.7.8 Service Water (SW) System

BASES

BACKGROUND

The SW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, and a normal shutdown, the SW system also provides this function for various safety related and nonsafety related components. The safety related functions of the SW System are covered by this LCO.

The SW System consists of a single loop header supplied by two separate, 100% capacity, safety related pump trains (Ref. 1) (see Figure B 3.7.8-1). The physical design of the SW System is such that one 100% capacity pump from each class 1E electrical bus (Buses 17 and 18) is arranged on a common piping header which then supplies the SW loop header. For the purposes of this LCO, a SW train is based on electrical source only.

Each train is powered from a separate Class 1E electrical bus and consists of two 100% capacity pumps and associated discharge check valves and manual isolation valves. The SW loop header begins from the discharge of the trains and supplies the safety related and nonsafety related components cooled by SW. The pumps in the system are normally manually aligned. One pump in each train is selected to automatically start after diesel generator supply breaker closure on its respective bus when a safety injection signal is absent. Upon receipt of a safety injection signal, one SW pump on each train will automatically start in a predetermined sequence.

The SW loop header supplies the cooling water to all safety related and nonsafety related components. The nonsafety related and long-term safety functions (e.g., component cooling water heat exchangers) can be isolated from the loop header through use of redundant motor operated isolation valves. These valves automatically close on a coincident safety injection signal and undervoltage signal on Buses 14 and 16.

The suction source for the SW System is the screenhouse which is a seismic structure located on Lake Ontario. The discharge from the SW System supplied loads returns back to Lake Ontario. The principal safety related functions of the SW system is the removal of decay heat from the reactor via the Component Cooling Water (CCW) System, provide cooling water to the diesel generators (DGs) and containment recirculation fan coolers (CRFCs) and to provide a safety related source of water to the Auxiliary Feedwater (AFW) System.

APPLICABLE
SAFETY
ANALYSES

The design basis of the SW System is for one SW train in conjunction with a 100% capacity containment cooling system (i.e., CRFC) to provide for heat removal following a steam line break (SLB) inside containment to ensure containment integrity. The SW System is also designed, in conjunction with the CCW System and a 100% capacity Emergency Core Cooling System and containment cooling system, to remove the loss of coolant accident (LOCA) heat load from the containment sump during the recirculation phase (Ref. 2). This prevents the containment sump fluid from increasing in temperature during the recirculation phase following a LOCA and provides for a gradual reduction in the temperature of this fluid as it is recirculated to the Reactor Coolant System by the ECCS pumps. The SW System is designed to perform its function with a single failure of any active component, assuming a coincident loss of offsite power.

Upon an undervoltage signal, all running SW pumps are stripped from their respective safeguards bus. The selected SW pumps are sequenced to start after a 40 second time delay following energization of the electrical bus supplying the selected pump (i.e., Bus 17 or Bus 18) if no safety injection signal is present.

Following the receipt of a safety injection signal, one preselected SW pump is designed to start on each train within 17 seconds (if not already running) to supply the system loads. The two non-selected SW pumps will continue to operate if they were already running. If a coincident safety injection and undervoltage signal occurs, then each nonessential load within the SW System is isolated by redundant motor operated valves that are powered by separate Class 1E electrical trains.

The SW trains and loop header are assumed to supply to following components following an accident:

- a. The CRFCs, DGs and safety injection pump bearing housing coolers immediately following a safety injection signal (i.e., after the loop header becomes refilled);
- b. The preferred AFW and SAFW pumps within 10 minutes following receipt of a low SG level signal; and
- c. The CCW heat exchangers within 22.4 minutes following a safety injection signal.

The SW system, in conjunction with the CCW System, can also cool the plant from residual heat removal (RHR) entry conditions ($T_{avg} < 350^{\circ}\text{F}$) to MODE 5 ($T_{avg} < 200^{\circ}\text{F}$) during normal operations. The time required to cool from 350°F to 200°F is a function of the number of CCW and RHR System trains that are operating. Since SW is comprised of a large loop header, a passive failure can be postulated during this cooldown period

which results in failing the SW System to potentially multiple safety related functions. The SW system has been evaluated to demonstrate the capability to meet cooling needs with an assumed 500 gal leak. The SW System is also vulnerable to external events such as tornados. The plant has been evaluated for the loss of SW under these conditions with the use of alternate cooling mechanisms (e.g., providing for natural circulation using the atmospheric relief valves and the AFW Systems) with acceptable results (Ref. 1).

The temperature of the fluid supplied by the SW System is also a consideration in the accident analyses. If the cooling water supply to the containment recirculation fan coolers and CCW heat exchangers is too warm, the accident analyses with respect to containment pressure response following a SLB and the containment sump fluid temperature following a LOCA may no longer be bounding. As the cooling water supply temperature is lowered, the containment heat removal systems become more efficient which causes the backpressure in containment to be reduced, resulting in increased peak clad temperatures. The bounding minimum cooling water temperature assumed in the accident analysis is 30°F, which is lower than the freezing point of the cooling water supply.

The SW system satisfies Criterion 3 of the NRC Policy Statement.

LCO

In the event of a DBA, one SW train and the loop header is required to be OPERABLE to provide the minimum heat removal capability to ensure that the system functions to remove post accident heat loads as assumed in the safety analyses. To ensure this requirement is met, two trains of SW and the loop header must be OPERABLE (see Figure B 3.7.8-1). At least one SW train will operate assuming that the worst case single active failure occurs coincident with the loss of offsite power.

A SW train is defined based on electrical power source such that SW Pumps A and C form one train and SW Pumps B and D form the second train. A SW train is considered OPERABLE when one pump in the train is OPERABLE and capable of taking suction from the screenhouse and providing cooling water to the loop header as assumed in the accident analyses. This includes consideration of available net positive suction head (NPSH) to the SW pumps and the temperature of the suction source. The following are the minimum requirements of the screenhouse bay with respect to OPERABILITY of the SW system:

- a. Level \geq 14 feet; and
- b. Temperature \leq 85°F.

The screenhouse bay level verification should normally be performed using LI-3006. Monitoring screenhouse bay temperature (normally performed by using T3001 or T3002) is an acceptable means of ensuring inlet temperature to safety related loads are within limits. In addition, if a SW pump fails on Inservice Testing Program surveillance (e.g., pump developed head), the pump is only declared inoperable when the flowrate to required components is below that required to provide the heat removal capability assumed in the accident analyses (Ref. 1).

An OPERABLE SW train also requires that all nonessential and nonsafety related loads can be isolated by the six motor operated isolation valves which are powered from the same Class 1E electrical train as the pumps. Therefore, motor operated valves 4609, 4614, 4615, 4616, 4663, and 4670 must be OPERABLE and capable of closing for SW Pumps A and C while valves 4613, 4664, 4733, 4734, 4735, and 4780 must be OPERABLE and capable of closing for SW Pumps B and D.

The SW loop header is considered OPERABLE when the associated piping, valves, and the instrumentation and controls required to provide cooling water from each OPERABLE SW train to the following safety related components are available and capable of performing their safety related function:

- a. Four CRFCs;
- b. Two CCW heat exchangers;
- c. Two DGs;
- d. Three preferred AFW pumps;
- e. Two standby AFW pumps; and
- f. Three safety injection pump bearing housing coolers.

An OPERABLE SW loop header also requires a flow path through the diesel generator (4665, 4760, 4669, and 4668B) and CRFC (4623, 4640, 4756 and 4639) cross-ties.

The SW loop header begins at the common piping at the discharge of both SW pump trains and ends at the first isolation valve for each of the above components. Since the SW System discharges back to Lake Ontario, the cooling water flow path through the above components and subsequent discharge is addressed under their respective LCO. This includes:

- a. LCO 3.5.2, "ECCS - MODES 1, 2, and 3;"
- b. LCO 3.5.3, "ECCS - MODE 4;"

- c. LCO 3.6.6, "CS, CRFC, and Containment Post-Accident Charcoal Systems;"
- d. LCO 3.7.5, "AFW Systems;"
- e. LCO 3.7.7, "CCW System;"
- f. LCO 3.8.1, "AC Sources - MODES 1, 2, 3, and 4;" and
- g. LCO 3.8.2, "AC Sources - MODES 5 and 6."

The SW piping inside containment for the CRFCs and the reactor compartment coolers also serves as a containment isolation boundary. This is addressed under LCO 3.6.3, "Containment Isolation Boundaries."

APPLICABILITY

In MODES 1, 2, 3, and 4, the SW System is a normally operating system which must be capable of performing its post accident safety functions. The failure to perform this safety function could result in the loss of reactor core cooling during the recirculation phase following a LOCA or loss of containment integrity following a SLB.

In MODES 5 and 6, the OPERABILITY requirements of the SW system are determined by LCO 3.7.7 and LCO 3.8.2.

ACTIONS

A.1

If one SW train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE SW train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE SW train could result in loss of SW System function. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this time period.

B.1 and B.2

If the SW train cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1

With both SW trains or the loop header inoperable, the plant is in a condition outside of the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

Required Action C.1 is modified by a Note requiring that the applicable Conditions and Required Actions of LCO 3.7.7, "CCW System," be entered for the component cooling water heat exchanger(s) made inoperable by SW. This note is provided since the inoperable SW system may prevent the plant from reaching MODE 5 as required by LCO 3.0.3 if both CCW heat exchangers are rendered inoperable.

SURVEILLANCE
REQUIREMENTS

SR 3.7.8.1

This SR verifies that adequate NPSH is available to operate the SW pumps and that the SW suction source temperature is within the limits assumed by the accident analyses and the system design. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

SR 3.7.8.2

Verifying the correct alignment for manual, power operated, and automatic valves in the SW flow path provides assurance that the proper flow paths exist for SW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation; rather, it involves verification, through a system walkdown, that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

This SR is modified by a Note indicating that the isolation of the SW flow to individual components or systems may render those components inoperable, but does not affect the OPERABILITY of the SW System.

SR 3.7.8.3

This SR verifies that all SW loop header cross-tie valves are locked in the correct position. This includes verification that manual valves 4623, 4639, 4640, 4665, 4668B, 4669, 4756, and 4760 are locked open and that manual valves 4610, 4611, 4612, and 4779 are locked closed. The 31 day Frequency is based on engineering judgement, is consistent with the procedural controls governing locked valves, and ensures correct valve positions.

SR 3.7.8.4

This SR verifies proper automatic operation of the SW motor operated isolation valves on an actual or simulated actuation signal (i.e., coincident safety injection and undervoltage signal). SW is a normally operating system that cannot be fully actuated as part of normal testing. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

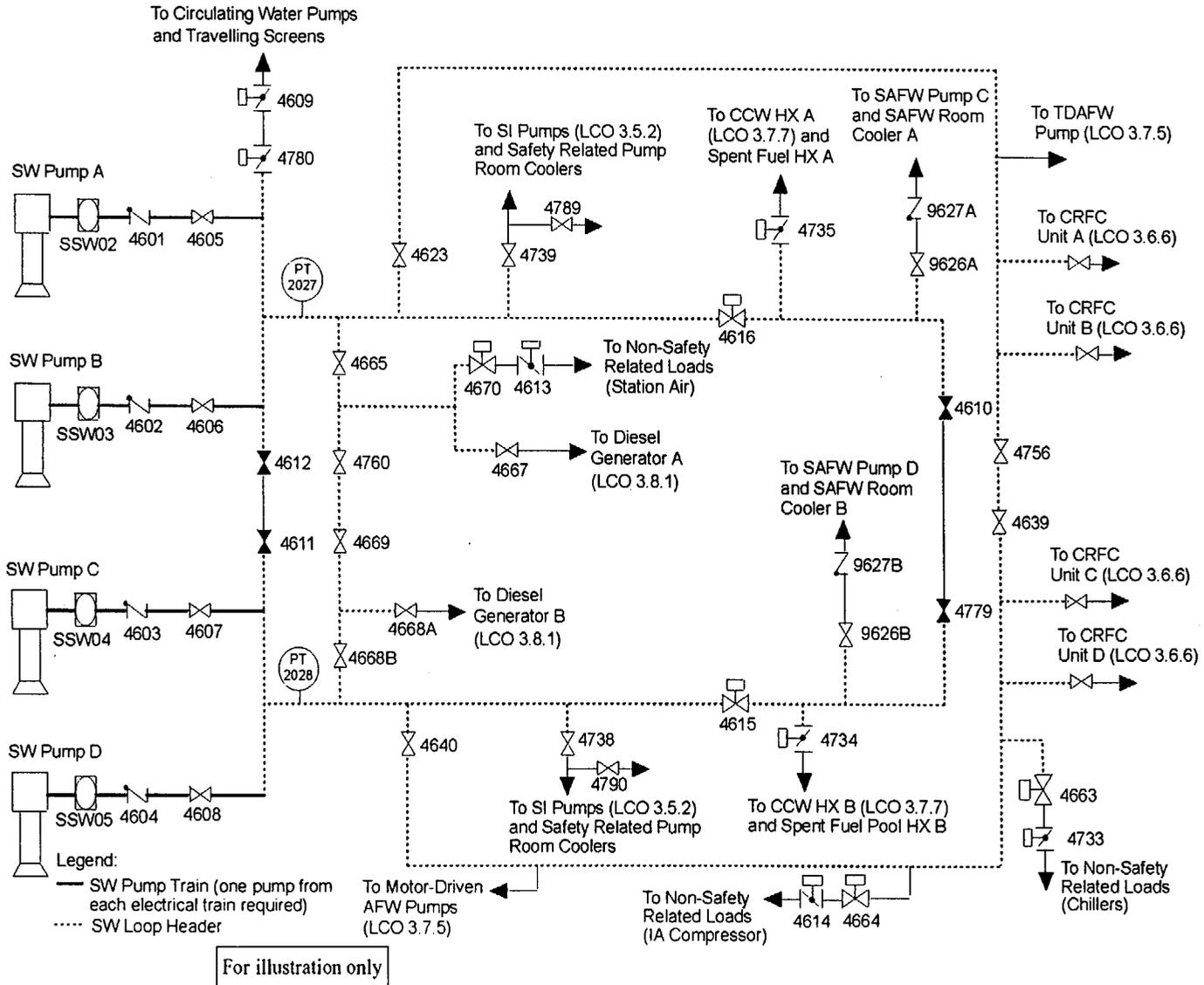
SR 3.7.8.5

This SR verifies proper automatic operation of the SW pumps on an actual or simulated actuation signal. This includes the actuation of the SW pumps following an undervoltage signal and following a coincident safety injection and undervoltage signal. SW is a normally operating system that cannot be fully actuated as part of normal testing during normal operation. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 9.2.1.
 2. UFSAR, Section 6.2.
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Figure B 3.7.8-1
SW System



B 3.7 PLANT SYSTEMS

B 3.7.9 Control Room Emergency Air Treatment System (CREATS)

BASES

BACKGROUND

According to Atomic Industry Forum (AIF) GDC 11 (Ref. 1), a control room shall be provided which permits continuous occupancy under any credible postaccident condition without excessive radiation exposures of personnel. Exposure limits are provided in GDC 19 of 10 CFR 50, Appendix A (Ref. 2) which requires that control room personnel be restricted to 5 rem whole body, or its equivalency, for the duration of the accident. The CREATS provides a protected environment from which operators can control the plant following an uncontrolled release of radioactivity for 30 days without exceeding this 5 rem whole body limit. The CREATS is part of the Control Building ventilation system.

The CREATS consists of a high efficiency particulate air (HEPA) filter, activated charcoal adsorbers for removal of gaseous activity (principally iodines), and two fans (control room return air fan and emergency return air fan) (see Figure B 3.7.9-1). Ductwork, dampers, and instrumentation also form part of the system (Ref. 3).

The CREATS is an emergency system, parts of which may operate during normal plant operations. Actuation of the CREATS places the system in one of five separate states of the emergency mode of operation, depending on the initiation signal. The following are the normal and emergency modes of operation for the CREATS:

CREATS Mode A

The CREATS is in the standby mode with the exception that the control room return air fan is in operation.

CREATS Mode B

This is the CREATS configuration following an accident with a radiation release as detected by radiation monitor R-1. Upon receipt of an actuation signal, the control room emergency return air fan will actuate and system dampers align to recirculate a maximum of 2000 cfm (approximately one fourth of the Control Building Ventilation System design) through the CREATS charcoal and HEPA filters. All outside air that enters the CREATS, as controlled by an air adjust switch (S-81), is also circulated through the CREATS charcoal and HEPA filters.

CREATS Mode C

This is the same CREATS configuration as Mode B with the exception that all outside air is isolated to the control room by one damper in each air supply flow path.

CREATS Mode D

This is the CREATS configuration following the detection of smoke within the Control Building. Upon receipt of an actuation signal, the system continues to draw outside air. However, the control room emergency return air fan will actuate and system dampers align to recirculate a maximum of 2000 cfm through the CREATS and HEPA filters. This effectively purges the control room air environment.

CREATS Mode E

This is the same CREATS configuration as Mode D with exception that all outside air is isolated to the control room by one damper in each air supply flow path.

CREATS Mode F

This is the CREATS configuration following the detection of a toxic gas as indicated by the chlorine or ammonia detectors, or high radiation as detected by R-36 (gas), R-37 (particulate), or R-38 (iodine). Upon receipt of an actuation signal, the system aligns itself consistent with Mode C except that two dampers in each air supply path are isolated.

Normally open air supply isolation dampers are arranged in series so that the failure of one damper to close will not result in a breach of isolation.

The air entering the control room is continuously monitored by radiation and toxic gas detectors. One detector output above the setpoint will cause actuation of the emergency radiation state or toxic gas isolation state, as required. The actions of the toxic gas and high radiation state (Mode F) are more restrictive, and will override the actions of the emergency radiation state (Mode B or C). Only the high radiation state CREATS Mode F is addressed by this LCO.

APPLICABLE
SAFETY
ANALYSES

The location of components and CREATS related ducting within the control room emergency zone envelope ensures an adequate supply of filtered air to all areas requiring access. The CREATS provides airborne radiological protection for the control room operators in MODES 1, 2, 3, and 4, as demonstrated by the control room accident dose analyses for the most limiting design basis loss of coolant accident and steam generator tube rupture (Ref. 3). This analysis shows that with credit for the CREATS, or with credit for instantaneous isolation of the control room coincident with the accident initiator and no CREATS filtration train available, the dose rates to control room personnel remain within GDC 19 limits.

During movement of irradiated fuel assemblies or during CORE ALTERATIONS, the CREATS ensures control room habitability in the event of a fuel handling accident. It has been demonstrated that the CREATS is not required in the event of a waste gas decay tank rupture (Ref. 5).

The CREATS satisfies Criterion 3 of the NRC Policy Statement.

LCO

The CREATS is comprised of a filtration train and two independent and redundant isolation damper trains all of which are required to be OPERABLE. Total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a large radioactive release.

The CREATS is considered OPERABLE when the individual components necessary to permit CREATS Mode F operation are OPERABLE (see Figure B 3.7.9-1). The CREATS filtration train is OPERABLE when the associated:

- a. Control room return air and emergency return air fans are OPERABLE and capable of providing forced flow;
- b. HEPA filters and charcoal adsorbers for the emergency return air fan are not excessively restricting flow, and are capable of performing their filtration functions; and
- c. Ductwork, valves, and dampers (including AKD06 and AKD09) are OPERABLE, and air circulation can be maintained.

The CREATS isolation dampers are considered OPERABLE when the damper (AKD01, AKD04, AKD05, AKD08, and AKD10) can close on an actuation signal to isolate outside air or is closed with motive force removed. Two dampers are provided for each outside air path.

In addition, the control room emergency zone boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and personnel access doors. The control room emergency zone boundary also includes the control room lavatory exhaust damper (AKD02), which closes on an actuation signal, and associated ductwork. Opening of the personnel access doors for entry and exit does not violate the control room emergency zone boundary. A personnel access door or a ventilation system ductwork access door may be opened for extended periods provided a dedicated individual is stationed at the access door to ensure closure, if required (i.e., the individual performs the isolation function), the door is able to be closed within 30 seconds upon indication of the need to close the door, and the CREATS filtration train is OPERABLE. The ventilation ductwork may also be opened for extended periods provided that the CREATS filtration train is declared inoperable, the fans are off, and the portion of ductwork that is open is isolated from the control room by a damper that is closed with motive force removed.

APPLICABILITY

In MODES 1, 2, 3, and 4, the CREATS must be OPERABLE to control operator exposure during and following a DBA.

During movement of irradiated fuel assemblies or during CORE ALTERATIONS, the CREATS must be OPERABLE to cope with the release from a fuel handling accident.

ACTIONS

A.1 and A.2

With the CREATS filtration train inoperable, action must be taken to restore OPERABLE status within 48 hours or isolate the control room from outside air. In this Condition, the isolation dampers are adequate to perform the control room protection function but no means exist to filter the release of radioactive gas within the control room. The 48 hour Completion Time is based on the low probability of a DBA occurring during this time frame, and the ability of the CREATS dampers to isolate the control room.

Required Action A.2 is modified by a Note which allows the control room to be unisolated for ≤ 1 hour every 24 hours. This allows fresh air makeup to improve the working environment within the control room and is acceptable based on the low probability of a DBA occurring during this makeup period.

B.1

With one CREATS isolation damper inoperable for one or more outside air flow paths, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREATS isolation damper is adequate to perform the control room protection function. However, the overall reliability is reduced because a single failure in the OPERABLE CREATS isolation damper could result in loss of CREATS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining isolation damper to provide the required isolation capability.

C.1 and C.2

In MODE 1, 2, 3, or 4, if the Required Actions of Conditions A or B cannot be completed within the required Completion Time, the plant must be placed in a MODE that minimizes accident risk. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1, D.2.1, and D.2.2

During movement of irradiated fuel assemblies or during CORE ALTERATIONS, if the Required Actions of Conditions A or B cannot be completed within the required Completion Time, action must be taken to immediately place the OPERABLE isolation damper(s) in CREATS Mode F. This action ensures that the remaining damper(s) are OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

An alternative to Required Action D.1 is immediately suspend activities that could result in a release of radioactivity that might enter the control room. This requires the suspension of CORE ALTERATIONS and the suspension of movement of irradiated fuel assemblies. This places the plant in a condition that minimizes risk. This does not preclude the movement of fuel or other components to a safe position.

E.1

In MODE 1, 2, 3, or 4, if both CREATS isolation dampers for one or more outside air flow paths are inoperable, the CREATS may not be capable of performing the intended function and the plant is in a condition outside the accident analyses. Failure of the integrity of the control room emergency zone boundary (i.e., walls, floors, ceilings, ductwork, personnel access doors, or control room lavatory exhaust damper AKD02) also results in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

F.1 and F.2

During movement of irradiated fuel assemblies or during CORE ALTERATIONS with two CREATS isolation dampers for one or more outside air flow paths inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might enter the control room. This requires the suspension of CORE ALTERATIONS and the suspension of movement of irradiated fuel assemblies. This places the plant in a condition that minimizes accident risk. This does not preclude the movement of fuel or other components to a safe position.

SURVEILLANCE
REQUIREMENTSSR 3.7.9.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each CREATS filtration train once every 31 days for ≥ 15 minutes provides an adequate check of this system. The 31 day Frequency is based on the reliability of the equipment.

SR 3.7.9.2

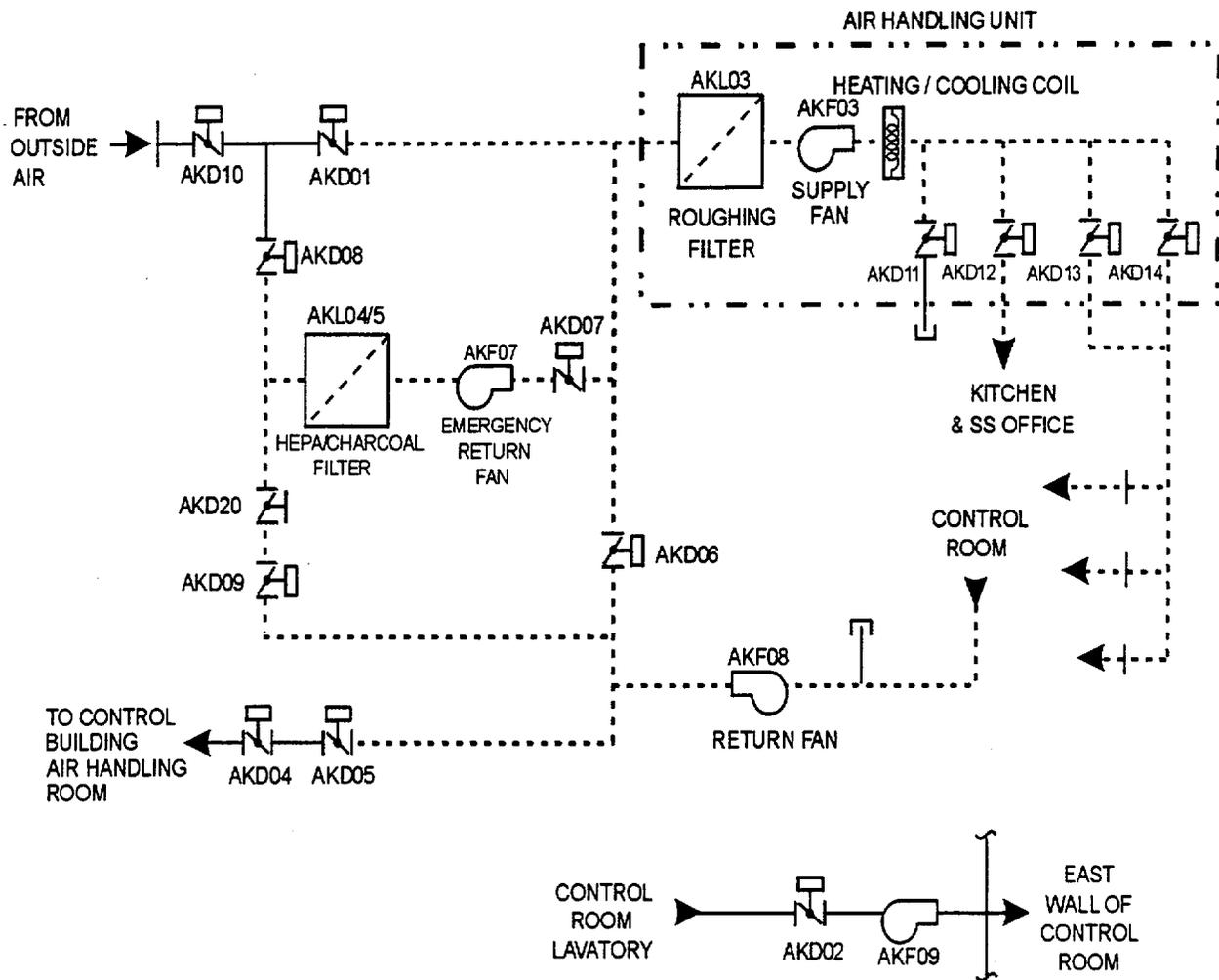
This SR verifies that the required CREATS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The CREATS filter tests are in general accordance with Regulatory Guide 1.52 (Ref. 4). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. The minimum required flowrate through the CREATS filtration train is 2000 cubic feet per minute ($\pm 10\%$). Specific test Frequencies and additional information are discussed in detail in the VFTP. However, the maximum surveillance interval for refueling outage tests is based on 24 month refueling cycles and not 18 month cycles as defined by Regulatory Guide 1.52 (Ref. 4).

SR 3.7.9.3

This SR verifies that the CREATS filtration train starts and operates and each CREATS isolation damper actuates on an actual or simulated actuation signal. The Frequency of 24 months is based on Regulatory Guide 1.52 (Ref. 4).

REFERENCES

1. Atomic Industry Forum (AIF) GDC 11, Issued for comment July 10, 1967.
 2. 10 CFR 50, Appendix A, GDC 19.
 3. UFSAR, Section 6.4.
 4. Regulatory Guide 1.52, Revision 2.
 5. Letter from Robert C. Mecredy, RG&E, to Guy S. Vissing, NRC, Subject: Application for Amendment to Facility Operating License Control Room Emergency Air Treatment System (CREATS) Applicability Change (LCO 3.3.6 and LCO 3.7.9), dated July 21, 2000.
-
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Legend:

----- CREATS Filtration Train

Notes:

1. Outside air flowpath isolation dampers include AKD01, AKD04, AKD05, AKD08, and AKD10.
2. The CREATS filtration train does not include the Air Handling Unit.

For illustration only

Figure B 3.7.9-1
CREATS

B 3.7 PLANT SYSTEMS

B 3.7.10 Auxiliary Building Ventilation System (ABVS)

BASES

BACKGROUND

The ABVS filters airborne radioactive particulates from the area of the spent fuel pool (SFP) following a fuel handling accident. The ABVS, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the Auxiliary Building including the SFP area.

The ABVS consists of an air handling unit, a series of exhaust fans, charcoal filters, ductwork, and dampers (Ref. 1). The exhaust fans include the following fans which all discharge into a common ductwork that supplies the Auxiliary Building main exhaust fans A and B (see Figure B 3.7.10-1):

- a. Intermediate Building exhaust fans A and B;
- b. Auxiliary Building exhaust fan C;
- c. Auxiliary Building charcoal filter fans A and B;
- d. Auxiliary Building exhaust fan G; and
- e. Control access exhaust fans A and B.

The only components which filter the environment associated with the SFP are the Auxiliary Building main exhaust fans and Auxiliary Building exhaust fan C. Therefore, these are the only fans considered with respect to the ABVS in this LCO.

Auxiliary Building exhaust fan C takes suction from the SFP and decontamination pit areas on the operating level of the Auxiliary Building. The air is first drawn through the SFP Charcoal Adsorber System which consists of roughing filters and charcoal adsorbers. The roughing filters protect the charcoal adsorbers from being fouled with dirt particles while the charcoal adsorbers remove the radioactive iodines from the atmosphere. Auxiliary Building exhaust fan C then discharges into the common ductwork that supplies the Auxiliary Building main exhaust fans. This common ductwork contains a high efficiency particulate air (HEPA) filter which is not credited in the dose analyses.

The Auxiliary Building main exhaust fans are each 100% capacity fans which can maintain a negative pressure on the operating floor of the Auxiliary Building through orientation of the system dampers. This negative pressure causes air flow on the operating floor to be toward the SFP which ensures that air in the vicinity of the SFP is first filtered through the SFP Charcoal Adsorber System. The Auxiliary Building main exhaust fans and exhaust fan C are powered from non-Engineered Safeguards Features buses.

The Auxiliary Building main exhaust fans discharge to the plant vent stack. The plant vent stack is continuously monitored for noble gases (R-14), particulates (R-13) and iodine (R-10B). During normal power operation, the ABVS is placed in the "out" mode by the interlock mode switch where "out" defines the status of the SFP charcoal filters. This causes all exhaust fans without any HEPA or charcoal filters (excluding the Auxiliary Building Main exhaust fans) and Auxiliary Building exhaust fan C to trip upon a signal from R-10B, R-13 or R-14 to stop the release of any radioactive gases. During fuel movement within the Auxiliary Building, the interlock mode switch is placed in the "in" mode such that only exhaust fans without any HEPA or charcoal filters (excluding Auxiliary Building main exhaust fans) are tripped.

APPLICABLE
SAFETY
ANALYSES

The ABVS design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident. The analysis of the fuel handling accident, given in Reference 2, assumes that all fuel rods in an assembly are damaged. The DBA analysis of the fuel handling accident assumes that Auxiliary Building exhaust fan C, the SFP Charcoal Adsorber System, and one Auxiliary Building main exhaust fan are OPERABLE. The accident analysis accounts for the reduction in airborne radioactive material provided by the minimum filtration system components which result in offsite doses well within the limits of 10 CFR 100 (Ref. 3). The failure of any or all of these filtration system components results in doses which are slightly higher but still within 10 CFR 100 limits. The fuel handling accident assumptions and the analysis follow the guidance provided in Regulatory Guide 1.25 (Ref. 4).

The remainder of the ABVS described in the Background is not required for any DBA since it is non-safety related and supplied only from offsite power sources.

The ABVS satisfies Criterion 3 of the NRC Policy Statement.

LCO

The ABVS is required to be OPERABLE to ensure that offsite doses are well within the limits of 10 CFR 100 (Ref. 3) following a fuel handling accident in the Auxiliary Building. The failure of the ABVS coincident with a fuel handling accident results in doses which are slightly higher but still within 10 CFR 100 limits.

The ABVS is considered OPERABLE when the individual components necessary to control exposure in the Auxiliary Building following a fuel handling accident are OPERABLE and in operation (see Figure B 3.7.10-1). The ABVS is considered OPERABLE when its associated:

- a. Auxiliary Building exhaust fan C and either Auxiliary Building main exhaust fan A or B is OPERABLE and in operation;
- b. Auxiliary Building main exhaust fan HEPA filter and SFP charcoal adsorbers are not excessively restricting flow, and the SFP Charcoal Adsorber System is capable of performing its filtration function;
- c. Ductwork, valves, and dampers are OPERABLE, and air circulation and negative pressure can be maintained on the Auxiliary Building operating floor; and
- d. Interlock mode switch is placed in the "in" mode.

APPLICABILITY

During movement of irradiated fuel in the Auxiliary Building, the ABVS is required to be OPERABLE to alleviate the consequences of a fuel handling accident. The ABVS is only required when one or more fuel assemblies in the Auxiliary Building has decayed < 60 days since being irradiated. Any fuel handling accident which occurs after 60 days results in offsite doses which are well within 10 CFR 100 limits (Ref. 3) due to the decay rate of iodine.

Since a fuel handling accident can only occur as a result of fuel movement, the ABVS is not MODE dependant and only required when irradiated fuel is being moved.

ACTIONS

A.1

When the ABVS is inoperable, action must be taken to place the plant in a condition in which the LCO does not apply. Action must be taken immediately to suspend movement of irradiated fuel assemblies in the Auxiliary Building. This does not preclude the movement of fuel to a safe position.

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies in the Auxiliary Building which have decayed < 60 days since being irradiated, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE
REQUIREMENTS

SR 3.7.10.1

This SR verifies the OPERABILITY of the ABVS. During fuel movement operations, the ABVS is designed to maintain a slight negative pressure in the Auxiliary Building to prevent unfiltered LEAKAGE. This SR ensures that Auxiliary Building exhaust fan C, and either Auxiliary Building main exhaust fan A or B are in operation and that the ABVS interlock mode switch is in the correct position. The Frequency of 24 hours is based on engineering judgement and shown to be acceptable through operating experience.

SR 3.7.10.2

This SR verifies the integrity of the Auxiliary Building enclosure. The ability of the Auxiliary Building to maintain negative pressure with respect to the uncontaminated outside environment must be periodically verified to ensure proper functioning of the ABVS. During fuel movement operations, the ABVS is designed to maintain a slight negative pressure in the Auxiliary Building to prevent unfiltered leakage. This SR ensures that a negative pressure is being maintained in the Auxiliary Building. The Frequency of 24 hours is based on engineering judgement and shown to be acceptable through operating experience.

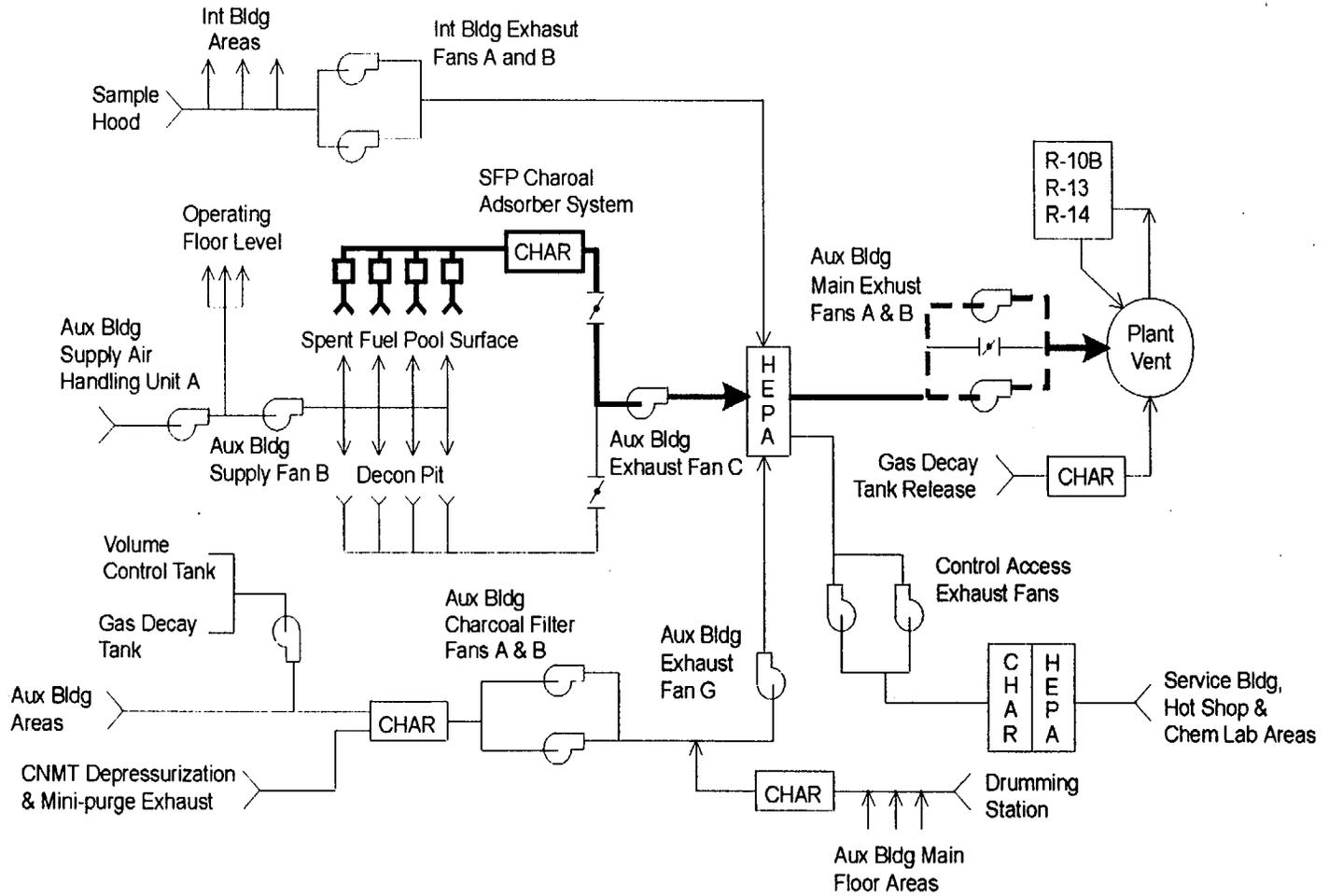
SR 3.7.10.3

This SR verifies that the required SFP Charcoal Adsorber System testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The SFP Charcoal Adsorber System filter tests are in general accordance with Regulatory Guide 1.52 (Ref. 5). The VFTP includes testing charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). There is no minimum required flowrate through the SFP charcoal adsorbers since SR 3.7.10.2 requires verification that a negative pressure is maintained during fuel movement in the Auxiliary Building. As long as this minimum pressure is maintained by drawing air from the surface of the SFP through the SFP charcoal adsorbers, the assumptions of the accident analyses are met. Specific test frequencies and additional information are discussed in detail in the VFTP. However, the maximum surveillance interval for refueling outage tests is based on 24 month refueling cycles and not 18 month cycles as defined by Regulatory Guide 1.52 (Ref. 5).

REFERENCES

1. UFSAR, Section 9.4.2.
 2. UFSAR, Section 15.7.3.2.
 3. 10 CFR 100.
 4. Regulatory Guide 1.25, Rev. 0.
 5. Regulatory Guide 1.52, Rev. 2.
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Figure B 3.7.10-1
ABVS



Legend:

- Flowpath required by LCO (Aux Bldg Exhaust Fan C HEPA filter not required for LCO but Aux Bldg operating floor must be at a negative pressure)
- - -** 1 of 2 flowpaths required by LCO
- SFP Roughing filters

For illustration only

B 3.7 PLANT SYSTEMS

B 3.7.11 Spent Fuel Pool (SFP) Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel pool (SFP) meets the assumptions of iodine decontamination factors following a fuel handling accident. The specified water level provides protection against exceeding the offsite dose limits.

The SFP is a seismically designed structure located in the Auxiliary Building (Ref. 1). The pool is internally clad with stainless steel that has a leak chase system at each weld seam to minimize accidental drainage through the liner. The SFP is also provided with a barrier between the spent fuel storage racks and the fuel transfer system winch. This barrier, up to the height of the spent fuel racks, prevents inadvertent drainage of the SFP via the fuel transfer tube.

The SFP Cooling System is designed to maintain the pool $\leq 120^{\circ}\text{F}$ during normal conditions and refueling operations (Ref. 2). The cooling system normally takes suction near the surface of the SFP such that a failure of any pipe in the system will not drain the pool. The cooling system return line to the pool also contains a 0.25 inch vent hole located near the SFP surface level to prevent siphoning. Finally, control board alarms exist with respect to the SFP level and temperature. These features all help to prevent inadvertent draining of the SFP.

APPLICABLE
SAFETY
ANALYSES

The minimum water level in the SFP is an assumption of the fuel handling accident described in the UFSAR (Ref. 3) and Regulatory Guide 1.25 (Ref. 4). The resultant 2 hour thyroid dose per person at the exclusion area boundary as based on this assumption is a small fraction of the 10 CFR 100 (Ref. 5) limits.

Based on the requirements of Reference 4, there must be 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface during a fuel handling accident. With 23 ft of water available, the assumptions of Reference 4 can be used directly. These assumptions include the use of a decontamination factor of 100 in the analysis for iodine. A decontamination factor of 100 enables the analysis to assume that 99% of the total iodine released from the pellet to cladding gap of all dropped fuel assembly rods is retained by the SFP water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory.

In practice, this LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel storage racks, however, there may be < 23 ft of water between the top of the fuel bundle and the surface, indicated by the width of the bundle and difference between the top of the rack and active fuel. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although analysis shows that only the first few rows fail from a hypothetical maximum drop.

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

LCO	The SFP water level is required to be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required during movement of irradiated fuel assemblies within the SFP.
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APPLICABILITY	This LCO applies during movement of irradiated fuel assemblies in the spent fuel pool, since the potential for a release of fission products exists. Since a fuel handling accident can only occur during movement of fuel, this LCO is not applicable during other conditions. During refueling operations in MODE 6, the SFP water level (and boron concentration) are in equilibrium with the refueling water cavity. The water level under these conditions is then controlled by LCO 3.9.6, "Refueling Cavity Water Level" which requires the refueling cavity water level to be maintained ≥ 23 feet above the top of the reactor vessel flange. A refueling cavity water level of ≥ 23 feet above the top of the reactor vessel flange will result in > 23 feet of water above the top of the active fuel in the storage racks assuming that atmospheric pressure within containment and the Auxiliary Building are equivalent.
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ACTIONS

A.1

When the initial conditions assumed in the fuel handling accident analysis cannot be met, steps should be taken to preclude the accident from occurring. When the SFP water level is lower than the required level, the movement of irradiated fuel assemblies in the SFP is immediately suspended. This action effectively precludes the occurrence of a fuel handling accident. This does not preclude movement of a fuel assembly to a safe position (e.g., movement to an available rack position).

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply since if moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not be applicable. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE
REQUIREMENTS

SR 3.7.11.1

This SR verifies sufficient SFP water is available in the event of a fuel handling accident. The water level in the spent fuel pool must be checked periodically during movement of irradiated fuel assemblies to ensure the fuel handling accident assumptions are met. The 7 day Frequency is appropriate because the volume in the pool is normally stable and the SFP is designed to prevent drainage below 23 ft. Water level changes are controlled by plant procedures and are acceptable based on operating experience.

Verification of SFP water level can be accomplished by several means. The top of the upper SFP pump suction line is 23 ft above the fuel stored in the pool. If there is ≥ 23 ft of water above the reactor vessel flange (as required by LCO 3.9.6), with equal pressure in the containment and the Auxiliary Building, then at least 23 ft of water is available above the top of the active fuel in the storage racks.

In addition to the physical design features, there are two SFP level alarms (LAL 634) which are available to alert the operators of changing SFP level. A low level alarm will actuate when the SFP water level falls 4 inches or more from the normal level while a high level alarm will actuate when the SFP water level rises 4 inches or more from the normal level. These alarms must receive a calibration consistent with industry practices before they are to be used to meet this SR.

REFERENCES

1. UFSAR, Section 9.1.2.
 2. UFSAR, Section 9.1.3.
 3. UFSAR, Section 15.7.3.
 4. Regulatory Guide 1.25, Rev. 0.
 5. 10 CFR 100.11.
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B 3.7 PLANT SYSTEMS

B 3.7.12 Spent Fuel Pool (SFP) Boron Concentration

BASES

BACKGROUND

The water in the spent fuel pool (SFP) normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that a limiting k_{eff} of 0.95 be maintained in the absence of soluble boron. Hence, the design of both SFP regions is based on the use of unborated water such that the SFP design and configuration control (i.e., controlling the movement of the fuel assembly and checking the location of each assembly after movement) maintains each region in a subcritical condition during normal operation with the regions fully loaded. The Region 2 SFP design uses boraflex material that is secured to the rack structure. Testing has demonstrated that boraflex degradation is occurring such that boron must be credited to maintain $k_{\text{eff}} \leq 0.95$ at all times until a long-term solution is reached (Reference 5).

The double contingency principle discussed in ANSI N-16.1-1975 (Ref. 1) and Reference 2 allows credit for soluble boron under abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenarios are associated with the movement of fuel from Region 1 to Region 2, and accidental misloading of a fuel assembly in Region 2. Either scenario could potentially increase the reactivity of Region 2. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the storage racks with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with LCO 3.7.13, "Spent Fuel Pool (SFP) Storage" and by maintaining the minimum boron concentration required to address boraflex degradation. Within 7 days prior to movement of an assembly into a SFP region, it is necessary to perform SR 3.7.12.1. Prior to moving an assembly into a SFP region, it is also necessary to perform SR 3.7.13.1 or SR 3.7.13.2 as applicable.

APPLICABLE
SAFETY
ANALYSES

The postulated accidents in the SFP can be divided into two basic categories (Refs. 3 and 4). The first category are events which cause a loss of cooling in the SFP. Changes in the SFP temperature could result in an increase in positive reactivity. However, the positive reactivity is ultimately limited by a combination of voiding (which would result in the addition of negative reactivity) and the SFP geometry, and the high boron concentration in the SFP. The second category is related to the movement of fuel assemblies in the SFP (i.e., a fuel handling accident)

and is the most limiting accident scenario with respect to reactivity. The types of accidents within this category include an incorrectly transferred fuel assembly (e.g., transfer from Region 1 to Region 2 of an unirradiated or an insufficiently depleted fuel assembly) and a dropped fuel assembly. However, for both of these accidents, the negative reactivity effect of the soluble boron compensates for the increased reactivity. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents which credit use of the soluble boron may be limited to a small fraction of the total operating time.

The concentration of dissolved boron in the SFP satisfies Criterion 2 of the NRC Policy Statement.

LCO The SFP boron concentration is required to be ≥ 2300 ppm. With the assumed boraflex available per the original design of the SFP storage racks, this value is 450 ppm. The specified concentration of dissolved boron in the SFP preserves the assumptions used in the analyses of the potential critical accident scenarios as described in References 3 and 4 (i.e., a fuel handling accident) and the boraflex degradation issue described in Reference 5. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage.

APPLICABILITY This LCO applies whenever fuel assemblies are stored in the SFP to compensate for boraflex degradation to ensure the SFP k_{eff} remains ≤ 0.95 at all times. This is expected to be corrected by June 30, 2001 per Specification 4.3.1.1.b.

ACTIONS A.1 and A.2

When the concentration of boron in the SFP 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 efficiently achieved by immediately suspending the movement of fuel assemblies. The initiation of actions to restore concentration of boron is simultaneous with suspending movement of fuel assemblies. This is necessary to compensate for any boraflex degradation within the SFP racks.

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply since 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 3.7.12.1

This SR verifies that the concentration of boron in the SFP is within the limit. As long as this SR is met, the analyzed accidents are fully addressed. The 7 day Frequency is appropriate since the boron is credited with maintaining the SFP subcritical due to boraflex degradation. Also, the volume and boron concentration in the pool is normally stable and all water level changes and boron concentration changes are controlled by plant procedures.

This SR is required to be performed prior to fuel assembly movement into Region 1 or Region 2 and must continue to be performed until the necessary SFP verification is accomplished (i.e., SR 3.7.13.1 and SR 3.7.13.2).

REFERENCES

1. ANSI N16.1-1975, "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."
 2. Letter from B.K. Grimes, NRC, to All Power Reactor Licensees, Subject: "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978.
 3. Framatome Technologies, Inc., "R.E. Ginna Nuclear Power Plant, Spent Fuel Pool Re-racking Licensing Report," Section 4, February 1997.
 4. UFSAR, Section 15.7.3.
 5. Letter from R.C. Mecredy, RG&E, to G.S. Vissing, NRC, Subject: "Boraflex Degradation," dated March 30, 1998.
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B 3.7 PLANT SYSTEMS

B 3.7.13 Spent Fuel Pool (SFP) Storage

BASES

BACKGROUND

The spent fuel pool (SFP) is divided into two separate and distinct regions (see Figure B 3.7.13-1) which, for the purpose of criticality considerations, are considered as separate pools (Ref. 1). Region 1, with 294 storage positions, is designed to accommodate new or spent fuel utilizing a checkerboard arrangement. All fuel assemblies stored in Region 1 must have a k-infinity that is ≤ 1.458 . The existing design uses Integral Fuel Burnable Absorbers (IFBAs) as the poison for fuel assemblies with enrichments > 4.05 wt% to help achieve this k-infinity limit. IFBAs consist of neutron absorbing material which provides equivalencing reactivity holddown (i.e., neutron poison) that allows storage of higher enrichment fuel. The neutron absorbing material is a non-removable or integral part of the fuel assembly once it is applied. The infinite multiplication factor, K-infinity, is a reference criticality point of each fuel assembly that if maintained ≤ 1.458 , will result in a $k_{\text{eff}} \leq 0.95$ for Region 1. The K-infinity limit is derived for constant conditions of normal reactor core configuration (i.e., typical geometry of fuel assemblies in vertical position arranged in an infinite array) at cold conditions (i.e., 68°F and 14.7 psia). Fuel assemblies with minimum burnups above the curve in Figure 3.7.13-1 (area A) may be stored at any location within Region 1. Fuel assemblies with minimum burnups below the curve in Figure 3.7.13-1 (area B) may be stored in cells with lead-in funnels only.

Region 2, with 1075 storage positions, is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within the acceptable domains according to Figure 3.7.13-2, in the accompanying LCO. Fuel assemblies with initial enrichments and burnups within domain A1 of Figure 3.7.13-2 may be stored in any location in Region 2. Fuel assemblies with initial enrichments and burnups within domain A2 of Figure 3.7.13-2 shall be stored face-adjacent to a Type A1 or A2 assembly, or a water cell (empty cell). Fuel assemblies with initial enrichments and burnups within domain B of Figure 3.7.13-2 shall be stored face-adjacent to a Type A1 assembly or a water cell (empty cell). Fuel assemblies with initial enrichments and burnups within domain C of Figure 3.7.13-2 shall be stored face-adjacent to a water cell (empty cell) only. The word "face-adjacent" on Figure 3.7.13-2 is defined to mean that the flat surface of a fuel assembly in one cell faces the flat surface of the fuel assembly in the next cell. The storage of fuel assemblies which are within the acceptable range of Figure 3.7.13-2 in Region 2 ensures a $K_{\text{eff}} \leq 0.95$ in this region.

Consolidated rod storage canisters can also be stored in either region in the SFP provided that the minimum burnup of Figure 3.7.13-1 and Figure 3.7.13-2 are met (Ref. 2). The canisters are stainless steel containers which contain the fuel rods of a maximum of two fuel assemblies (i.e., 358 rods). All bowed, broken, or otherwise failed fuel rods are first stored in a stainless steel tube of 0.75 inch outer diameter before being placed in a canister. Each canister will accommodate 110 failed fuel rod tubes.

The water in the SFP normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that a limiting k_{eff} of 0.95 be maintained in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water such that the SFP design and configuration control (i.e., controlling the movement of the fuel assembly and checking the location of each assembly after movement) maintains each region in a subcritical condition during normal operation with the regions fully loaded. The SFP design uses boraflex material that is secured to the rack structure. Testing has demonstrated that boraflex degradation is occurring such that boron must be temporarily credited to maintain $k_{\text{eff}} \leq 0.95$ at all times until a long-term solution is reached (Reference 6).

The double contingency principle discussed in ANSI N16.1-1975 (Ref. 3) and Reference 4 allows credit for soluble boron under abnormal or accident conditions, since only a single accident need be considered at one time. For example, the most severe accident scenarios are associated with the movement of fuel from Region 1 to Region 2, and accidental misloading of a fuel assembly in Region 2. Either scenario could potentially increase the reactivity of Region 2. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the storage racks with no movement of assemblies may therefore be achieved by controlling the location of each assembly in accordance with this LCO and by maintaining the minimum boron concentration required to address boraflex degradation per LCO 3.7.12. Within 7 days prior to movement of an assembly into a SFP region, it is necessary to perform SR 3.7.12.1. Prior to moving an assembly into a SFP region, it is also necessary to perform SR 3.7.13.1 or SR 3.7.13.2 as applicable.

APPLICABLE
SAFETY
ANALYSES

The postulated accidents in the SFP can be divided into two basic categories (Refs. 2 and 5). The first category are events which cause a loss of cooling in the SFP. Changes in the SFP temperature could result in an increase in positive reactivity. However, the positive reactivity is ultimately limited by a combination of voiding (which would result in the addition of negative reactivity), the SFP geometry, and the high boron concentration in the SFP. The second category is related to the movement of fuel assemblies in the SFP (i.e., a fuel handling accident) and is the most limiting accident scenario with respect to reactivity. The types of accidents within this category include an incorrectly transferred fuel assembly (e.g., transfer from Region 1 to Region 2 of an unirradiated or an insufficiently depleted fuel assembly) and a dropped fuel assembly. However, for both of these accidents, the negative reactivity effect of the soluble boron compensates for the increased reactivity. By closely controlling the movement of each assembly and by checking the location of each assembly after movement, the time period for potential accidents may be limited to a small fraction of the total operating time.

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

LCO

The restrictions on the placement of fuel assemblies within the SFP ensure the k_{eff} of the SFP will always remain < 0.95 . For fuel assemblies stored in Region 1, each assembly must have a K-infinity of ≤ 1.458 with initial enrichment and burnup within the acceptable area of Figure 3.7.13-1. For fuel assemblies stored in Region 2, initial enrichment and burnup shall be within the acceptable area of Figure 3.7.13-2. The word "face-adjacent" on Figure 3.7.13-2 is defined to mean that the flat surface of a fuel assembly in one cell faces the flat surface of the assembly in the next cell.

The x-axis of both figures is the nominal U-235 enrichment wt% which does not include the ± 0.05 wt% tolerance that is allowed for fuel manufacturing and listed in Specification 4.3.1.1.

APPLICABILITY This LCO applies whenever any fuel assembly is stored in the SFP.

ACTIONS A.1

When the configuration of fuel assemblies stored in either Region 1 or Region 2 of the SFP is not within the LCO limits, the immediate action is to initiate action to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Specification 4.3.1.1. This compliance can be made by relocating the fuel assembly to a different region or to an acceptable new location within the same region.

Required ACTION A.1 is modified by a Note indicating that LCO 3.0.3 does not apply since 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 ACTION is independent of reactor operation. Therefore, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE SR 3.7.13.1
REQUIREMENTS

This SR verifies by administrative means that the K-infinity of each fuel assembly is ≤ 1.458 prior to storage in Region 1 and that the initial enrichment and burnup is in accordance with Figure 3.7.13-1. If the initial enrichment of a fuel assembly is ≤ 4.05 wt%, a K-infinity of ≤ 1.458 is always maintained. For fuel assemblies with enrichment > 4.05 wt%, a minimum number of IFBAs must be present in each fuel assembly such that k-infinity ≤ 1.458 prior to storage in Region 1. This verification is only required once for each fuel assembly since the burnable poisons, if required, are an integral part of the fuel assembly and will not be removed. The initial enrichment of each assembly will also not change (i.e., increase) while partially burned assemblies are less reactive than when they were new (i.e., fresh). Performance of this SR ensures compliance with Specification 4.3.1.1.

Though not required for this LCO, this SR must also be performed after completion of fuel movement within Region 1 to exit the Applicability of LCO 3.7.12, "SFP Boron Concentration."

SR 3.7.13.2

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.13-2 in the accompanying LCO prior to storage in Region 2. Once a fuel assembly has been verified to be within the acceptable range of Figure 3.7.13-1, further verifications are no longer required since the initial enrichment or burnup will not adversely change. For fuel assemblies in the unacceptable range of Figure 3.7.13-1, performance of this SR will ensure compliance with Specification 4.3.1.1.

Though not required for this LCO, this SR must also be performed after completion of fuel movement within Region 2 to exit the Applicability of LCO 3.7.12.

REFERENCES

1. UFSAR, Section 9.1.2.
 2. Framatome Technologies, Inc., "R.E. Ginna Nuclear Power Plant, Spent Fuel Pool Re-racking Licensing Report," Section 4, February 1997.
 3. ANSI N16.1-1975, "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."
 4. Letter from B.K. Grimes, NRC, to All Power Reactor Licensees, Subject: "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978.
 5. UFSAR, Section 15.7.3.
 6. Letter from R.C. Mecredy, RG&E, to G.S. Vissing, NRC, Subject: "Boraflex Degradation," dated March 30, 1998.
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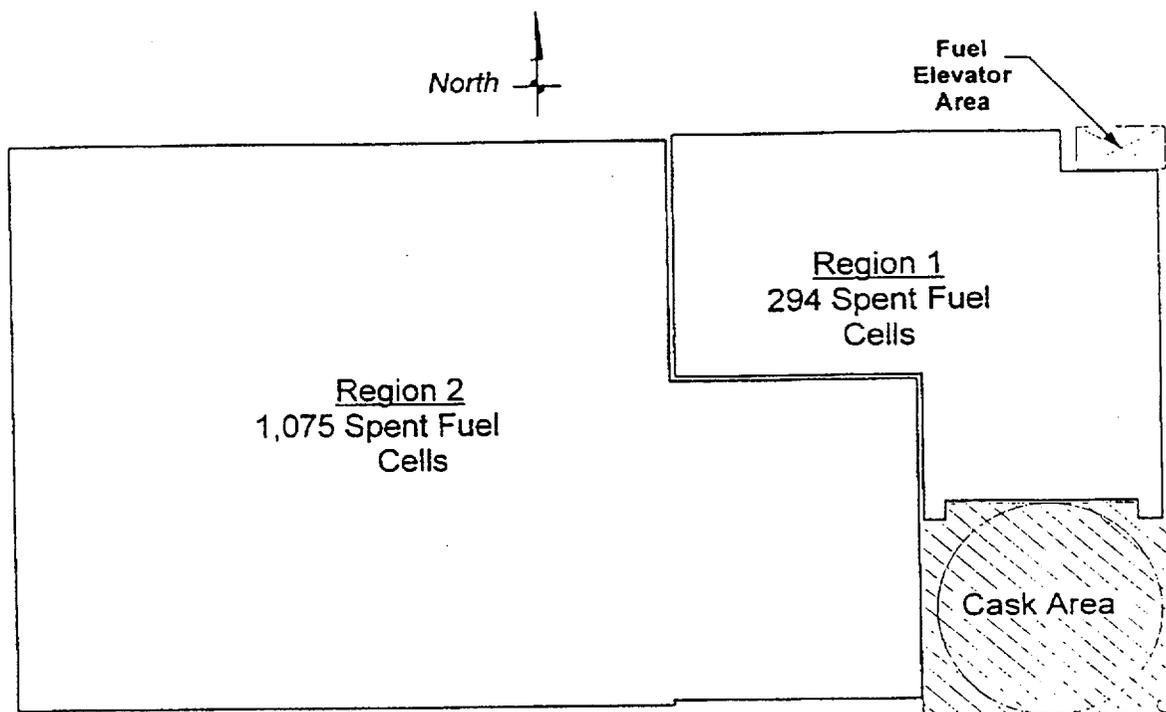


Figure B 3.7.13-1
Spent Fuel Pool

B 3.7 PLANT SYSTEMS

B 3.7.14 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator (SG) tube leakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes can be observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and Design Basis accidents (DBAs).

This limit is based on an activity value that might be expected from a 0.1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 $\mu\text{Ci/gm}$ (LCO 3.4.16, "RCS Specific Activity"). A steam line break (SLB) is assumed to result in the release of the noble gas and iodine activity contained in the SG inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours). I-131, with a half life of 8.04 days, concentrates faster than it decays, but does not reach equilibrium because of blowdown and other losses.

With the specified activity limit, the resultant 2 hour thyroid dose to a person at the exclusion area boundary (EAB) would be approximately 10 rem if the main steam safety valves (MSSVs) were left open for 2 hours following a trip from full power. Operating a plant at the allowable limits could result in a 2 hour EAB exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits.

APPLICABLE
SAFETY
ANALYSES

The accident analysis of the SLB, (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 $\mu\text{Ci/gm}$ DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an SLB do not exceed a small fraction of the plant EAB limits (Ref. 1) for whole body and thyroid dose rates.

With the loss of offsite power, the remaining SG is available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric relief valve (ARV). The Auxiliary Feedwater System supplies the necessary makeup to the SG. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the SG connected to the failed steam line is assumed to be released directly to the environment within 60 seconds. The unaffected SG is assumed to discharge steam and any entrained activity through the MSSVs and ARV for the initial two hours of the event. Primary coolant was assumed to be $3.0 \mu\text{Ci}/\text{gm}$ for this analysis based on previously allowed limits which is a factor of three greater than current limits specified in LCO 3.4.16. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

The specific activity of the secondary coolant is required to be $\leq 0.10 \mu\text{Ci}/\text{gm}$ DOSE EQUIVALENT I-131 to limit the radiological consequences of a DBA to a small fraction of the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the plant in an operational MODE that would minimize the radiological consequences of a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere from a SLB.

In MODES 5 and 6, the SGs are not being used for heat removal. Both the RCS and SGs are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

A.1 and A.2

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity is not within limits the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE
REQUIREMENTS

SR 3.7.14.1

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. 10 CFR 100.11.
 2. Letter from D. M. Crutchfield, NRC, to J.E. Maier, RG&E, Subject: "SEP Topic, XV-2, Spectrum of Steam System Piping Failures Inside and Outside Containment; XV-12, Spectrum of Rod Ejection Accidents; XV-16, Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment; XV-17, Steam Generator Tube Failure; and XV-20, Radiological Consequences of Fuel Damaging Accidents - R.E. Ginna," dated September 24, 1981.
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B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 AC Instrument Bus Sources - MODES 5 and 6

BASES

BACKGROUND

The Background section of the Bases for LCO 3.8.7, "AC Instrument Bus Sources - MODES 1, 2, 3, and 4" is applicable to these Bases, with the following modifications.

In MODE 5 or 6, the number of required AC instrument buses may be reduced since less energy is retained within the reactor coolant system than during higher MODES. Also, a significant number of required testing and maintenance activities must be performed under these conditions such that equipment and systems, including the AC instrument bus sources, must be removed from service. The minimum required AC instrument bus electrical subsystem is based on the requirements of LCO 3.8.10, "Distribution Systems - MODES 5 and 6."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC instrument bus power sources to each required AC instrument bus during MODES 5 and 6 ensures that:

- a. Systems needed to mitigate a fuel handling accident are available;
- b. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and
- c. Instrumentation and control capability is available for monitoring and maintaining the plant in a cold shutdown condition or refueling condition.

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

- a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.
- b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.
- c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.
- d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) for systems assumed to function during an event.

In the event of an accident while in MODE 5 or 6, this LCO ensures the capability to support systems necessary to mitigate the postulated events during shutdown, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

The AC instrument bus power sources satisfy Criterion 3 of the NRC Policy Statement.

LCO

Maintaining the required AC instrument bus sources OPERABLE ensures that the redundancy incorporated into the design of the instrumentation and controls required during MODES 5 and 6 is maintained. The two inverters ensure an uninterruptible supply of AC electrical power to AC Instrument Bus A and C even if the 480 V safeguards buses are de-energized. The Class 1E 480 V safeguard bus supply to Instrument Bus B provides a reliable source. The non-Class 1E CVT powered from offsite power provides an available power source, if required.

(continued)

BASES

LCO
(continued)

For an inverter to be OPERABLE, the associated instrument bus must be powered by the inverter with output voltage within tolerances with power input to the inverter from a 125 VDC power source (see LCO 3.8.5, "DC Sources - MODES 5 and 6).

For a Class 1E CVT to be OPERABLE, the associated instrument bus must be powered by the CVT with the output voltage within tolerances with power to the CVT from a Class 1E 480 V safeguards bus. The 480 V safeguards bus must be powered from an acceptable AC source (see LCO 3.8.2, "AC Sources - MODES 5 and 6).

The non-Class 1E CVT may be used to power one required instrument bus provided the instrument bus output voltage is within tolerances and the redundant required instrument bus is capable of being powered from an OPERABLE DG and that instrument bus has output voltages within tolerances.

The instrument bus power sources must be sufficiently independent such that a loss of all offsite power sources, a loss of onsite standby power, or a worst case single failure does not affect more than one required instrument bus. This ensures the availability of sufficient power to the required AC instrument buses to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The AC instrument power sources required to be OPERABLE in MODES 5 and 6 provide assurance that systems required to mitigate the effects of a DBA and to maintain the plant in the cold shutdown or refueling condition are available.

AC Instrument Bus power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

(continued)

BASES (continued)

ACTIONS

A.1

Although two trains may be required by LCO 3.8.10, "Distribution Systems - MODES 5 and 6," the remaining OPERABLE AC instrument bus train may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS and operations with a potential for positive reactivity additions. By allowing the option to declare required features inoperable with the associated AC instrument bus power source inoperable, appropriate restrictions will be implemented in accordance with the LCO ACTIONS of the affected required features. This condition must be entered when the inverters for Instrument Bus A or C are required and inoperable, or the Class 1E CVT for Instrument Bus B is required and inoperable.

A.2.1, A.2.2, A.2.3, and A.2.4

With one or more required AC instrument bus power sources inoperable, the option exists to declare all required features inoperable per Required Action A.1. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. Therefore, immediate suspension of CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions is an acceptable option to Required Action A.1. Performance of Required Actions A.2.1, A.2.2, and A.2.3 shall not preclude completion of movement of a component to a safe position or normal cooldown of the coolant volume for the purpose of system temperature control within established procedures.

It is further required to immediately initiate action to restore the required AC instrument bus power source and to continue this action until restoration is accomplished in order to provide the necessary power to the plant safety systems.

(continued)

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, and A.2.4 (continued)

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC instrument bus power source should be completed as quickly as possible in order to minimize the time the plant safety systems may be without power or powered from an alternate power source.

SURVEILLANCE
REQUIREMENTS

SR 3.8.8.1

This SR verifies correct static switch alignment to the required AC instrument buses. This SR verifies that the inverter is functioning properly and the AC instrument bus is energized from the inverter. The verification ensures that the required power is available for the instrumentation connected to the AC instrument bus. The Frequency of 7 days takes into account the redundant capability of the inverter and other indications available in the control room that alert the operator to inverter malfunctions.

SR 3.8.8.2

This SR verifies the correct Class 1E CVT alignment when Instrument Bus B is required. This verifies that the Class 1E CVT is functioning properly and supplying power to AC Instrument Bus B. The verification ensures that the required power is available for the instrumentation of the RPS and ESF connected to the AC instrument bus. The Frequency of 7 days takes into account the redundant instrument buses and other indications available in the control room that alert the operator to the Class 1E CVT malfunctions.

REFERENCES

None.

BASES

LCO
(continued)

c. Technical Support Center battery connections to DC power Battery A and B (Ref. 6)

TSC/Battery A Fused Disconnect Switch (Note 3)
TSC/Battery B Fused Disconnect Switch (Note 3)

Notes:

1. If tie breaker/connection is closed such that both trains are connected, declare both electrical power distribution subsystems inoperable.
2. If tie breaker is closed with Bus 15 (or Bus 13) being supplied by Transformer No. 1 (i.e., the turbine/generator) declare electrical power distribution Train B (or Train A) inoperable. If Bus 15 (or bus 13) is being supplied by offsite power, tie breaker may be closed without impacting LCO.
3. An individual TSC/Battery fused disconnect switch may be closed with the affected DC distribution subsystem considered OPERABLE provided:
 - i. No single failure, malfunction, or operator action can lead to cross-tying the affected DC distribution subsystem to the redundant train or the TSC battery, and
 - ii. Any connected loads have been evaluated for the loading affects on the supply source and the adequacy of fault protection.

(continued)

BASES

LCO
(continued)

The trains as specified in Table B 3.8.9-1 only identify the major AC, DC, and AC instrument bus electrical power distribution subsystem components. A train is defined to begin from the boundary of the power source for the respective subsystem (as defined in the power source LCOs), and continues up to the isolation device for the supplied safety related or ESF component (e.g., safety injection pump). The isolation device for the supplied safety related or ESF component is only considered part of the train when the device is not capable of opening to isolate the failed component from the train (e.g., breaker unable to open an overcurrent). Otherwise, the failure of the isolation device to close to provide power to the component is addressed by the respective component's LCO. The isolation device for nonsafety related components are considered part of the train since these devices must be available to protect the safety related functions. Therefore, the train boundary essentially ends at the motor control center or bus which supplies multiple components.

The inoperability of any component within the above defined train boundaries renders the train inoperable.

APPLICABILITY

The electrical power distribution subsystems 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.

Electrical power distribution subsystem requirements for MODES 5 and 6 are addressed in LCO 3.8.10, "Distribution Systems - MODES 5 and 6."

(continued)

BASES (continued)

ACTIONS

A.1

With one AC electrical power distribution train inoperable, the remaining AC electrical power distribution train is capable of supporting the minimum safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition. The overall reliability is reduced, however, because a single failure in the remaining AC power distribution train could result in the minimum required ESF functions not being supported. Therefore, the required AC buses, load centers, motor control centers, and distribution panels which comprise a train must be restored to OPERABLE status within 8 hours.

The worst case Condition A scenario is one train without AC power (i.e., no offsite power to the train and the associated DG inoperable). In this Condition, the plant is more vulnerable to a complete loss of AC power.

The Completion Time for restoring the inoperable train before requiring a plant shutdown is limited to 8 hours because of:

- a. The potential for decreased safety if the plant operator's attention is diverted from the evaluations and actions necessary to restore power to the affected train; and
- b. The potential for an event in conjunction with a single failure of a redundant component in the OPERABLE train with AC power which results in the loss of multiple safety functions.

(continued)
