5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

5.6.1 Occupational Radiation Exposure Report

-----NOTE-----
A single submittal may be made for ANO. The submittal should combine sections common to both units.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person-rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

5.6.2 Annual Radiological Environmental Operating Report

-----NOTE-----
A single submittal may be made for ANO. The submittal should combine sections common to both units.

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

---NOTE---

A single submittal may be made for ANO. The submittal shall combine sections common to both units . The submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

2.1.1 Variable Low RCS Pressure – Temperature Protective Limits
3.1.1 SHUTDOWN MARGIN (SDM)
3.1.8 PHYSICS TESTS Exceptions – MODE 1
3.1.9 PHYSICS TEST Exceptions - MODE 2
3.2.1 Regulating Rod Insertion Limits
3.2.2 AXIAL POWER SHAPING RODS (APSR) Insertion Limits
3.2.3 AXIAL POWER IMBALANCE Operating Limits
3.2.4 QUADRANT POWER TILT (QPT)
3.2.5 Power Peaking
3.3.1 Reactor Protection System (RPS) Instrumentation
3.4.1 RCS Pressure, Temperature, and Flow DNB limits
3.4.4 RCS Loops – MODES 1 and 2
3.9.1 Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

Babcock & Wilcox Topical Report BAW-10179-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses" (the approved revision at the time the reload analyses are performed). The approved revision number shall be identified in the COLR.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

5.6.6 Reactor Building Inspection Report

Any degradation exceeding the acceptance criteria of the containment structure detected during the tests required by the Containment Inspection Program shall undergo an engineering evaluation within 60 days of the completion of the inspection surveillance. The results of the engineering evaluation shall be reported to the NRC within an additional 30 days of the time the evaluation is completed. The report shall include the cause of the condition that does not meet the acceptance criteria, the applicability of the conditions to the other unit, the acceptability of the concrete containment without repair of the item, whether or not repair or replacement is required and, if required, the extent, method, and completion date of necessary repairs, and the extent, nature, and frequency of additional examinations.

5.6.7 Steam Generator Tube Surveillance Reports

a. Following each inservice inspection of steam generator tubes, the complete results of the inspection shall be reported to the NRC. This report, to be submitted within 90 days of inspection completion, shall include:

   1. Number and extent of tubes inspected;

   2. Location and percent of wall-thickness penetration for each indication of an imperfection;

   3. Identification of tubes plugged and tubes sleeved;

   4. Number of tubes repaired by rerolling and number of indications detected in the new roll area of the repaired tubes;

   5. Summary of the condition monitoring and operational assessment results when applying TEC alternate repair criteria; and

   6. Summary of the condition monitoring and the operational assessment results (including growth) when applying the upper tubesheet ODIGA alternate repair criteria.

b. In addition, the Commission shall be notified of the results of steam generator tube inspections which fall into Category C-3 as denoted in Table 5.5.9-2 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.

b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP), or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously displays radiation dose rates in the area; or

2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation

a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:

   1. All such door and gate keys shall be maintained under the administrative control of the shift supervisor, radiation protection manager, or his or her designee.

   2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.

b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.

d. Each individual or group entering such an area shall possess:

1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or

2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or

3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

   (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or

   (ii) Be under the surveillance as specified in the RWP, or equivalent, while in the area by means of closed circuit television, or personnel qualified in radiation protection procedures responsible for controlling personnel radiation exposure in the area and with the means to communicate with individuals in the area who are covered by such surveillance.

4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

e. Except for individuals qualified in radiation protection procedures, or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them. These continuously escorted personnel will receive a pre-job briefing prior to entry into such areas. This dose rate determination, knowledge, and pre-job briefing does not require documentation prior to initial entry.

f. Such individual areas that are within a larger area where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate, nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.

B 2.0 SAFETY LIMITS (SLS)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that reactor core SLs ensure specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and abnormalities. This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur and by requiring that the fuel centerline temperature stays below the melting temperature.

Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature and pressure can be related to DNB through the use of a critical heat flux (CHF) correlation. The BAW-2 (Ref. 2) and BWC (Ref. 3) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady state operation, normal operational transients and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC).

The 95 percent confidence level that DNB will not occur is preserved by ensuring that the DNBR remains greater than the DNBR design limit based on the applicable CHF correlation for the core design. In the development of the applicable DNBR design limit, uncertainties in the core state variables, power peaking factors, manufacturing-related parameters, and the CHF correlation may be statistically combined to determine a statistical DNBR design limit. This statistical design limit protects the respective CHF design limit. Additional retained thermal margin may also be applied to the statistical DNBR design limit to yield a higher thermal design limit for use in establishing DNB-based core safety and operating limits. In all cases, application of statistical DNB design methods preserves a 95 percent probability at a 95 percent confidence level that DNB will not occur (Ref. 4).

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. The maximum fuel centerline temperatures are given by the relationships defined in SL 2.1.1.1 for the respective fuel designs and are dependent on whether the TACO2 (Ref. 5) or TACO3 (Ref. 6) analysis was utilized. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

BACKGROUND (continued)

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding-water (zirconium-water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding. The oxidized cladding then exists in a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) prevents violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and abnormalities. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience fuel centerline melting.

The RPS setpoints, in combination with all the LCOs, are designed to prevent any analyzed combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a DNBR of less than the DNBR limit and preclude the existence of flow instabilities (Ref. 7).

Automatic enforcement of these reactor core SLs is provided by the following:

a. RCS High Pressure trip;

b. RCS Low Pressure trip;

c. Nuclear Overpower trip;

d. RCS Variable Low Pressure trip (also known as Pressure Temperature Trip);

e. Reactor Coolant Pump to Power trip;

f. Nuclear Overpower RCS Flow and AXIAL POWER IMBALANCE trip; and

g. RCS High Temperature trip.

APPLICABLE SAFETY ANALYSES (continued)

The SL represents a design requirement for establishing the RPS trip setpoints identified previously.

SAFETY LIMITS

SL 2.1.1.1, SL 2.1.1.2, and SL 2.1.1.3 ensure that the minimum DNBR is not less than the safety analyses limit and that fuel centerline temperature stays below the melting point, or the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or the exit quality is within the limits defined by the DNBR correlation. In addition, the COLR identifies the pressure/temperature operating region that keeps the reactor from reaching an SL when operating up to design power.

The COLR presents the most limiting condition of pressure/temperature combinations for all possible reactor coolant pump maximum THERMAL POWER combinations. Analyses have been performed which bound the three pump and two pump (one pump in each loop) allowed operating conditions based on the expected minimum flow rates and maximum ALLOWABLE THERMAL POWER for these operating conditions.

The SLs are preserved by monitoring the process variable AXIAL POWER IMBALANCE to ensure that the core operates within the fuel design criteria. AXIAL POWER IMBALANCE protective limits are preserved by their corresponding RPS setpoints in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," and are provided in the COLR. The trip setpoints are derived by adjusting the measurement system independent AXIAL POWER IMBALANCE protective limits given in the COLR to allow for measurement system observability and instrumentation errors.

The AXIAL POWER IMBALANCE protective limits are separate and distinct from the AXIAL POWER IMBALANCE operating limits defined by LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits." The AXIAL POWER IMBALANCE operating limits in LCO 3.2.3, also specified in the COLR, preserve initial conditions of the safety analyses but are not reactor core SLs.

APPLICABILITY

SL 2.1.1.1, SL 2.1.1.2, and SL 2.1.1.3 only apply in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. Automatic protection actions serve to prevent RCS heatup to reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1.

In MODES 3, 4, 5, and 6, Applicability is not required, since the reactor is not generating significant THERMAL POWER

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

2.2.1 AND 2.2.2

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, the requirement to go to MODE 3 places the plant in a MODE in which these SLs are not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the plant to a MODE of operation where these SLs are not applicable and reduces the probability of fuel damage.

2.2.5

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 8).

REFERENCES

1. SAR, Section 1.4, GDC 10.

2. BAW-10000A, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," Babcock & Wilcox, Lynchburg, VA, May 1976 .

3. BAW-10143P-A, "BWC Correlation of Critical Heat Flux," Babcock & Wilcox, Lynchburg, VA, April 1985.

4. BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," Rev. 2, Babcock & Wilcox, Lynchburg, VA, October 1997.

5. BAW-10141P-A, Rev. 1, "TACO2 Fuel Pin Performance Analysis," Babcock & Wilcox, Lynchburg, VA, June 1983.

5. BAW-10162P-A, "TACO3 Fuel Pin Thermal Analysis Code," Babcock & Wilcox, Lynchburg, VA, October 1989.

7. SAR, Chapters 3 & 14.

8. 10 CFR 50.72.

B 2.0 SAFETY LIMITS (SLS)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

In SAR, Section 1.4 (Ref. 1), GDC 14, "Reactor Coolant Pressure Boundary (RCPB)," and GDC 15, "Reactor Coolant System Design", address RCPB design and protection, respectively. The ANO-1 discussion regarding how GDC 15 is accomplished states that analysis and evaluation of all normal and abnormal operating conditions and transients are integrally related to all RCS and associated systems design. SAR Chapter 14 (Ref. 2) lists these abnormal operating conditions and transients and terms them "abnormalities". In addition, GDC 28, "Reactivity Limits" (Ref. 1), specifies that reactivity accidents including rod ejection do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psig. During normal operation and abnormalities, the RCS pressure is kept from exceeding the design pressure by more than 10% in order to remain in accordance with the design codes (Ref. 3 and 4). Hence, the safety limit is 2750 psig. To ensure system integrity, all RCS components were hydrostatically tested at 125% of design pressure prior to initial operation, according to the design code requirements. Inservice leak testing at not less than 2155 psig is also required, prior to MODE 2, following any opening of the reactor coolant system in accordance with ASME code, Section XI; IWA-5000. When performed at the end of refueling outages, this leak test also satisfies the requirements of IWB-2500, Table IWB-2500-1; Category B-P items B15.10, B15.20, B15.30, B15.40, B15.50, B15.60, and B15.70 for all Class I pressure retaining components (Ref. 5).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, operating in conjunction with the Reactor Protection System trip settings, ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME code for Nuclear Power Plant Components (Ref. 3). The design basis transient that is most influential for establishing the required relief capacity, and hence the valve size requirements and lift settings, is a rod withdrawal event from low power.

The startup event analysis (rod withdrawal at low power) (Ref. 2) is performed using conservative assumptions relative to pressure control devices.

APPLICABLE SAFETY ANALYSES (continued)

More specifically, no credit is taken for operation of the following:

a. Electromatic relief valve (ERV);

b. Steam line turbine bypass valves;

c. Control system runback of reactor and turbine power; and

d. Pressurizer spray valve.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under USAS B31.7 (Ref. 4), is 110% of design pressure. Therefore, the SL on maximum allowable RCS pressure is 2750 psig.

Overpressurization of the RCS can result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 6).

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES during overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized significantly.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the RCS pressure SL.

<u>2.2.3</u>

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 6).

SAFETY LIMIT VIOLATIONS (continued)

The allowed Completion Time of 1 hour is based on the importance of reducing power level to a MODE where the potential for challenges to safety systems is minimized.

2.2.4

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes.

Exceeding the RCS pressure SL in MODE 3, 4, or 5 is potentially more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. This action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

2.2.5

If the RCS pressure SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 7).

REFERENCES

1. SAR, Section 1.4, GDC 14, GDC 15, and GDC 28, 1988.
2. SAR, Chapter 14.
3. ASME Boiler and Pressure Vessel Code, Section III, 1965-S67, Article NB-7000.
4. USAS B31.7, Nuclear Power Piping, 1969.
5. ASME Boiler and Pressure Vessel Code, Section XI, Article IW-5000.
6. 10 CFR 100.
7. 10 CFR 50.72.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs	LCO 3.0.1 through LCO 3.0.7 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

LCO 3.0.1	LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the unit is in the MODES or other specified conditions of the Applicability statement of each Specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to OPERABLE status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a shutdown may be required to place the unit in a MODE or condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation of the unit that is not further restricted by the Completion Time. In this case, compliance with the Required Actions provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specification.

BASES

LCO APPLICABILITY (continued)

LCO 3.0.2 (continued)

The nature of some Required Actions of some Conditions necessitates that, once the Condition is entered, the Required Actions must be completed even though the associated Conditions no longer exist. The individual LCO's ACTIONS specify the Required Actions where this is the case. An example of this is in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

The Completion Times of the Required Actions are also applicable when a system or component is removed from service intentionally. Reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Additionally, if intentional entry into ACTIONS would result in redundant equipment being inoperable, alternatives should be used instead. Doing so limits the time both subsystems/trains of a safety function are inoperable and limits the time conditions exist which may result in LCO 3.0.3 being entered. Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed.

When a change in MODE or other specified condition is required to comply with Required Actions, the unit may enter a MODE or other specified condition in which another Specification becomes applicable. In this case, the Completion Times of the associated Required Actions would apply from the point in time that the new Specification becomes applicable and the ACTIONS Condition(s) are entered.

LCO 3.0.3

LCO 3.0.3 establishes the actions that must be implemented when an LCO is not met and:

a. An associated Required Action and Completion Time is not met and no other Condition applies; or

b. The condition of the unit is not specifically addressed by the associated ACTIONS. This means that no combination of Conditions stated in the ACTIONS can be made that exactly corresponds to the actual condition of the unit. Sometimes, possible combinations of Conditions are such that entering LCO 3.0.3 is warranted; in such cases, the ACTIONS specifically state a Condition corresponding to such combinations and also that LCO 3.0.3 be entered immediately.

BASES

LCO APPLICABILITY (continued)

LCO 3.0.3 (continued)

This Specification delineates the time limits for placing the unit in a safe MODE or other specified condition when operation cannot be maintained within the limits for safe operation as defined by the LCO and its ACTIONS. It is not intended to be used as an operational convenience that permits routine voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

Upon entering LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 are consistent with the discussion of Section 1.3, Completion Times.

A unit shutdown required in accordance with LCO 3.0.3 may be terminated and LCO 3.0.3 exited if any of the following occurs:

a. The LCO is now met.

b. A Condition exists for which the Required Actions have now been performed.

c. ACTIONS exist that do not have expired Completion Times. These Completion Times are applicable from the point in time that the Condition is initially entered and not from the time LCO 3.0.3 is exited.

The time limits of LCO 3.0.3 allow 37 hours for the unit to be in MODE 5 when a shutdown is required during MODE 1 operation. If the unit is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE applies. If a lower MODE is reached in less time than allowed, however, the total allowable time to reach MODE 5, or other applicable MODE, is not reduced. For example, if MODE 3 is reached in 2 hours, then the time allowed for reaching MODE 4 is the next 11 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to MODE 1, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

BASES

LCO APPLICABILITY (continued)

LCO 3.0.3 (continued)

In MODES 1, 2, 3, and 4, LCO 3.0.3 provides actions for Conditions not covered in other Specifications. The requirements of LCO 3.0.3 do not apply in MODES 5 and 6 because the unit is already in the most restrictive Condition required by LCO 3.0.3. The requirements of LCO 3.0.3 do not apply in other specified conditions of the Applicability (unless in MODE 1, 2, 3, or 4) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

Exceptions to LCO 3.0.3 are provided in instances where requiring a unit shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the unit. An example of this is in LCO 3.7.12, "Fuel Handling Area Ventilation System." LCO 3.7.12 has an Applicability of "During movement of irradiated fuel assemblies in the fuel handling area." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.12 are not met while in MODE 1, 2, 3, or 4, there is no safety benefit to be gained by placing the unit in a shutdown condition. The Required Action of LCO 3.7.12 of "Suspend movement of irradiated fuel assemblies in the fuel handling area" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

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

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

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

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

BASES

LCO APPLICABILITY (continued)

LCO 3.0.4 (continued)

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

Exceptions to LCO 3.0.4 are stated in the individual Specifications. The exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

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

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

LCO 3.0.5

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

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

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the required testing to demonstrate

BASES

LCO APPLICABILITY (continued)

LCO 3.0.5 (continued)

OPERABILITY. This Specification does not provide time to perform any other preventive or corrective maintenance.

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

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

LCO 3.0.6

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

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

BASES

LCO APPLICABILITY (continued)

LCO 3.0.6 (continued)

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry in Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.15, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross train checks to identify a loss of safety function for those support systems that support multiple and redundant safety systems are required. The cross train check verifies that the supported systems of the remaining OPERABLE support systems are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

This loss of safety function does not require the assumption of additional single failures or loss of offsite power. Since operation is being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross train inoperabilities. This explicit cross train verification for inoperable AC electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABILITY).

When a loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of function is solely due to a single Technical Specification support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump

BASES

LCO APPLICABILITY (continued)

LCO 3.0.6 (continued)

suction source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately address the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system.

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform special evolutions. Test Exception LCOs 3.1.8 and 3.1.9 allow specified Technical Specification (TS) requirements to be changed to permit performances of these special tests and operations, which otherwise could not be performed if required to comply with the requirements of these TS. Unless otherwise specified, all the other TS requirements remain unchanged. This will ensure all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed to perform the special test or operation will remain in effect.

The Applicability of a Test Exception LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with Test Exception LCOs is optional. A special operation may be performed either under the provisions of the appropriate Test Exception LCO or under the other applicable TS requirements. If it is desired to perform the special operation under the provisions of the Test Exception LCO, the requirements of the Test Exception LCO shall be followed.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the MODES or other specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the OPERABILITY of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known to be not met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a MODE or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a Special Test Exception (STE) LCO are only applicable when the STE LCO is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given MODE or other specified condition.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on inoperable equipment because the ACTIONS define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status.

BASES

SR APPLICABILITY (continued)

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

Some examples of this process are:

a. Emergency feedwater (EFW) pump turbine maintenance during refueling that requires testing at steam pressures > 800 psi. However, if other appropriate testing is satisfactorily completed, the EFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the EFW pump testing.

b. High pressure injection (HPI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPI considered OPERABLE. This allows operation to reach the specified pressure to complete the necessary post maintenance testing.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers unit operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply.

BASES

SR APPLICABILITY (continued)

SR 3.0.2 (continued)

These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is the Reactor Building Leakage Rate Testing Program.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides an adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay

BASES

SR APPLICABILITY (continued)

SR 3.0.3 (conntinued)

period of 24 hours to perform the Surveillance. SR 3.0.3 also provides a time limit for completion of Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Satisfactory completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a MODE or other specified condition in the Applicability.

This Specification ensures that system and component OPERABILITY requirements and variable limits are met before entry into MODES or other specified conditions in the Applicability for which these systems and components ensure safe operation of the unit. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components to OPERABLE status before entering an associated MODE or other specified condition in the Applicability.

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

BASES

SR APPLICABILITY (continued)

SR 3.0.4 (continued)

conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to MODE or other specified condition changes.

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

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

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

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

The reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions per GDC 26 (Ref. 1). In MODES 3, 4, and 5, SDM requirements provide sufficient reactivity margin to maintain the core subcritical during these conditions.

In MODES 1 and 2 while critical, SDM requirements are met by the worth of the withdrawn CONTROL RODS which provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and abnormalities. In MODE 2 while subcritical and in MODE 3, with all safety rods withdrawn and the RPS not in Shutdown Bypass, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all CONTROL RODS , assuming the single CONTROL ROD of highest reactivity worth is fully withdrawn. In MODES 3, 4, or 5, when all safety rods are not fully withdrawn or the RPS is in Shutdown Bypass, the SDM defines the degree of subcriticality required to be maintained, assuming the CONTROL ROD of highest reactivity worth is fully withdrawn.

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of CONTROL RODS and soluble boric acid in the Reactor Coolant System (RCS). In MODES 1 and 2, the CONTROL RODS can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, for analyzed events initiated in MODES 1 and 2, the CONTROL RODS, together with the Chemical Addition and Makeup and Purification System, provide SDM during power operation and are capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn (Ref. 1).

The Chemical Addition and Makeup and Purification System can compensate for fuel depletion, during operation and all xenon burnout reactivity changes, and maintain the reactor subcritical under cold conditions (Ref. 1).

During operation in MODES 1 and 2, SDM control is ensured by operating with the safety rods fully withdrawn (LCO 3.1.5, "Safety Rod Insertion Limits") and the regulating rods within the limits of LCO 3.2.1, "Regulating Rod Insertion Limits." In MODE 3, consideration must be given to the position of the safety rods and whether the RPS is in Shutdown Bypass in determining the required SDM. When the unit is

BACKGROUND (continued)

in MODES 3, 4, and 5, the SDM requirements are met by means of adjustments to the RCS boron concentration. Shutdown boron concentration requirements assume the highest worth rod is stuck in the fully withdrawn position to account for a postulated inoperable CONTROL ROD prior to reactor shutdown.

APPLICABLE SAFETY ANALYSES

For analyzed events in MODES 1 and 2 while critical, the minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and abnormalities, with assumption of the highest worth rod stuck out following a reactor trip.

In MODES 1 and 2 while critical, the acceptance criteria for SDM requirements are that specified acceptable fuel design limits are maintained. The SDM requirements must ensure that:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events; and

b. The reactivity transients associated with postulated accident conditions are controllable with acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for abnormalities, and ≤ 280 cal/gm energy deposition for the rod ejection accident).

In MODES 3, 4, and 5, the SDM requirements must ensure that the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

In MODES 1 and 2 while critical, SDM satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3). In MODE 2 while subcritical and in MODES 3, 4, and 5, SDM satisfies Criterion 4 of 10 CFR 50.36.

LCO

In MODES 1 and 2, and in MODE 3 when all safety rods are fully withdrawn and the RPS is not in Shutdown Bypass, SDM is a core design condition that can be ensured through CONTROL ROD positioning (regulating and safety groups) and through the soluble boron concentration.

In MODE 3, when all safety rods are not fully withdrawn or the RPS is in Shutdown Bypass, and in MODES 4 and 5, SDM represents a required degree of subcriticality that assumes the highest reactivity worth CONTROL ROD is fully withdrawn.

APPLICABILITY

In MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to ensure that the reactor remains subcritical.

In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5 and LCO 3.2.1. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met.

In the determination of the required combination of boration flow rate and boron source concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid addition tank (BAAT) or the borated water storage tank (BWST). The operator should borate with the best source available for the unit conditions.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

The SDM is verified by performing a reactivity balance calculation. The reactivity effects that are considered in the reactivity balance are dependent upon the operational MODE of the unit. In general, the reactivity balance includes the following reactivity effects:

a. RCS boron concentration;

b. CONTROL ROD position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.1.1 (continued)

f. Samarium concentration;

g. Isothermal temperature coefficient (ITC);

h. Moderator temperature coefficient (MTC); and

i. Doppler defect.

Using the ITC accounts for Doppler reactivity in this calculation when the reactor is subcritical or critical but below the point of adding heat (POAH), and the fuel temperature will be changing at the same rate as the RCS.

Using the MTC and Doppler defect accounts for the reactivity effects of power operation above the POAH.

The Frequency of 24 hours is based on the generally slow change in required boron concentration, and also allows sufficient time for the operator to collect the required data, which may include performing a boron concentration analysis, and complete the calculation.

REFERENCES

1. SAR, Section 1.4, GDC 26.

2. SAR, Chapter 3.

3. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Balance

BASES

BACKGROUND

According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and abnormalities. Therefore, the reactivity balance is used as a measure of the agreement between the predicted core reactivity and the actual core reactivity during power operation. The periodic confirmation of the predicted core reactivity is necessary to ensure that safety analyses of design basis transients and accidents remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, CONTROL ROD, or burnable poison worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity. These could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing the predicted core reactivity with the actual core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations in ensuring the reactor can be brought safely to cold, subcritical conditions. The difference between the actual and predicted core reactivity is commonly referred to as a reactivity anomaly.

When the reactor is critical in MODE 1 or 2, a reactivity balance exists where the net reactivity is zero (referred to as the actual core reactivity state). A comparison of predicted core reactivity and the actual core reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions and the net reactivity is known to be zero. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as soluble boron and burnable absorbers, producing zero net reactivity.

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and the fuel remaining from the previous cycle provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical, the excess positive reactivity of the fuel is compensated by burnable absorbers, CONTROL RODS, APSRs, thermal feedback from the fuel and moderator, fission product poisons (mainly xenon and samarium), epithermal energy neutron absorbers, neutron leakage and the reactor coolant system (RCS) boron concentration. During cycle operation, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the primary method of compensating for the reduction in excess reactivity is through a reduction in the RCS boron concentration.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for core reactivity are the establishment of the reactivity balance limit to ensure that unit operation is maintained within the assumptions of the safety analyses.

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation is, therefore, dependent upon an accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as CONTROL ROD withdrawal accidents or rod ejection accidents, are very sensitive to accurate prediction of core reactivity (Ref. 2). These accident analysis evaluations rely on computer codes which have been qualified against available test data, operating unit data, and analytical benchmarks. Monitoring the core reactivity balance ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the requirements for reactivity control during the operating cycle.

The comparison between the actual reactivity condition of the critical reactor and the predicted initial core reactivity provides an opportunity for the normalization of the calculational models used to predict core reactivity. If the predicted core reactivity and the actual core reactivity at reference core conditions at beginning of cycle (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict reactivity requirements may not be accurate. If reasonable agreement between the actual and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the predicted reactivity condition from the actual reactivity condition during the operating cycle may be an indication that the calculational model is not adequate for the operating cycle or that an unexpected change in core conditions has occurred.

The normalization of the predicted reactivity parameters to the actual reactivity value is typically performed after reaching RTP following startup from a refueling outage, with the RCS temperature, CONTROL RODS, and APSRs in their reference positions and fission product poisons at their expected equilibrium concentrations. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated, as core conditions change during the cycle.

Reactivity balance satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled, once the core design is fixed. During operation, therefore, the conditions of the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the accident analyses are no longer valid, or that the uncertainties in the nuclear design methodology are larger than expected. A limit on the reactivity of ± 1% $\Delta k/k$ has been established, based on engineering judgment. A ± 1% $\Delta k/k$ deviation in the predicted reactivity from the actual reactivity condition of the reactor is larger than expected for normal operation and should therefore be evaluated.

When the predicted core reactivity is within 1% $\Delta k/k$ of the actual reactivity value at steady state thermal conditions, the core is considered to be operating within acceptable design limits.

APPLICABILITY

In MODES 1 and 2, the limits on the core reactivity balance must be maintained to ensure an acceptable SDM and continued adherence to the assumptions used in the accident analyses. As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed.

This Specification does not apply in MODES 3, 4, and 5, because the reactor is shutdown and the net reactivity condition of the reactor can not be easily determined and changes to core reactivity due to fuel depletion cannot occur.

In MODE 6, boron concentration requirements (LCO 3.9.1, "Refueling Boron Concentration") ensure that fuel movements are performed within acceptable bounds.

ACTIONS

A.1 and A.2

Should an anomaly develop between the actual core reactivity and the predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with the input assumptions used in the core design calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 7 days is based on the low probability of an abnormality or

ACTIONS (continued)

A.1 and A.2 (continued)

accident occurring during this period, and allows sufficient time to assess the physical condition of the core and complete the evaluation of the core design and safety analysis.

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core reference conditions at the time of the reactivity balance, then a recalculation of the reactivity balance may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the appropriate reactivity parameter may be renormalized, and operation in MODE 1 may continue. If operational restrictions or additional surveillance requirements are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

The required Completion Time of 7 days is adequate for preparing operating restrictions or surveillances that may be required to allow continued reactor operation.

B.1

If the core reactivity balance cannot be restored to within the $\pm$ 1% $\Delta k/k$ limit, the unit must be brought to a MODE in which the LCO does not apply. As a conservative measure, the unit must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then boration required by Required Action A.1 of LCO 3.1.1 would occur. The allowed Completion Time of 6 hours is reasonable, based on operating experience to reach the required unit conditions from RTP in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by a periodic reactivity balance calculation that compares the predicted core reactivity to the actual core reactivity condition (net reactivity of zero condition). The comparison is made considering that core conditions are fixed or stable, including CONTROL ROD and APSR positions, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed once prior to entering MODE 1 after each fuel loading as an initial check on core reactivity conditions and design calculations at BOC. A Note is included in the SR to indicate that the normalization of predicted

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.2.1 (continued)

core reactivity to the measured value may take place within the first 60 effective full power days (EFPD) after each fuel loading. The required Frequency of 31 EFPD, following the initial 60 EFPD after entering MODE 1 is acceptable, based on the slow rate of core reactivity changes due to fuel depletion and the presence of other indicators (QPT, etc.) for prompt indication of an anomaly. The 60 EFPD after entering MODE 1 allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations. Another Note is included in the SRs to indicate that the performance of the Surveillance is not required for entry into MODE 2.

REFERENCES

1. SAR, Section 1.4, GDC 26, GDC 28, and GDC 29.

2. SAR, Chapter 3A and 14.

3. 10 CFR 50.36

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to GDC 11 (Ref. 1), the reactor core and associated Reactor Coolant System (RCS) shall be designed so that in the power operating range the net affect of the prompt inherent nuclear feedback characteristic tends to compensate for a rapid increase in reactivity.

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). Therefore, with a negative MTC a coolant temperature increase will cause a reactivity decrease. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

Both initial and reload cores are designed so that the beginning of cycle (BOC) MTC is less than or equal to zero when THERMAL POWER is 95% RTP or greater. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional burnable absorbers to yield an MTC at BOC within the range analyzed in the plant accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles are evaluated to ensure the MTC does not become more negative than the value assumed in the safety analyses.

APPLICABLE SAFETY ANALYSES

Reference 2 contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are initial conditions in the safety analyses, and both values must be bounded. Values used in the analyses consider worst case conditions, such as very large soluble boron concentrations for overheating events, to ensure the accident results are bounding.

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis; and

b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

APPLICABLE SAFETY ANALYSES (continued)

Accidents that cause core overheating (either decreased heat removal or increased power production) must be evaluated for results when the MTC is positive. Reactivity accidents that cause increased power production include the CONTROL ROD withdrawal transient from either zero or full THERMAL POWER. The limiting overheating event relative to plant response is based on the maximum difference between core power and steam generator heat removal during a transient. The most limiting event with respect to positive MTC is the startup accident.

Accidents that cause core overcooling must be evaluated for results when the MTC is most negative. The event that produces the most rapid cooldown of the RCS, and is therefore the most limiting event with respect to the negative MTC, is a steam line break (SLB) event. Following the reactor trip for the postulated EOC SLB event, the large moderator temperature reduction, combined with the large negative MTC, may produce reactivity increases that are as much as the shutdown reactivity. When this occurs, a substantial fraction of core power may be produced with all CONTROL ROD assemblies inserted, except the most reactive one. Even if the reactivity increase produces slightly subcritical conditions, a large fraction of core power may be produced through the effects of subcritical neutron multiplication.

MTC values are bounded in reload safety evaluations, assuming steady state conditions at BOC and EOC.

In MODES 1 and 2 while critical, MTC satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3). In MODE 2 while subcritical, MTC satisfies Criterion 4 of 10 CFR 50.36.

LCO

LCO 3.1.3 requires the MTC to be within specified limits to ensure the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation. The LCO establishes a maximum positive value that can not be exceeded. The limit of +0.9E-4 $\Delta k/k/^{\circ}F$ (corrected to 95% RTP) on positive MTC, when THERMAL POWER is < 95% RTP, ensures that core overheating accidents will not violate the accident analysis assumptions. The requirement for a non-positive MTC, when THERMAL POWER is ≥ 95% RTP, ensures that core operation will be stable.

MTC is a core physics parameter determined by the fuel and fuel cycle design and cannot be controlled directly once the core design is fixed during operation, therefore, the LCO can only be ensured through measurement. The surveillance check at BOC on MTC provides confirmation that the MTC is behaving as anticipated, so that the acceptance criteria are met.

APPLICABILITY

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from power operation will not violate the design assumptions of the accident analysis. In MODE 2, the limits must also be maintained to ensure that startup and subcritical accidents, such as the uncontrolled CONTROL ROD or group withdrawal, will not violate the assumptions of the accident analysis. In MODES 3, 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents (DBAs) using the MTC as an analysis assumption are initiated from these MODES. However, the variation of MTC with temperature in MODES 3, 4, and 5 for DBAs initiated in MODES 1 and 2 is accounted for in the subject accident analysis. The variation of MTC with temperature assumed in the safety analysis, is accepted as valid once the BOC measurement is used for normalization.

ACTIONS

A.1

MTC is a core physics parameter determined by the fuel and fuel cycle designs, and cannot be controlled directly once the designs have been implemented in the core. If MTC exceeds its limits, the reactor must be placed in MODE 3. This eliminates the potential for violation of the accident analysis assumptions. The associated Completion Time of 6 hours is reasonable, considering the probability of an accident occurring during the time period that would require an MTC value within the LCO limits, for reaching MODE 3 conditions from RTP in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

The SR for measurement of the MTC at the beginning of each fuel cycle provides for confirmation of the limiting MTC values. The MTC changes slowly from most positive (least negative) to most negative value during fuel cycle operation, as the RCS boron concentration is reduced with fuel depletion.

The requirement for measurement, prior to initial operation in MODE 1, satisfies the confirmatory check on the most positive (least negative) MTC value. MTC values are extrapolated and compensated to permit direct comparison to the specified MTC limits.

REFERENCES

1. SAR, Section 1.4, GDC 11.

2. SAR, Chapter 3A and 14.

3. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 CONTROL ROD Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY of the CONTROL RODS is an initial condition assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial condition assumption in the safety analysis that directly affects core power distributions and assumptions of SDM.

The applicable criteria for these design requirements are GDC 10, "Reactor Design," and GDC 26, "Reactivity Control System Redundancy-and Capability" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

Mechanical or electrical failures may cause a CONTROL ROD to become inoperable or to become misaligned from its group. CONTROL ROD inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available CONTROL ROD worth for reactor shutdown. Therefore, CONTROL ROD alignment and OPERABILITY are related to core operation within design power peaking limits and the core design requirement of a minimum SDM.

Limits on CONTROL ROD alignment and OPERABILITY have been established, and all CONTROL ROD positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

CONTROL RODS are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its rod 3/4 inch for one revolution of the leadscrew, but at varying rates depending on the signal output from the Control Rod Drive Control System (CRDCS).

The CONTROL RODS are arranged into rod groups that are radially symmetric. Therefore, movement of the CONTROL RODS does not introduce radial asymmetries in the core power distribution. The CONTROL RODS provide required negative reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating rods provide reactivity control during normal operation and transients, and their movement is normally controlled in automatic by a rod control system.

The axial position of the CONTROL RODS is indicated by three independent systems, which are the relative position indicators, the absolute position indicators, and the zone reference indicators (see LCO 3.1.7, "Position Indicator Channels").

BACKGROUND (continued)

The relative position indicator transducer is a potentiometer that is driven by electrical pulses from the CRDCS. There is one counter for each CONTROL ROD drive. Individual rods in a group, when aligned to the same power supply, all receive the same signal to move; therefore, the counters for all rods in a group should normally indicate the same position. The Relative Position Indicator System is considered highly precise. However, if a rod does not move for each demand pulse, the counter will still count the pulse and incorrectly reflect the position of the rod.

The Absolute Position Indicator System provides a highly accurate indication of actual CONTROL ROD position, but at a lower precision than the relative position indicators. This system is based on the signals from a series of reed switches spaced along a tube.

Other reed switches included in the same tube with the absolute position indicator matrix provide full in and full out limit indications and position indications at 0%, 25%, 50%, 75%, and 100% travel. This series of seven indicators are called zone reference indicators.

APPLICABLE SAFETY ANALYSES

CONTROL ROD misalignment and inoperability accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing CONTROL ROD inoperability or misalignment are that:

a. There shall be no violations of:

   1. specified acceptable fuel design limits, or

   2. Reactor Coolant System (RCS) pressure boundary integrity; and

b. The core must remain subcritical after an abnormality or accident.

Two types of misalignment are distinguished during MODES 1 and 2. During movement of a CONTROL ROD group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs when one CONTROL ROD drops partially or fully into the reactor core. This event causes an initial power reduction followed by a return towards the original power due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in excessive local linear heat rates (LHRs).

The accident analysis and reload safety evaluations define regulating rod insertion limits that ensure the required SDM can always be achieved if the maximum worth CONTROL ROD is stuck fully withdrawn (Ref. 3). If a CONTROL ROD is stuck in or dropped in, continued operation is permitted if the increase in local LHR is within

APPLICABLE SAFETY ANALYSES (continued)

the design limits. The Required Action statements in the LCOs provide conservative reductions in THERMAL POWER and verification of SDM to ensure continued operation remains within the bounds of the safety analysis (Ref. 3).

Continued operation of the reactor with a misaligned or dropped CONTROL ROD is allowed if the local core LHRs are verified to be within their limits in the COLR. When a CONTROL ROD is misaligned, the assumptions that are used to determine the regulating rod insertion limits, APSR insertion limits, AXIAL POWER IMBALANCE limits, and QPT limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and local core LHRs must be verified directly by incore mapping. Bases Section 3.2, "Power Distribution Limits," contains a more complete discussion of the relation of LHR to the operating limits.

In MODES 1 and 2 while critical, the CONTROL ROD group alignment limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 4). In MODE 2 while subcritical, the CONTROL ROD group alignment limits satisfy Criterion 4 of 10 CFR 50.36.

LCO

The limits on CONTROL ROD group alignment, safety rod insertion, and APSR alignment, together with the limits on regulating rod insertion, APSR insertion, AXIAL POWER IMBALANCE, and QPT, ensure the reactor will operate within the fuel design criteria. The Required Actions in these LCOs ensure that deviations from the alignment limits will either be corrected or that THERMAL POWER will be adjusted, so that excessive local LHRs will not occur and the requirements on SDM and ejected rod worth are preserved.

The limit for individual CONTROL ROD misalignment is 6.5% (approximately 9 inches) deviation from the group average position. This value is established, based on the distance between reed switches, with additional allowances for uncertainty in the absolute position indicator amplifiers, group average position calculator, and asymmetric alarm or fault detector outputs. Therefore, no additional uncertainties are required to be incorporated in the implementing procedures.

For the purpose of complying with this LCO, the position of a misaligned rod is not included in the calculation of the rod group average position. A CONTROL ROD is not considered to be inoperable due solely to misalignment. A CONTROL ROD is considered to be inoperable if it is not free to insert into the core within the required insertion time, or as directed by LCO 3.1.7, "Position Indicator Channels."

Failure to meet the requirements of this LCO may produce unacceptable LHRs, or unacceptable SDM or ejected rod worth, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on CONTROL ROD OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which significant neutron (or fission) power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down and resultant local power peaking would not exceed fuel design limits. In MODES 3, 4, 5, and 6, the OPERABILITY of the CONTROL RODS has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5, and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during MODE 6.

ACTIONS

A.1.1

Compliance with Required Actions of Condition A allows for continued power operation with one CONTROL ROD inoperable, or misaligned from its group average position, or both. Since the rod may be inserted farther than the group average insertion for a long time, SDM must be evaluated. Ensuring the SDM meets the minimum requirement established in the COLR within 1 hour is adequate to determine that further degradation of the SDM is not occurring.

A.1.2

If the SDM is less than the limit specified in the COLR, then the restoration of the required SDM requires increasing the RCS boron concentration, since the CONTROL ROD may remain misaligned and not be providing its normal negative reactivity on tripping. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour to initiate boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.

A.2.1

Alignment of the inoperable or misaligned CONTROL ROD may be accomplished by either moving the single CONTROL ROD to the group average position, or by moving the remainder of the group to the position of the single inoperable or misaligned CONTROL ROD. Either action can be used to restore the CONTROL RODS to a radially symmetric pattern. However, this must be done without violating the CONTROL ROD group sequence, overlap, and insertion limits of LCO 3.2.1, "Regulating Rod Insertion Limits," given in the COLR. THERMAL POWER must

ACTIONS (continued)

A.2.1 (continued)

also be restricted, as necessary, to the value allowed by the insertion limits of LCO 3.2.1. The required Completion Time of 2 hours is acceptable because local xenon redistribution during this short interval will not cause a significant increase in LHR. This option of inserting the group to the position of the misaligned rod is not available if a safety rod is misaligned, since the limits of LCO 3.1.5, "Safety Rod Insertion Limits," would be violated. If realignment of the CONTROL ROD to the group average or alignment of the group to the misaligned CONTROL ROD is not completed within 1 hour, the rod shall be considered inoperable.

A.2.2.1

Reduction of THERMAL POWER to ≤ 60% ALLOWABLE THERMAL POWER ensures that local LHR increases, due to a misaligned rod, will not cause the core design criteria to be exceeded. The required Completion Time of 2 hours allows the operator sufficient time for reducing THERMAL POWER.

A.2.2.2

The existing CONTROL ROD configuration must not cause an ejected rod to exceed the limit of 0.65% $\Delta k/k$ at RTP or 1.00% $\Delta k/k$ at zero power (Ref. 3). This evaluation may require a computer calculation of the maximum ejected rod worth based on nonstandard configurations of the CONTROL ROD groups. The evaluation must determine the ejected rod worth for the duration of time that operation is expected to continue with a misaligned rod. Should fuel cycle conditions at some later time become more bounding than those at the time of the rod misalignment, additional evaluation will be required to verify the continued acceptability of operation. The required Completion Time of 72 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and sufficient time is provided to perform the required evaluation.

A.2.2.3

Performance of SR 3.2.5.1 provides a determination of the local core LHRs using the Incore Detector System. Verification of the local core LHRs from an incore power distribution map is necessary to ensure that excessive local LHRs will not occur due to CONTROL ROD misalignment. This is necessary because the assumption that all CONTROL RODS are aligned (used to determine the regulating rod insertion, AXIAL POWER IMBALANCE, and QPT limits) is not valid when the CONTROL RODS are not aligned. The required Completion Time of 72 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and adequate time is allowed to obtain an incore power distribution map.

ACTIONS (continued)

A.2.2.3 (continued)

Required Action A.2.2.3 is modified by a Note that requires the performance of SR 3.2.5.1 only when THERMAL POWER is greater than 20% RTP. This establishes a Required Action that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

B.1

If the Required Actions and associated Completion Times for Condition A are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems.

C.1.1

More than one CONTROL ROD becoming inoperable or misaligned from their group average position, or both, is not expected and may violate the minimum SDM requirement. Therefore, SDM must be evaluated. Ensuring the SDM meets the minimum requirement within 1 hour allows the operator adequate time to determine the SDM.

C.1.2

If the SDM is less than the limit specified in the COLR, then the restoration of the required SDM requires increasing the RCS boron concentration to provide negative reactivity. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.

C.2

If more than one CONTROL ROD is inoperable or misaligned from their group average position, continued operation of the reactor may cause the misalignment to increase, as the regulating rods insert or withdraw to control reactivity. If the CONTROL ROD misalignment increases, local power peaking may also increase, and local LHRs will also increase if the reactor continues operation at THERMAL POWER. The SDM is decreased when one or more CONTROL RODS become inoperable at a given THERMAL POWER level, or if one or more CONTROL RODS become misaligned by insertion from the group average position.

ACTIONS (continued)

C.2 (continued)

Therefore, it is prudent to place the reactor in MODE 3. LCO 3.1.4 does not apply in MODE 3 since excessive power peaking cannot occur. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual CONTROL RODS are aligned within 6.5% of their group average height limits at a 12 hour Frequency allows the operator to detect a rod that is beginning to deviate from its expected position. The specified Frequency takes into account other CONTROL ROD position information that is continuously available to the operator in the control room, so that during actual CONTROL ROD motion, deviations can immediately be detected.

SR 3.1.4.2

Verifying each CONTROL ROD is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each CONTROL ROD could result in radial tilts. Exercising each individual CONTROL ROD every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each CONTROL ROD by approximately 1.5% (approximately 2 inches) will not cause radial or axial power tilts, or oscillations, to occur. No additional allowances for instrument uncertainty are required to be incorporated in the implementing procedures for this parameter. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between typical performances of SR 3.1.4.2 (determination of CONTROL ROD OPERABILITY by movement), if a CONTROL ROD(S) is discovered to be immovable, but is otherwise determined to be capable of being fully inserted, the CONTROL ROD(S) may continue to be considered OPERABLE unless inoperable for some other reason. At any time, if a CONTROL ROD(S) is immovable, a determination of the capability to fully insert (OPERABILITY) the CONTROL ROD(S) must be made, and appropriate action taken.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.3

Verification of CONTROL ROD drop time allows the operator to determine that the maximum CONTROL ROD drop time permitted is consistent with the assumed CONTROL ROD drop time used in the safety analysis. The CONTROL ROD drop time given in the safety analysis is 1.66 seconds to 3/4 position insertion (Ref. 5). This 1.66 seconds includes 0.14 seconds delay time for opening of the CRD breakers and for CRDM unlatch. Using the CONTROL ROD position versus time and time versus reactivity insertion curves gives a value of 1.4 seconds to 2/3 reactivity insertion upon which the accident analysis is based (Ref. 3). The former value is used in the Surveillance because the zone reference lights are located at 25% insertion intervals. The zone reference lights will activate at 3/4 insertion to give an indication of the CONTROL ROD drop time and CONTROL ROD location. The CONTROL ROD drop time is the total elapsed time from the loss of power to the control rod drive (CRD) breaker under voltage coils until the CONTROL ROD has completed approximately 104 inches of travel from the fully withdrawn position. The safety analysis has included a CRD breaker time delay of 0.080 seconds in SAR Chapter 14 (Ref. 3). If the trip test measurement is begun with the opening of the CRD breakers, the required trip insertion time shall be reduced to 1.58 seconds and the CRD breaker time delay shall be verified to be less than or equal to 0.080 seconds.

Measuring CONTROL ROD drop times, prior to reactor criticality after reactor vessel head removal, ensures that the reactor internals and CRDM will not interfere with CONTROL ROD motion or CONTROL ROD drop time. This Surveillance is performed during a unit outage, due to the unit conditions needed to perform the SR and the potential for an unplanned unit transient if the Surveillance were performed with the reactor at power.

This testing is normally performed with all reactor coolant pumps operating and average moderator temperature ≥ 525°F to simulate a reactor trip under actual conditions. However, if the CONTROL ROD drop times are determined with less than four reactor coolant pumps operating, a Note allows operation to continue, provided operation is restricted to the pump combination utilized during the CONTROL ROD drop time determination or pump combinations providing less total reactor coolant flow.

REFERENCES

1. SAR, Section 1.4, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. SAR, Chapter 3A and 14.
4. 10 CFR 50.36.
5. SAR, Chapter 3.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Safety Rod Insertion Limit

BASES

BACKGROUND

The insertion limits of the CONTROL RODS are initial condition assumptions in all safety analyses that assume CONTROL ROD insertion upon reactor trip. The insertion limits directly affect core power distributions and assumptions of available SDM, ejected rod worth, and initial reactivity insertion rate.

The applicable criteria for the reactivity and power distribution design requirements are GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Limits on safety rod insertion have been established, and all CONTROL ROD positions are monitored and controlled during operation in MODES 1 and 2 to ensure that the reactivity limits, ejected rod worth, and SDM limits are preserved.

The regulating groups are used for precise reactivity control of the reactor. The positions of the regulating groups are normally automatically controlled by the automatic control system, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). In MODES 1 and 2, the regulating groups must be maintained above designated insertion limits and are typically near the fully withdrawn position during normal operations. Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature and fuel burnup.

The safety groups can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The safety groups are controlled manually by the control room operator. Prior to entry into MODE 2 from MODE 3, the safety groups must be fully withdrawn. The safety groups must be completely withdrawn from the core prior to withdrawing any regulating groups during an approach to criticality. The safety groups remain in the fully withdrawn position until the reactor is shut down. They add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

APPLICABLE SAFETY ANALYSES

On a reactor trip, all CONTROL RODS, except the most reactive rod, are assumed to insert into the core. The safety groups shall be at their fully withdrawn limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The regulating groups may be partially inserted in the core as allowed by LCO 3.2.1, "Regulating Rod Insertion Limits." The safety group and regulating rod group insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") following a reactor trip from RTP. The combination of regulating groups and safety groups (less the most reactive rod, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power and to achieve the required SDM at rated no load temperature (Ref. 3).

The acceptance criteria for addressing safety and regulating rod group insertion limits and inoperability or misalignment are that:

a. There shall be no violations of:

   1. specified acceptable fuel design limits, or
   2. RCS pressure boundary integrity; and

b. The core must remain subcritical after an abnormality. Although the SAR does not state this as an acceptance criteria for the main steam line break event, B & W has placed a design objective on this event that the core remains subcritical throughout the event (Ref. 4).

In MODES 1 and 2 while critical, the safety rod insertion limits satisfy Criteria 2 and 3 of 10 CFR 50.36 (Ref. 5). In MODE 2 while subcritical, the safety rod insertion limits satisfy Criterion 4 of 10 CFR 50.36.

LCO

The safety groups must be fully withdrawn any time the reactor is in MODE 1 or 2. This LCO in combination with LCO 3.2.1 ensures that a sufficient amount of negative reactivity is available to shut down the reactor and achieve the required SDM following a reactor trip.

This LCO has been modified by a Note indicating the LCO requirement is suspended for those safety rods which are inserted solely due to testing in accordance with SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the safety group to move below the LCO limits, which would normally violate the LCO.

APPLICABILITY

The safety groups must be within their insertion limits with the reactor in MODES 1 and 2. This LCO in combination with LCO 3.2.1 ensures that a sufficient amount of negative reactivity is available to shut down the reactor and achieve the required SDM following a reactor trip. Refer to LCO 3.1.1 for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

ACTIONS

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

The safety rod must be declared inoperable within a 1 hour time frame. This requires entry into LCO 3.1.4, "CONTROL ROD Group Alignment Limits." In addition, since the safety rod may be inserted farther than the group average insertion for a long time, SDM must be evaluated. Ensuring the SDM meets the minimum requirement within 1 hour is adequate to determine that further degradation of the SDM is not occurring.

Restoration of the required SDM, if necessary, requires increasing the boron concentration, since the safety rod may remain misaligned and not be providing its normal negative reactivity on tripping. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.

The allowed Completion Time of 1 hour provides an acceptable time for evaluating and repairing minor problems without allowing the unit to remain in an unacceptable condition for an extended period of time.

B.1.1 and B.1.2

When more than one safety rod is not fully withdrawn, there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration to restore SDM.

In this situation, SDM verification must include the worth of any rod not capable of being fully inserted as well as the CONTROL ROD of maximum worth.

ACTIONS (continued)

B.2

If more than one safety rod is not fully withdrawn, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from RTP in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that each safety rod is fully withdrawn ensures the safety rods are available to provide reactor shutdown capability.

Verification that individual safety rod positions are fully withdrawn at a 12 hour Frequency allows the operator to detect a safety rod beginning to deviate from its expected position. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of the safety rods.

REFERENCES

1. SAR, Section 1.4, GDC 10, GDC 26, and GDC 28.

2. 10 CFR 50.46.

3. SAR, Chapters 3 and 4.

4. BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," Rev. 2.

5. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 AXIAL POWER SHAPING ROD (APSR) Alignment Limits

BASES

BACKGROUND

The OPERABILITY of the APSRs and APSR alignment are initial condition assumptions in the safety analysis that directly affect core power distributions. The applicable criteria for these power distribution design requirements are GDC 10, "Reactor Design," and GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Mechanical or electrical failures may cause an APSR to become inoperable or to become misaligned from its group. APSR inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution. Therefore, APSR alignment and OPERABILITY are related to core operation within design power peaking limits.

Limits on APSR alignment and OPERABILITY have been established, and all APSR and CONTROL ROD positions are monitored and controlled during power operation to ensure that the power distribution limits defined by the design peaking limits are preserved.

APSRs are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its rod 3/4 inch for one revolution of the leadscrew, but at varying rates depending on the signal output from the Control Rod Drive Control System (CRDCS).

The APSRs are arranged into groups that are radially symmetric. Therefore, movement of the APSRs does not introduce radial asymmetries in the core power distribution. The APSRs, which are used to assist in control of the axial power distribution, are positioned manually and do not trip.

LCO 3.1.6 is conservatively based on use of black (Ag-In-Cd) APSRs and bounds use of gray (Inconel) APSRs. The reactivity worth of black APSRs is greater than that of gray APSRs; thus the impact of black APSR misalignment on the core power distribution is greater.

APPLICABLE SAFETY ANALYSES

There are no explicit safety analyses associated with misaligned APSRs. However, alignment of the APSRs is required to prevent inducing a QUADRANT POWER TILT. The LCOs governing APSR alignment are provided because the power distribution analysis supporting LCO 3.2.1, LCO 3.2.3 and LCO 3.2.4 assumes the APSRs are aligned.

During movement of an APSR group, one rod may stop moving while the other rods in the group continue. This condition may cause excessive power peaking. Continued operation of the reactor with a misaligned APSR is allowed if Section 3.2, "Power Distribution Limits," are preserved.

Because ANO-1 uses gray APSRs, the APSR alignment limits satisfy Criterion 4 of 10 CFR 50.36 (Ref. 3).

LCO

The limits on CONTROL ROD group alignment, safety rod withdrawal, and APSR alignment, together with the limits on regulating rod insertion, APSR insertion, AXIAL POWER IMBALANCE, and QPT, ensure the reactor will operate within the fuel design criteria. The Required Action in this LCO ensures deviations from the alignment limits will be adjusted so that excessive local LHRs will not occur.

The limit for individual APSR misalignment is 6.5% (approximately 9 inches) deviation from the group average position. This value is established based on the distance between reed switches, with additional allowances for uncertainty in the absolute position indicator amplifiers, group average position calculator, and asymmetric alarm or fault detector outputs. Therefore, no additional uncertainties are required to be incorporated in the implementing procedures. The position of an inoperable APSR is not included in the calculation of the APSR group's average position.

Failure to meet the requirements of this LCO may produce unacceptable LHRs, which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on APSR OPERABILITY and alignment are applicable in MODES 1 and 2, because these are the only MODES in which significant neutron (or fission) power is generated, and the OPERABILITY and alignment of APSRs have the potential to affect the safety of the unit. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down, and excessive local LHRs cannot occur from APSR misalignment.

ACTIONS

A.1

The ACTIONS described below are required if one APSR is inoperable. The unit is not allowed to operate with more than one inoperable APSR. This would require the reactor to be placed in MODE 3, in accordance with LCO 3.0.3.

An alternate to realigning a single misaligned APSR to the group average position is to align the remainder of the APSR group to the position of the misaligned or inoperable APSR, while maintaining APSR insertion, in accordance with the limits in the COLR. This restores the alignment requirements. Deviations up to 2 hours will not cause significant xenon redistribution to occur. This alternative assumes the APSR group movement does not cause the limits of LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," to be exceeded. For this reason, APSR group movement is only practical for instances where small movements of the APSR group are sufficient to re-establish APSR alignment.

The reactor may continue in operation with the APSR misaligned if the limits on power peaking are surveilled within 2 hours to determine if power peaking is still within limits. Also, since any additional movement of the APSRs may result in additional imbalance, Required Action A.1 also requires the power peaking surveillance to be performed again within 2 hours after each APSR movement.

B.1

The unit must be brought to a MODE in which the LCO does not apply if the Required Actions and associated Completion Times cannot be met. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems. In MODE 3, APSR group alignment limits are not required because the reactor is not generating significant THERMAL POWER and excessive local LHRs cannot occur from APSR misalignment.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Verification at a 12 hour Frequency that individual APSR positions are within 6.5% of the group average height limits allows the operator to detect an APSR beginning to deviate from its expected position. In addition, APSR position is continuously available to the operator in the control room so that during actual APSR motion, deviations can immediately be detected.

REFERENCES

1. SAR, Section 1.4, GDC 10 and GDC 28.
2. 10 CFR 50.46.
3. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Position Indicator Channels

BASES

BACKGROUND

According to the SAR discussion of GDC 13 (Ref. 1), adequate instrumentation and controls are provided to maintain operating variables within prescribed ranges for normal operation and monitor accident conditions as appropriate to assure adequate safety. LCO 3.1.7 is required to ensure OPERABILITY of the CONTROL ROD and APSR position indicators, and thereby ensure compliance with the CONTROL ROD and APSR alignment and insertion limits.

The OPERABILITY, including position indication, of the CONTROL RODS is an initial condition assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment for the CONTROL RODS and APSRs is assumed in the safety analysis, which directly affect core power distributions and assumptions of available SDM.

Mechanical or electrical failures may cause a CONTROL ROD or APSR to become misaligned from its group. CONTROL ROD or APSR misalignment may cause increased local linear heat rates (LHRs), due to the asymmetric reactivity distribution, and a reduction in the total available CONTROL ROD worth for reactor shutdown. Therefore, CONTROL ROD and APSR alignment are related to core operation within design LHR limits and the core design requirement of a minimum SDM. CONTROL ROD and APSR position indication is needed to assess OPERABILITY and alignment.

Limits on CONTROL ROD and APSR alignment, and CONTROL ROD and APSR group position have been established, and all CONTROL ROD and APSR positions are monitored and controlled during operation to ensure that the power distribution and reactivity limits defined by the design LHR and SDM limits are preserved.

Three methods of CONTROL ROD and APSR position indication are provided in the Control Rod Drive Control System. The three means are by absolute position indicator, relative position indicator transducers, and zone reference indicators. The absolute position indicator transducer consists of a series of magnetically operated reed switches mounted in a tube parallel to the control rod drive mechanism (CRDM) motor tube extension. Switch contacts close when a permanent magnet mounted on the upper end of the CONTROL ROD or APSR assembly leadscrew extension comes near. As the leadscrew and CONTROL ROD or APSR move, the switches operate sequentially, producing an analog voltage proportional to position. Other reed switches included in the same tube with the absolute position indicator matrix provide full in and full out limit indications, and position indications at 0%, 25%, 50%, 75%, and 100% travel. This series of seven

BACKGROUND (continued)

indicators are called zone reference indicators. The relative position indicator transducer is a potentiometer, driven by a step motor that produces a signal proportional to CONTROL ROD or APSR position, based on the electrical pulse steps that drive the CRDM.

CONTROL ROD and APSR position indicating readout devices located in the control room consist of single rod position meters on a position indication panel and group average position meters. A selector switch permits either relative or absolute position indication to be displayed on all of the individual position indication meters. Indicator lights are provided on the individual position indication panel to indicate when each CONTROL ROD or APSR is fully withdrawn, fully inserted, enabled, or transferred, and whether a rod position asymmetry alarm condition is present. Additional indicators show full insertion, full withdrawal, and enabled for motion for each CONTROL ROD and APSR group. The consequence of continued operation with an inoperable absolute position indicator or relative position indicator channel is a decreased reliability in determining CONTROL ROD and APSR position. Therefore, the potential for operation in violation of design LHR or SDM limits is increased.

APPLICABLE SAFETY ANALYSES

CONTROL ROD and APSR position accuracy is essential during power operation. LHR, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2) with CONTROL RODS or APSRs operating outside their limits undetected. CONTROL ROD and APSR positions must be known in order to verify the core is operating within the group sequence, overlap, design LHRs, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Safety Rod Insertion Limits"; LCO 3.2.1, "Regulating Rod Insertion Limits"; and LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits"). The CONTROL ROD and APSR positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "CONTROL ROD Group Alignment Limits," and LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits"). CONTROL ROD and APSR positions are continuously monitored to provide operators with information that ensures the unit is operating within the bounds of the accident analysis assumptions.

In MODES 1 and 2 while critical, the CONTROL ROD and APSR position indicator channels satisfy Criterion 2 of 10 CFR 50.36 (Ref. 3). In MODE 2 while subcritical, the CONTROL ROD and APSR position indicator channels satisfy Criterion 4 of 10 CFR 50.36.

LCO

LCO 3.1.7 specifies that one position indicator channel be OPERABLE for each CONTROL ROD and APSR.

This requirement ensures that CONTROL ROD and APSR position indication during MODES 1 and 2 and PHYSICS TESTS is accurate, and that design assumptions are not challenged. OPERABILITY of the position indicator channel ensures that inoperable, misaligned, or mispositioned CONTROL RODS or APSRs can be detected. Therefore, LHR and SDM can be controlled within acceptable limits.

APPLICABILITY

In MODES 1 and 2, OPERABILITY of the position indicator channel is required, since the reactor is, or is capable of, generating THERMAL POWER in these MODES. In MODES 3, 4, 5, and 6, Applicability is not required because the reactor is shut down with the required minimum SDM and is not generating significant THERMAL POWER.

ACTIONS

A.1

If the required position indicator channel is inoperable for one or more rods, the position of the CONTROL ROD or APSR is not known with certainty. Therefore, each affected CONTROL ROD or APSR must be declared inoperable, and the limits of LCO 3.1.4 or LCO 3.1.6 apply. The required Completion Time for declaring the rod(s) inoperable is immediately. Therefore LCO 3.1.4 or LCO 3.1.6 is entered immediately, and the required Completion Times for the appropriate Required Actions in those LCOs apply without delay.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

A CHANNEL CHECK of the required position indication channel ensures that position indication for each CONTROL ROD and APSR remains OPERABLE and accurate. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. However, this CHANNEL CHECK will be used to detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.1 (continued)

When compared to other channels, the agreement criteria between channels is determined by the unit staff. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

The CHANNEL CHECK supplements less formal but more frequent checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required position indicator channel.

The required Frequency of 12 hours is adequate for verifying that no degradation in system OPERABILITY has occurred.

SR 3.1.7.2

A CHANNEL CALIBRATION of the required position indication channel verifies that the channel responds within the necessary range and accuracy.

The Frequency of 18 months is based on operating experience and consistency with the typical industry refueling cycle.

REFERENCES

1. SAR, Section 1.4, GDC 13.

2. SAR, Chapter 14.

3. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 PHYSICS TESTS Exceptions Systems - MODE 1

BASES

BACKGROUND

The purpose of this LCO is to permit PHYSICS TESTS to be conducted by providing exemptions from the requirements of other LCOs. Establishment of a test program to verify that structures, systems, and components will perform satisfactorily in service is required by Section XI of 10 CFR 50, Appendix B (Ref. 1). Testing is required as an integral part of the design, fabrication, construction, and operation of the unit. All functions necessary to ensure that specified design conditions are not violated during normal operation and abnormalities must be tested. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to:

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality; during startup, low power operations, and power ascension; at high powers; and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions, and that the core can be operated as designed (Ref. 3).

The inclusion of this PHYSICS TESTS Exception LCO is acceptable based on the use of approved written procedures, administrative controls, the requirements of 10 CFR 50.59, and the LCO 3.1.8 provisions in effect during the conduct of PHYSICS TESTS. PHYSICS TESTS procedures are written and approved in accordance with established guidelines. The procedures include all information necessary to permit a detailed execution of testing required to ensure the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long term power operation. Examples of PHYSICS TESTS include determination of critical boron concentration, CONTROL ROD group worths, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because reactor protection criteria are preserved by the LCOs still in effect and by the SRs. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on linear heat rate (LHR), ejected rod worth, and shutdown capability are maintained during the PHYSICS TESTS.

Reference 4 describes the initial testing of the facility, including PHYSICS TESTS. Table 13-2 (Ref. 5) summarizes the post-criticality tests. Requirements for reload fuel cycle PHYSICS TESTS are given in SAR Section 3A.9 (Ref. 3). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, one or more LCOs must sometimes be suspended to make completion of PHYSICS TESTS possible or practical.

This is acceptable as long as the fuel design criteria are not violated. When one or more of the limits specified in:

LCO 3.1.4, "CONTROL ROD Group Alignment Limits";
LCO 3.1.5, "Safety Rod Insertion Limits";
LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits";
LCO 3.2.1, "Regulating Rod Insertion Limits";
LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," for the restricted operation region only;
LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits"; and
LCO 3.2.4, "QUADRANT POWER TILT (QPT)"

are suspended for PHYSICS TESTS, the fuel design criteria are preserved by maintaining the LHR (in MODE 1 PHYSICS TESTS) within limits, maintaining ejected rod worth within limits by restricting regulating rod insertion to within the acceptable operating region or the restricted operating region, by limiting maximum THERMAL POWER and by maintaining SDM within the limit provided in the COLR. Therefore, surveillance of the LHR and SDM is required to verify that their limits are not exceeded. The limits for the LHR are specified in the COLR. Refer to the Bases for LCO 3.2.5 for a complete discussion of LHR. During PHYSICS TESTS, one or more of the LCOs that normally preserve the LHR limits may be suspended. However, the results of the safety analysis are not adversely impacted if verification that core LHRs are within their limits is obtained, while one or more of the LCOs is suspended. Therefore, SRs are placed on LHR during MODE 1 PHYSICS TESTS when THERMAL POWER exceeds 20% RTP to verify that the core LHRs remain within their limits. Periodic verification of these factors allows PHYSICS TESTS to be conducted while continuing to maintain the design criteria.

PHYSICS TESTS include measurement of core nuclear parameters or exercise of control components that affect process variables. Among the process variables involved are AXIAL POWER IMBALANCE and QPT, which represent initial condition input (power peaking) for the accident analysis. Also involved are the movable control components, i.e., the regulating rods and the APSRs, which affect power peaking. The limits for these variables are specified for each fuel cycle in the COLR.

APPLICABLE SAFETY ANALYSES (continued)

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36 (Ref. 6) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion for the other LCOs is provided in their respective Bases.

LCO

This LCO permits individual CONTROL RODS and APSRs to be positioned outside of their specified group alignment and withdrawal limits and to be assigned to other than specified CONTROL ROD groups, and permits AXIAL POWER IMBALANCE and QPT limits to be exceeded during the performance of PHYSICS TESTS. In addition, this LCO permits verification of the fundamental core characteristics and nuclear instrumentation operation.

The requirements of LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.2.1 (for the restricted operation region only, LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. THERMAL POWER is maintained ≤ 85% RTP;

b. Nuclear overpower trip setpoint is ≤ 10% RTP higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP;

c. LHR is maintained within limits specified in the COLR while operating at greater than 20% RTP; and

d. SDM is verified to be within the limit provided in the COLR.

Operation with THERMAL POWER ≤ 85% RTP during PHYSICS TESTS provides an acceptable thermal margin when one or more of the applicable LCOs is out of specification. Eighty-five percent RTP is consistent with the maximum power level for conducting the intermediate core power distribution test specified in Reference 3. The nuclear overpower trip setpoint is reduced so that a similar margin exists between the steady state condition and trip setpoint as exists during normal operation at RTP.

LCO provision c is modified by a Note that requires the adherence to LHR requirements only when THERMAL POWER is greater than 20% RTP. This establishes an LCO provision that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

APPLICABILITY

This LCO is applicable in MODE 1, when the reactor has completed low power testing and is in power ascension, or during power operation with THERMAL POWER > 5% RTP but ≤ 85% RTP. This LCO is applicable for power ascension testing, as described in SAR Section 3A.9 (Ref. 3). In MODE 2, Applicability of this LCO is not required because LCO 3.1.9, "PHYSICS TESTS Exceptions - MODE 2," addresses PHYSICS TESTS exceptions initiated in MODE 2. In MODES 3, 4, 5, and 6, Applicability is not required because PHYSICS TESTS are not performed in these MODES.

ACTIONS

A.1 and A.2

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the unit conditions. Boration will be continued until SDM is within limit. In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification. A Completion Time of one hour is provided for the operator to restore compliance with the excepted LCOs.

B.1

If THERMAL POWER exceeds 85% RTP, then 1 hour is allowed for the operator to reduce THERMAL POWER to within limits or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCO, addressed by PHYSICS TESTS exceptions.

If the nuclear overpower trip setpoint is not within the specified limits, then 1 hour is allowed for the operator to restore the nuclear overpower trip setpoint within limits or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCO, addressed by these PHYSICS TESTS exceptions.

ACTIONS (continued)

B.1 (continued)

If the results of the incore flux map indicate that LHR has exceeded its limit, then PHYSICS TESTS are suspended. This action is required because of direct indication that the core LHR, which is a fundamental initial condition for the safety analysis, is excessive. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

This Condition is modified by a Note that requires performance of the Required Action only when THERMAL POWER is greater than 20% RTP. This establishes an ACTIONS entry Condition that is consistent with LCO provision c and the Applicability of LCO 3.2.5, "Power Peaking."

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification that THERMAL POWER is ≤ 85% RTP ensures that the required additional thermal margin has been established prior to and during PHYSICS TESTS. The required Frequency of once per hour allows the operator adequate time to determine any degradation of the established thermal margin during PHYSICS TESTS.

SR 3.1.8.2

Verification that core LHRs are within their limits ensures that core LHR and departure from nucleate boiling ratio will remain within their limits, while one or more of the LCOs that normally control these design limits are out of specification. The required Frequency of 2 hours allows the operator adequate time for collecting a flux map and for performing the LHR verification, based on operating experience. If SR 3.2.5.1 is not met, PHYSICS TESTS are suspended and LCO 3.2.5 applies. This Frequency is more conservative than the Completion Time for restoration of the individual LCOs that preserve the LHR limits.

This SR is modified by a Note that requires performance only when THERMAL POWER is greater than 20% RTP. This establishes a performance requirement that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

SR 3.1.8.3

Verification that the nuclear overpower trip setpoint is within the limit specified for each PHYSICS TEST ensures that core protection at the reduced power level is established during the PHYSICS TESTS. Performing the verification once within 8 hours prior to the performance of PHYSICS TESTS at each testing plateau allows the operator adequate time for verifying the established trip setpoint before initiating PHYSICS TESTS.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.8.4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. Reactor Coolant System (RCS) boron concentration;

b. CONTROL ROD position;

c. Doppler defect;

d. Fuel burnup based on gross thermal energy generation;

e. Samarium concentration;

f. Xenon concentration; and

g. Moderator defect.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50.59.
3. SAR, Section 3A.9.
4. SAR, Section 13.3, 13.4 and 13.6.
5. SAR, Section 13.4, Table 13-2.
6. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.9 PHYSICS TESTS Exceptions - MODE 2

BASES

BACKGROUND

The purpose of this MODE 2 LCO is to permit PHYSICS TESTS to be conducted by providing exemptions from the requirements of other LCOs. Establishment of a test program to verify that structures, systems, and components will perform satisfactorily in service is required by Section XI of 10 CFR 50, Appendix B (Ref. 1). Testing is required as an integral part of the design, fabrication, construction, and operation of the unit. All functions necessary to ensure that specified design conditions are not violated during normal operation and abnormalities must be tested. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

The key objectives of a test program are to:

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality; during startup, low power operations, and power ascension; at high powers; and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions, and that the core can be operated as designed (Ref. 3).

The inclusion of this PHYSICS TESTS Exception LCO is acceptable based on the use of approved written procedures, administrative controls, the requirements of 10 CFR 50.59, and the LCO 3.1.9 provisions in effect during the conduct of PHYSICS TESTS. PHYSICS TESTS procedures are written and approved in accordance with established guidelines. The procedures include all information necessary to permit a detailed execution of testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long term power operation.

BACKGROUND (continued)

Examples of MODE 2 PHYSICS TESTS include determination of critical boron concentration, CONTROL ROD group worth, and reactivity coefficients.

APPLICABLE SAFETY ANALYSES

Reference 4 describes the initial testing of the facility, including PHYSICS TESTS. Table 13-2 (Ref. 5) summarizes the post-criticality tests. Requirements for reload fuel cycle PHYSICS TESTS are given in SAR Section 3A.9 (Ref. 3). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more of the LCOs must be suspended to make completion of PHYSICS TESTS possible or practical.

It is acceptable to suspend the following LCOs for PHYSICS TESTS because reactor protection criteria are preserved by the LCOs still maintained and by the SRs:

LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
LCO 3.1.4, "CONTROL ROD Group Alignment Limits";
LCO 3.1.5, "Safety Rod Insertion Limits";
LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits";
LCO 3.2.1, "Regulating Rod Insertion Limits";
LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality."

Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on THERMAL POWER and shutdown capability are maintained during the PHYSICS TESTS.

Shutdown capability is preserved by limiting THERMAL POWER and maintaining adequate SDM, when in MODE 2 PHYSICS TESTS. In MODE 2, the Reactor Coolant System (RCS) temperature must be within the narrow range instrumentation for unit control. The narrow range temperature instrumentation goes on scale at 520°F. Therefore, it is considered safe to allow the minimum RCS temperature to decrease to 520°F during MODE 2 PHYSICS TESTS, based on the low probability of an accident occurring and on prior operating experience.

PHYSICS TESTS include measurement of core nuclear parameters or exercise of control components that affect process variables.

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36 (Ref. 6) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria for the other LCOs is provided in their respective Bases.

LCO

This LCO permits individual CONTROL RODS and APSRs to be positioned outside of their specified group alignment and withdrawal limits and to be assigned to other than specified CONTROL ROD groups during the performance of PHYSICS TESTS. In addition, this LCO permits verification of the fundamental core characteristics.

This LCO also allows suspension of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.2.1, LCO 3.2.2, and LCO 3.4.2, provided:

a. THERMAL POWER is ≤ 5% RTP;

b. Nuclear overpower trip setpoints on the OPERABLE nuclear power range channels are set to ≤ 5% RTP;

c. Nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit is OPERABLE; and

d. SDM is within the limit provided in the COLR.

The limits of LCO 3.2.3 and LCO 3.2.4 are not exempted by this specification because they do not apply in MODE 2. Inhibiting CONTROL ROD withdrawal, based on startup rate, also limits local linear heat rate (LHR), departure from nucleate boiling ratio (DNBR), and peak RCS pressure during accidents initiated from low power.

APPLICABILITY

This LCO is applicable when the reactor is either subcritical or critical with THERMAL POWER ≤ 5% RTP. The Applicability is stated as "during PHYSICS TESTS initiated in MODE 2" to ensure that the 5% RTP maximum power level is not exceeded. Should the THERMAL POWER exceed 5% RTP, and consequently the unit enter MODE 1, this Applicability statement prevents exiting this Specification and its Required Actions. This LCO is applicable for initial criticality or low power testing, as described in SAR Section 3A.9 (Ref. 3). In MODE 1, Applicability of this LCO is not required because LCO 3.1.8, "PHYSICS TESTS Exceptions," addresses PHYSICS TESTS exceptions in MODE 1. In MODES 3, 4, 5, and 6, a test exception LCO is not required because the excepted LCOs do not apply in these MODES.

ACTIONS

A.1

If THERMAL POWER exceeds 5% RTP, a positive reactivity addition could be occurring, and a nuclear excursion could result. To ensure that local LHR, DNBR, and RCS pressure limits are not violated, the reactor is immediately tripped. The necessary prompt action requires manual operator action to open the control rod drive trip breakers without attempts to reduce THERMAL POWER by actuating the control system (i.e., CONTROL ROD insertion or RCS boration).

B.1 and B.2

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the unit conditions. Boration will be continued until SDM is within limit. In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification. A Completion Time of one hour is provided for the operator to restore compliance with the excepted LCOs.

C.1

If the nuclear overpower trip setpoint is > 5% RTP, then 1 hour is allowed for the operator to restore the nuclear overpower trip setpoint within limits or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification, in order to ensure that continuity of reactor operation is within initial condition limits. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions.

If the nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit function is inoperable, then 1 hour is allowed for the operator to restore the functions to OPERABLE status or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions.

The nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit function is not required when the reactor power level is above the operating range of the instrumentation channel. For example, if the reactor power level is above the source range channel operating range, then only the intermediate range high startup rate CONTROL ROD withdrawal inhibit is required to be functional.

SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

Verification that THERMAL POWER is ≤ 5% RTP ensures that local LHR, DNBR, and RCS pressure limits are not violated and that entry into Actions Condition A is performed promptly. Hourly verification is adequate for the operator to determine any change in core conditions, such as xenon redistribution occurring after a THERMAL POWER reduction, that could cause THERMAL POWER to exceed the specified limit.

SR 3.1.9.2

Verification that the nuclear overpower trip setpoint is within the limit specified for PHYSICS TESTS ensures that core protection at the reduced power level is established during PHYSICS TESTS. Performing the verification once within 8 hours prior to the performance of PHYSICS TESTS allows the operator adequate time for verifying the established trip setpoint before initiating PHYSICS TESTS.

SR 3.1.9.3

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. CONTROL ROD position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Samarium concentration;

f. Xenon concentration;

g. Isothermal temperature coefficient (ITC), when below the point of adding heat (POAH);

h. Moderator defect, when above the POAH; and

i. Doppler defect, when above the POAH.

Using the ITC accounts for Doppler reactivity in this calculation when the reactor is subcritical or critical but below the POAH, and the fuel temperature will be changing at the same rate as the RCS.

SURVEILLANCE REQUIREMENTS (continued)

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50.59.
3. SAR, Section 3A.9.
4. SAR, Section 13.3, 13.4 and 13.6.
5. SAR, Section 13.4, Table 13-2.
6. 10 CFR 50.36.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Regulating Rod Insertion Limits

BASES

BACKGROUND

The insertion limits of the regulating rods are initial condition assumptions used in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect the core power distributions, the worth of a potential ejected rod, the assumptions of SDM, and the initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are described in SAR, Section 1.4, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability," GDC 28, "Reactivity Limits" (Ref. 1), and in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Limits on regulating rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are not violated.

The regulating rod groups operate with a predetermined amount of position overlap, in order to approximate a linear relation between rod worth and rod position (integral rod worth). To achieve this approximately linear relationship, the regulating rod groups are withdrawn and operated in a predetermined sequence. The automatic control system controls reactivity by moving the regulating rod groups in sequence within analyzed ranges. The group sequence and overlap limits are specified in the COLR.

The regulating rods are used for precise reactivity control of the reactor. The positions of the regulating rods are normally controlled automatically by the automatic control system but can also be controlled manually. They are capable of rapid reactivity changes compared with borating or diluting the Reactor Coolant System (RCS).

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that ensure that the criteria specified in 10 CFR 50.46 (Ref. 2) are not violated. Together, LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," provide limits on control component operation and on monitored process variables to ensure that the core operates within the linear heat rate limits in the COLR. Operation within the linear heat rate limits given in the COLR prevents power peaks that would exceed the loss of coolant accident (LOCA) limits derived from the analysis of the Emergency Core Cooling Systems (ECCS) and prevents departure from nucleate boiling (DNB) during a loss of forced

BACKGROUND (continued)

reactor coolant flow accident. In addition to the linear heat rate limits, certain reactivity limits are met by regulating rod insertion limits. The regulating rod insertion limits also restrict the ejected CONTROL ROD worth to the values assumed in the safety analysis and support the minimum required SDM in MODES 1 and 2.

This LCO is required to minimize fuel cladding failures that breach the primary fission product barrier and release fission products into the reactor coolant in the event of a LOCA, loss of flow accident, ejected rod accident, or other postulated accidents requiring termination by a Reactor Protection System trip function.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or abnormalities. The LCOs governing regulating rod insertion, APSR position, AXIAL POWER IMBALANCE, and QPT preclude core power distributions that violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed 2200°F (Ref. 2).

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition.

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 4).

d. The CONTROL RODS must be capable of shutting down the reactor with a minimum required SDM which assumes the highest worth CONTROL ROD stuck fully withdrawn.

Fuel cladding damage does not occur when the core is operated outside the conditions of these LCOs during normal operation. However, fuel cladding damage could result if an accident occurs with the simultaneous violation of one or more of the LCOs limiting the regulating rod position, the APSR position, the AXIAL POWER IMBALANCE, and the QPT. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local linear heat rates (LHRs).

The SDM requirement is met by limiting the regulating and safety rod insertion limits such that sufficient inserted reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes that the maximum worth rod remains fully withdrawn upon trip (Ref. 4). Operation at the SDM based regulating rod insertion limit may also indicate that the maximum ejected rod worth could be equal to the limiting value.

APPLICABLE SAFETY ANALYSES (continued)

Operation at the regulating rod insertion limits may cause the local core power to approach the maximum linear heat generation rate or peaking factor with the allowed QPT present.

The regulating rod and safety rod insertion limits ensure that the safety analysis assumptions for SDM, ejected rod worth, and power distribution peaking factors remain valid (Refs. 3 and 4).

The regulating rod insertion limits LCO satisfies Criterion 2 of 10 CFR 50.36 (Ref. 5).

LCO

The limits on regulating rod group physical insertion, sequence, and overlap, as defined in the COLR, must be maintained because they ensure that the resulting power distribution is within the range of analyzed power distributions and that the SDM and ejected rod worth are maintained.

The overlap between regulating groups provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during regulating rod motion.

Error adjusted maximum allowable setpoints for regulating rod insertion are provided in the COLR. The setpoints are derived by an adjustment of the measurement system independent limits to allow for THERMAL POWER level uncertainty and rod position errors.

LCO 3.2.1 has been modified by a Note that suspends the LCO requirement for those regulating rods not within the limits of the COLR solely due to testing in accordance with SR 3.1.4.2, which verifies the freedom of the rods to move. This SR may require the regulating rods to move below the LCO limit, out of group sequence, or beyond group overlap requirements, which would otherwise violate the LCO.

APPLICABILITY

The regulating rod physical insertion, sequence, and overlap limits shall be maintained with the reactor in MODES 1 and 2. These limits maintain the validity of the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions used in the safety analyses. Applicability in MODES 3, 4, and 5 is not required, because neither the power distribution nor ejected rod worth assumptions are exceeded in these MODES. SDM in MODES 3, 4, and 5 is governed by LCO 3.1.1, "SHUTDOWN MARGIN (SDM)."

ACTIONS

The regulating rod insertion setpoints provided in the COLR are based on the initial conditions assumed in the accident analyses and on the SDM. Specifically, separate insertion setpoints are specified to determine whether the unit is operating in violation of the initial conditions (e.g., the range of power distributions) assumed in the accident analyses or whether the unit is in violation of the SDM or ejected rod worth limits. Separate insertion setpoints are provided because different Required Actions and Completion Times apply, depending on which insertion setpoint has been violated. The area between the boundaries of the acceptable operation and unacceptable operation regions, illustrated on the regulating rod insertion setpoint figures in the COLR, is the restricted operation region. The actions required when operation occurs in the restricted operation region are described under Condition A. The actions required when operation occurs in the unacceptable operation region are described under Condition D. The actions required when operation occurs with the regulating rod group sequence or overlap requirements not met are described under Condition C.

A.1

Operation with the regulating rods in the restricted operation region shown on the regulating rod insertion setpoint figures specified in the COLR potentially violates the LOCA LHR limits, or the loss of flow accident DNB peaking limits.

For verification that LHRs are within their limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three dimensional power distribution map. Verification that LHRs are within their limits ensures that operation with the regulating rods inserted into the restricted operation region does not violate the ECCS or DNB criteria. The required Completion Time of 2 hours is acceptable in that it allows the operator sufficient time for obtaining a power distribution map and for verifying the LHRs. Repeating SR 3.2.5.1 every 2 hours is acceptable because it ensures that continued verification of the LHRs is performed as core conditions (primarily regulating rod insertion and induced xenon redistribution) change.

Monitoring the LHRs does not provide verification that the reactivity insertion rate on the rod trip or the ejected rod worth limit is maintained, because worth is a reactivity parameter rather than a power peaking parameter. However, if the COLR figures do not show that a rod insertion setpoint is ejected rod worth limited, then the ejected rod worth is no more limiting than the SDM based rod insertion limit in the core design. Ejected rod worth limits are independently maintained by the Required Actions of Conditions A and D.

Required Action A.1 is modified by a Note that requires the performance of SR 3.2.5.1 only when THERMAL POWER is greater than 20% RTP. This establishes a Required Action that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

ACTIONS (continued)

A.2

Indefinite operation with the regulating rods inserted in the restricted operation region is not prudent. Even if power peaking monitoring per Required Action A.1 is continued, reactivity limits may not be met and the abnormal regulating rod insertion may cause an adverse xenon redistribution, may cause the limits on AXIAL POWER IMBALANCE to be exceeded, or may adversely affect the long term fuel depletion pattern. Therefore, restoration of regulating rod groups to within their limits is required within 24 hours after discovery of failure to meet the requirements of this LCO. This required Completion Time is reasonable based on the low probability of an event occurring simultaneously with the limit out of specification in this relatively short time period. In addition, it precludes long term depletion with abnormal group insertions, thereby limiting the potential for an adverse xenon redistribution.

B.1

If the regulating rods cannot be positioned within the acceptable operation region shown on the figures in the COLR within the required Completion Time (i.e., Required Action A.2 not met), then the setpoints can be restored by reducing the THERMAL POWER to a value allowed by the regulating rod insertion setpoints in the COLR. The required Completion Time of 2 hours is sufficient to allow the operator to complete the power reduction in an orderly manner and without challenging the unit systems. Operation for up to 2 hours more in the restricted operation region shown in the COLR is acceptable, based on the low probability of an event occurring simultaneously with the regulating rod position out of specification in this relatively short time period.

C.1

Operation with the regulating rod groups out of sequence or with the group overlap limits exceeded may represent a condition beyond the assumptions used in the safety analyses. The design calculations assume no deviation in nominal overlap between regulating rod groups. However, small deviations in group overlap, as allowed by the COLR, may occur and would not cause significant differences in core reactivity, in power distribution, or rod worth, relative to the design calculations. Group sequence must be maintained because design calculations assume the regulating rods withdraw and insert in a predetermined order. The Completion Time of 4 hours is intended to restrict operation in this condition because of the potential severity associated with gross violations of group sequence or overlap requirements. The 4 hour Completion Time is based on operating experience which supports the restoration time without unnecessarily challenging unit operation and the low probability of an event occurring simultaneously with the limit out of specification.

ACTIONS (continued)

D.1

Operation in the unacceptable operation region shown on the figures in the COLR corresponds to power operation with an SDM less than the minimum required value or with the ejected rod worth greater than the allowable value. The regulating rods may be inserted too far to provide sufficient negative reactivity insertion following a reactor trip and the ejected rod worth may exceed its initial condition limit. Therefore, the RCS boron concentration must be increased to restore the regulating rod insertion to a value that preserves the SDM and ejected rod worth limits. The required Completion Time of 15 minutes to initiate boration is reasonable, based on limiting the potential xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this Action. This period allows the operator sufficient time for aligning the required valves and for starting the boric acid pumps. Boration continues until the regulating rod group positions are restored to at least within the restricted operation region, which restores the minimum SDM and reduces the potential ejected rod worth to within its limit.

D.2.1

The required Completion Time of 2 hours from initial discovery of a regulating rod group in the unacceptable operation region until its restoration to within the restricted operation region shown on the figures in the COLR allows sufficient time for borated water to enter the RCS from the chemical addition and makeup and purification systems, thereby allowing the regulating rods to be withdrawn to the restricted operation region. Operation in the restricted operation region for up to 2 hours is reasonable, based on limiting the potential for an adverse xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this Action.

D.2.2

The SDM and ejected rod worth limit can also be restored by reducing the THERMAL POWER to a value allowed by the regulating rod insertion setpoints in the COLR. The required Completion Time of 2 hours is sufficient to allow the operator to complete the power reduction in an orderly manner and without challenging the unit systems. Operation for up to 2 hours in the restricted operation region shown in the COLR is acceptable, based on the low probability of an event occurring simultaneously with the limit out of specification in this relatively short time period. In addition, it precludes long term depletion with abnormal group insertions or configurations and limits the potential for an adverse xenon redistribution.

ACTIONS (continued)

E.1

If the Required Actions and associated Completion Times of Conditions C or D are not met, then the reactor is placed in MODE 3, in which this LCO does not apply. This Action ensures that the reactor does not continue operating in violation of the peaking limits, the ejected rod worth, the reactivity insertion rate assumed as initial conditions in the accident analyses, or the required minimum SDM assumed in the accident analyses. The required Completion Time of 6 hours is reasonable, based on operating experience regarding the amount of time required to reach MODE 3 from RTP without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

This Surveillance ensures that the sequence and overlap limits are not violated. A Surveillance Frequency of 12 hours is acceptable because little rod motion occurs during this period due to fuel burnup. Also, the Frequency takes into account other information available in the control room for monitoring the status of the regulating rods.

SR 3.2.1.2

Verification of the regulating rod insertion setpoints as specified in the COLR at a Frequency of 12 hours is sufficient to detect regulating rod banks that may be approaching the group insertion setpoints, because little rod motion due to fuel burnup occurs in 12 hours. Also, the Frequency takes into account other information available in the control room for monitoring the status of the regulating rods.

SR 3.2.1.3

Prior to achieving criticality, an estimated critical position for the CONTROL RODS is determined. Verification that SDM meets the minimum requirements ensures that sufficient SDM capability exists with the CONTROL RODS at the estimated critical position if it is necessary to shut down or trip the reactor after criticality. The Frequency of 4 hours prior to criticality provides sufficient time to verify SDM capability and establish the estimated critical position.

REFERENCES

1. SAR, Section 1.4, GDC 10, GDC 26 and GDC 28.
2. 10 CFR 50.46.
3. SAR, Chapter 3.
4. SAR, Chapter 14.
5. 10 CFR 50.36.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 AXIAL POWER SHAPING ROD (APSR) Insertion Limits

BASES

BACKGROUND

The insertion limits of the APSRs are initial condition assumptions in all safety analyses that are affected by core power distributions. The applicable criterion for these power distribution design requirements are SAR Section 1.4, GDC 10, "Reactor Design" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Limits on APSR insertion have been established, and all APSR positions are monitored and controlled during power operation to ensure that the power distribution defined by the design power peaking limits is maintained.

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that meet the criteria specified in Reference 2. Together, LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," provide limits on control component operation and on monitored process variables to ensure that the core operates within the linear heat rate (LHR) limits in the COLR. Operation within the LHR limits given in the COLR prevents power peaks that exceed the loss of coolant accident (LOCA) limits derived from the analysis of the Emergency Core Cooling Systems (ECCS) and prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident. The APSRs do not insert upon a reactor trip.

This LCO is required to minimize fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, loss of flow accident, ejected rod accident, or other postulated accident requiring termination by a Reactor Protection System trip function.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or abnormalities. Acceptance criteria for the safety and regulating rod insertion, APSR position, AXIAL POWER IMBALANCE, and QPT LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed 2200°F (Ref. 2);

APPLICABLE SAFETY ANALYSES (continued)

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 3); and

d. CONTROL RODS must be capable of shutting down the reactor with a minimum required SDM which assumes the highest worth CONTROL ROD stuck fully withdrawn (GDC 26, Ref. 1).

Fuel cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result should an accident occur simultaneously with violation of one or more of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and corresponding increased local linear heat rates.

Operation at the APSR insertion limits may approach the maximum allowable linear heat generation rate with the allowed QPT present.

In MODES 1 and 2 while critical, the APSR insertion limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 4). In MODE 2 while subcritical, the APSR insertion limits satisfy Criterion 4 of 10 CFR 50.36.

LCO

The setpoints on APSR physical insertion as defined in the COLR must be maintained because they serve the function of controlling the power distribution within an acceptable range.

The fuel cycle design assumes APSR withdrawal at the EFPD burnup window specified in the COLR. Prior to this window, the APSRs are maintained in accordance with operating guidelines provided by reactor engineering during steady state operation. After this window, the APSRs are not allowed to be reinserted for the remainder of the fuel cycle.

APPLICABILITY

The APSR physical insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits maintain the power distribution within the range assumed in the accident analyses. In MODES 1 and 2, the limits on APSR insertion specified by this LCO maintain the axial fuel burnup design conditions assumed in the reload safety evaluation analysis. Applicability in MODES 3, 4, and 5 is not required, because the reactor is subcritical.

ACTIONS

For steady state power operation, a normal position for APSR insertion is specified in the station operating procedures. The APSRs may be positioned as necessary for transient AXIAL POWER IMBALANCE control until the fuel cycle design requires them to be fully withdrawn. (Not all fuel cycles may incorporate APSR withdrawal.) APSR position limits are not imposed for gray APSRs, with two exceptions. If the fuel cycle design incorporates an APSR withdrawal (usually near end of cycle (EOC)), the APSRs may not be maintained in the fully withdrawn position prior to the fuel cycle burnup for the APSR withdrawal. If this occurs, the APSRs must be restored to their normal inserted position. Conversely, after the fuel cycle burnup for the APSR withdrawal occurs, the APSRs may not be reinserted for the remainder of the fuel cycle. These restrictions apply to ensure the axial burnup distribution that accumulates in the fuel will be consistent with the expected (as designed) distribution.

A.1

For verification that the core linear heat rates are within their limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three dimensional power distribution map. Successful verification that the LHRs are within their limits ensures that operation with the APSRs inserted or withdrawn in violation of the setpoints specified in the COLR do not violate either the ECCS or DNB criteria. The required Completion Time of 2 hours is reasonable to allow the operator to obtain a power distribution map and to verify the LHRs. Repeating SR 3.2.5.1 every 2 hours is reasonable to ensure that continued verification of the LHRs is obtained as core conditions (primarily the regulating rod insertion and induced xenon redistribution) change.

Required Action A.1 is modified by a Note that requires the performance of SR 3.2.5.1 only when THERMAL POWER is greater than 20% RTP. This establishes a Required Action that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

A.2

Indefinite operation with the APSRs positioned in violation of the setpoints specified in the COLR is not prudent. Even if LHR monitoring per Required Action A.1 is continued, the abnormal APSR positioning may cause an adverse xenon redistribution, may cause the limits on AXIAL POWER IMBALANCE to be exceeded, or may affect the long term fuel depletion pattern. Therefore, operation is allowed for up to 24 hours. This required Completion Time is reasonable based on the low probability of an event occurring simultaneously with the APSR position out of specification. In addition, it precludes long term depletion with the APSRs in positions that have not been analyzed, thereby limiting the potential for an adverse xenon redistribution. This time limit also ensures that the intended burnup distribution is maintained, and allows the operator sufficient time to reposition the APSRs to correct their positions.

ACTIONS (continued)

A.2 (continued)

Because the APSRs are not operated by the automatic control system, manual action by the operator is required to restore the APSRs to the positions specified in the COLR.

B.1

If the Required Action and associated Completion Time are not met, the reactor must be placed in MODE 3, in which this LCO does not apply. This action ensures that the fuel does not continue to be depleted in an unintended burnup distribution. The required Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 3 from RTP in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

Fuel cycle designs that allow APSR withdrawal near end of cycle (EOC) do not permit reinsertion of APSRs after the time of withdrawal. Verification that the APSRs are within their insertion setpoints at a 12 hour Frequency is sufficient to ensure that the APSR insertion setpoints are preserved. The 12 hour Frequency required for performing this verification is sufficient because APSRs are positioned by manual control and are normally moved infrequently. The Frequency takes into account other information available in the control room for monitoring the axial power distribution in the reactor core.

REFERENCES

1. SAR Section 1.4, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. SAR, Chapter 14.
4. 10 CFR 50.36.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL POWER IMBALANCE Operating Limits

BASES

BACKGROUND

This LCO is required to limit the core power distribution based on accident initial condition criteria.

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that satisfy the criteria specified in 10 CFR 50.46 (Ref. 1). This LCO provides limits on AXIAL POWER IMBALANCE to ensure that the core operates within the linear heat rate (LHR) limits given in the COLR. Operation within the LHR limits given in the COLR prevents power peaks that exceed the loss of coolant accident (LOCA) limits derived from the analysis of the Emergency Core Cooling Systems (ECCS) and prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident.

This LCO is required to limit fuel cladding failures that breach the primary fission product barrier and release fission products into the reactor coolant in the event of a LOCA, loss of forced reactor coolant flow accident, or other postulated accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by maintaining the validity of the assumptions in the safety analyses related to the initial power distribution and reactivity.

Fuel cladding failure during a postulated LOCA is limited by restricting the maximum LHR so that the peak cladding temperature does not exceed 2200°F (Ref. 1). Peak cladding temperatures > 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, peak cladding temperature is usually most limiting.

Proximity to the DNB condition is expressed by the departure from nucleate boiling ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and anticipated transients is limited to the DNBR correlation limit for the particular fuel design in use and is accepted as an appropriate margin to DNB. The DNB correlation limit ensures that there is at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB.

The measurement system independent limits on AXIAL POWER IMBALANCE are determined analytically by the reload safety evaluation analysis without adjustment for measurement system error and uncertainty. Operation beyond these limits could invalidate the assumptions used in the accident analyses regarding the core power distribution. The AXIAL POWER IMBALANCE setpoints provided in the COLR account for measurement system error and uncertainty.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and abnormalities. The LCOs based on power distribution, LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," preclude core power distributions that would violate the following fuel design criteria:

a. During a large break LOCA, peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition.

The regulating rod positions, the APSR positions, the AXIAL POWER IMBALANCE, and the QPT are process variables that characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside this LCO during normal operation. However, fuel cladding damage could result should an accident occur with simultaneous violation of one or more of the LCOs governing the four process variables cited above. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

The regulating rod insertion, the APSR positions, the AXIAL POWER IMBALANCE, and the QPT are monitored and controlled during power operation to ensure that the power distribution is within the bounds set by the safety analyses. The axial power distribution is maintained primarily by the AXIAL POWER IMBALANCE and the APSR position limits; and the radial power distribution is maintained primarily by the QPT limits. The regulating rod insertion limits affect both the radial and axial power distributions.

The dependence of the core power distribution on burnup, regulating rod insertion, APSR position, and spatial xenon distribution is taken into account when the reload safety evaluation analysis is performed.

Operation at the AXIAL POWER IMBALANCE limit must be interpreted as operating the core at the maximum allowable LHR assumed as initial conditions for the accident analyses with the allowed QPT present.

AXIAL POWER IMBALANCE satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2).

LCO

The power distribution LCO limits have been established based on correlations between power peaking and easily measured process variables: regulating rod position, APSR position, AXIAL POWER IMBALANCE, and QPT. The AXIAL POWER IMBALANCE envelope contained in the COLR represents the setpoints beyond which the core power distribution could either exceed the LOCA LHR limits or cause a reduction in the DNBR below the Safety Limit during the loss of flow accident with the allowable QPT present and with the APSR positions consistent with the limitations on APSR withdrawal determined by the fuel cycle design and specified by LCO 3.2.2.

The AXIAL POWER IMBALANCE maximum allowable setpoints (measurement system dependent limits) applicable for the full Incore Detector System, the Minimum Incore Detector System, and the Excore Detector System are provided in the COLR.

APPLICABILITY

In MODE 1, the limits on AXIAL POWER IMBALANCE must be maintained when THERMAL POWER is > 40% RTP to prevent the core power distribution from exceeding the LOCA and loss of flow assumptions used in the accident analyses. Applicability of these limits at ≤ 40% RTP in MODE 1 is not required. This operation is acceptable based on engineering judgment because the combination of AXIAL POWER IMBALANCE with the maximum allowable THERMAL POWER level will not result in LHRs sufficiently large to violate the fuel design limits. In MODES 2, 3, 4, 5, and 6, this LCO is not applicable because the reactor is not generating sufficient THERMAL POWER to produce fuel damage.

ACTIONS

A.1

The AXIAL POWER IMBALANCE operating setpoints that maintain the validity of the assumptions regarding the power distributions in the accident analyses of the LOCA and the loss of flow accident are provided in the COLR. Operation within the AXIAL POWER IMBALANCE setpoints given in the COLR is the acceptable region of operation. Operation in violation of the AXIAL POWER IMBALANCE setpoints given in the COLR is the restricted region of operation.

Operation with AXIAL POWER IMBALANCE in the restricted region shown on the AXIAL POWER IMBALANCE figures in the COLR potentially violates the LOCA LHR limits or the loss of flow accident DNB peaking limits or both. For verification that core local LHRs are within their specified limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three dimensional power distribution map. Verification that core local LHRs are within their specified limits ensures that

ACTIONS (continued)

A.1 (continued)

operation with the AXIAL POWER IMBALANCE in the restricted region does not violate the ECCS or 95/95 DNB criteria. The required Completion Time of 2 hours provides reasonable time for the operator to obtain a power distribution map and to determine and verify that the core local LHRs are within their specified limits. The 2 hour Frequency provides reasonable time to ensure that continued verification of the core local LHRs is obtained as core conditions (primarily regulating rod insertion and induced xenon redistribution) change, because little rod motion occurs in 2 hours due to fuel burnup, the potential for xenon redistribution is limited, and the probability of an event occurring in this short time frame is low.

A.2

Indefinite operation with the AXIAL POWER IMBALANCE in the restricted region is not prudent. Even if LHR monitoring per Required Action A.1 is continued, excessive AXIAL POWER IMBALANCE over an extended period of time may cause a potentially adverse xenon redistribution to occur. Therefore, LHR monitoring is only allowed for a maximum of 24 hours. This required Completion Time is reasonable based on the low probability of a limiting event occurring simultaneously with the AXIAL POWER IMBALANCE outside the setpoints of this LCO. In addition, this limited Completion Time precludes long term depletion of the reactor fuel with excessive AXIAL POWER IMBALANCE and gives the operator sufficient time to reposition the APSRs or regulating rods to reduce the AXIAL POWER IMBALANCE because adverse effects of xenon redistribution and fuel depletion are limited.

B.1

If the Required Actions and the associated Completion Times of Condition A are not met, the AXIAL POWER IMBALANCE may exceed its specified limits and the reactor may be operating with a global axial power distribution mismatch. Continued operation in this configuration may induce an axial xenon oscillation and may result in an increased linear heat generation rate when the xenon redistributes. Reducing THERMAL POWER to $\leq$ 40% RTP reduces the maximum LHR to a value that does not exceed the LHR initial condition limits assumed in the accident analyses. The required Completion Time of 4 hours is reasonable based on limiting a potentially adverse xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this Action.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The AXIAL POWER IMBALANCE can be monitored by both the Incore and Excore Detector Systems. The AXIAL POWER IMBALANCE maximum allowable setpoints are derived from their corresponding measurement system independent limits by adjusting for both the system observability errors and instrumentation errors. Although they may be based on the same measurement system independent limits, the setpoints for the different systems are not identical because of differences in the errors applicable for each of these systems. The uncertainty analysis that defines the required error adjustment to convert the measurement system independent limits to full incore detector system limits assumes that 75% of the detectors in each quadrant are OPERABLE. Detectors located on the core major axes are assumed to contribute one half of their output to each quadrant; detectors in the center assembly are assumed to contribute one quarter of their output to each quadrant. For AXIAL POWER IMBALANCE measurements using the Incore Detector System, the Minimum Incore Detector System consists of OPERABLE detectors configured as follows:

a. Nine detectors shall be arranged such that there are three detectors in each of three strings and there are three detectors lying in the same axial plane, with one plane at the core midplane and one plane in each axial core half;

b. The axial planes in each core half shall be symmetrical about the core midplane; and

c. The detector strings shall not have radial symmetry.

Figure B 3.2.3-1 (Minimum Incore Detector System for AXIAL POWER IMBALANCE Measurement) depicts an example of this configuration. This arrangement is chosen to reduce the uncertainty in the measurement of the AXIAL POWER IMBALANCE by the Minimum Incore Detector System. For example, the requirement for placing one detector of each of the three strings at the core midplane puts three detectors in the central region of the core where the neutron flux tends to be higher. It also helps prevent measuring an AXIAL POWER IMBALANCE that is excessively large when the reactor is operating at low THERMAL POWER levels. The third requirement for placement of detectors (i.e., radial asymmetry) reduces uncertainty by measuring the neutron flux at core locations that are not radially symmetric.

Verification of the AXIAL POWER IMBALANCE indication every 12 hours ensures that the AXIAL POWER IMBALANCE setpoints are not violated and takes into account other information and alarms available in the control room. This Surveillance Frequency is acceptable because the mechanisms that can cause AXIAL POWER IMBALANCE, such as xenon redistribution or control rod drive mechanism malfunctions that cause slow AXIAL POWER IMBALANCE increases, can be discovered by the operator before the specified limits are violated.

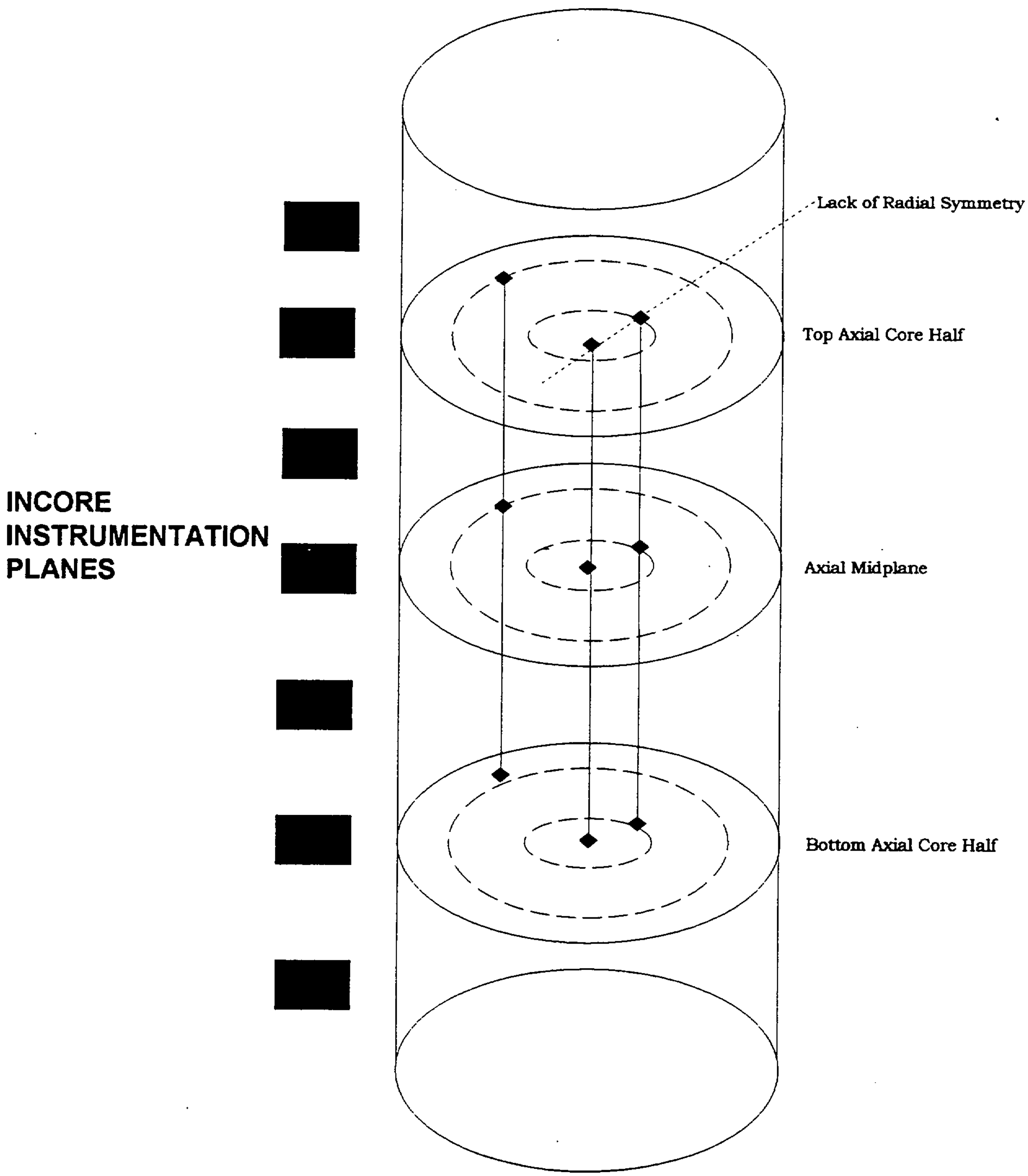
REFERENCES

1. 10 CFR 50.46.

2. 10 CFR 50.36.

Figure B 3.2.3-1

Minimum Incore System for AXIAL POWER IMBALANCE Measurement



B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT (QPT)

BASES

BACKGROUND

This LCO is required to limit the core power distribution based on accident initial condition criteria.

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 1). Together, LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," provide limits on control component operation and on monitored process variables to ensure that the core operates within the linear heat rate (LHR) limits given in the COLR. Operation within the LHR limits given in the COLR prevents power peaks that exceed the loss of coolant accident (LOCA) limits derived by Emergency Core Cooling Systems (ECCS) analysis and prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident.

This LCO is required to limit fuel cladding failures that breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, loss of forced reactor coolant flow, or other accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by maintaining the validity of the assumptions used in the safety analysis related to the initial power distribution and reactivity.

Fuel cladding failure during a postulated LOCA is limited by restricting the maximum LHR so that the peak cladding temperature does not exceed 2200°F (Ref. 1). Peak cladding temperatures > 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, peak cladding temperature is usually most limiting.

Proximity to the DNB condition is expressed by the departure from nucleate boiling ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and anticipated transients is limited to the DNBR correlation limit for the particular fuel design in use, and is accepted as an appropriate margin to DNB. The DNBR correlation limit ensures that there is at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB.

BACKGROUND (continued)

The measurement system independent limits on QPT are determined analytically by the reload safety evaluation analysis without adjustment for measurement system error and uncertainty. Operation beyond these limits could invalidate core power distribution assumptions used in the accident analysis. The error adjusted maximum allowable setpoints (measurement system dependent limits) for QPT are specified in the COLR.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and abnormalities. The LCOs based on power distribution (LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4) preclude core power distributions that violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed 2200°F (Ref. 1).

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition.

QPT is one of the process variables that characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside this LCO during normal operation. However, fuel cladding damage could result if an accident occurs with simultaneous violation of one or more of the LCOs governing the core power distribution. Changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.

The dependence of the core power distribution on burnup, regulating rod insertion, APSR position, and spatial xenon distribution is taken into account during the reload safety evaluation analysis. An allowance for QPT is accommodated in the analysis and resultant LCO limits. The increase in peaking taken for QPT is developed from a database of full core power distribution calculations (Ref. 2). The calculations consist of simulations of many power distributions with tilt causing mechanisms (e.g., dropped or misaligned CONTROL RODS, broken APSR fingers fully inserted, misloaded assemblies, and burnup gradients). An increase of < 2% peak power per 1% QPT is supported by the analysis, therefore a value of 2% peak power increase per 1% QPT is used to bound peak power increases due to QPT.

Operation at the AXIAL POWER IMBALANCE or rod insertion limits must be interpreted as operating the core at the maximum allowable LHR for accident initial conditions with the allowed QPT present.

QPT satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

The power distribution LCO limits have been established based on correlations between power peaking and easily measured process variables: regulating rod position, APSR position, AXIAL POWER IMBALANCE, and QPT. The regulating rod insertion setpoints and the AXIAL POWER IMBALANCE boundaries contained in the COLR represent the measurement system dependent limits at which the core power distribution could either exceed the LOCA LHR limits or cause a reduction in DNBR below the safety limit during a loss of flow accident with the allowable QPT present and with an APSR position consistent with the limitations on APSR position determined by the fuel cycle design and specified by LCO 3.2.2.

The allowable setpoints for steady state and maximum setpoints for QPT applicable for the full symmetrical Incore Detector System, Minimum Incore Detector System, and Excore Detector System are provided in the COLR. The setpoints for the three systems are derived by adjustment of the measurement system independent QPT limits also given in the COLR to allow for system observability and instrumentation errors.

APPLICABILITY

In MODE 1, the limits on QPT must be maintained when THERMAL POWER is > 20% RTP to prevent the core power distribution from exceeding the design limits. The minimum power level of 20% RTP is large enough to obtain meaningful QPT indications without compromising safety.

In MODE 2, the combination of QPT with maximum ALLOWABLE THERMAL POWER level does not result in LHRs sufficiently large to violate the fuel design limits, and therefore, applicability in this MODE is not required. Although not specifically addressed in the LCO, QPTs greater than the maximum setpoint specified in the COLR in MODE 1 with THERMAL POWER < 20% RTP are allowed based on engineering judgement.

In MODES 3, 4, 5, and 6, this LCO is not applicable, because the reactor is not generating significant THERMAL POWER and QPT is indeterminate.

ACTIONS

A.1.1

The steady state setpoint specified in the COLR provides an allowance for QPT that may occur during normal operation. A peaking increase to accommodate QPTs up to the steady state setpoint is allowed by the regulating rod insertion limits of LCO 3.2.1 and the AXIAL POWER IMBALANCE limits of LCO 3.2.3.

ACTIONS (continued)

A.1.1 (continued)

Operation with QPT greater than the steady state setpoint specified in the COLR potentially violates the LOCA LHR limits, or loss of flow accident DNB peaking limits, or both. For verification that core local LHRs are within their specified limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three dimensional power distribution map. Verification that core local LHRs are within their limits ensures that operation with QPT greater than the steady state setpoint does not violate the ECCS or 95/95 DNB criteria. The required Completion Time of once per 2 hours is a reasonable amount of time to allow the operator to obtain a power distribution map and to verify the core local LHRs. Repeating SR 3.2.5.1 every 2 hours is a reasonable Frequency at which to ensure that continued verification of the core local LHRs is obtained as core conditions that influence QPT change.

A.1.2.1

The safety analysis has shown that a conservative corrective action is to reduce THERMAL POWER by 2% RTP or more from the ALLOWABLE THERMAL POWER for each 1% of QPT in excess of the steady state setpoint. This action limits the local LHR to a value corresponding to the assumed accident initial condition limits. The required Completion Time of 2 hours is reasonable, based on limiting the potential for xenon redistribution, the low probability of an accident occurring, and the steps required to complete the Required Action.

If QPT can be reduced to less than or equal to the steady state setpoint in < 2 hours, the reactor may return to normal operation without undergoing a power reduction. Significant radial xenon redistribution does not occur within this amount of time.

The required Completion Time of 2 hours after the last performance of SR 3.2.5.1 allows reduction of THERMAL POWER in the event the operators cannot or choose not to continue to perform SR 3.2.5.1 as required by Required Action A.1.1.

A.1.2.2

Power operation is allowed to continue if THERMAL POWER is reduced in accordance with Required Action A.1.2.1. The same reduction (i.e., 2% RTP or more) is also applicable to the nuclear overpower based on Reactor Coolant System (RCS) flow and AXIAL POWER IMBALANCE trip setpoint, for each 1% of QPT in excess of the steady state limit. This reduction maintains both core protection and thermal margins at the reduced THERMAL POWER level similar to that at RTP. The required Completion Time of 10 hours or 10 hours after the last performance of SR 3.2.5.1 is reasonable based on the need to limit the potentially adverse xenon redistribution, the low probability of an accident occurring while operating with the QPT limits not met, and the number of steps required to complete the Required Action.

ACTIONS (continued)

A.1.2.3

Power operation is allowed to continue if restrictions are imposed on the allowed degree of regulating group insertion. This Required Action requires a reduction in the regulating group insertion setpoints given in the COLR by ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state setpoint. Based on engineering judgment, this action is intended to reduce the potential power peaking associated with regulating rod group insertion into the core.

The Completion Time of 10 hours is reasonable based on the need to limit the potentially adverse xenon redistribution, the low probability of an accident occurring while operating with QPT limits not met, and the number of steps required to complete the Required Action. The second Completion Time of 10 hours after the last performance of SR 3.2.5.1 is based on the same reasoning and is provided in the event the operators cannot or choose not to continue to perform SR 3.2.5.1 as required by Required Action A.1.1.

A.1.2.4

Power operation is allowed to continue if restrictions are imposed on the allowed Operational Power Imbalance Setpoints given in the COLR. This Required Action results in a reduction in the allowed THERMAL POWER level as a function of AXIAL POWER IMBALANCE by ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit. Based on engineering judgment, this action is intended to reduce the potential power peaking associated with the combined affects of operating with an AXIAL POWER IMBALANCE and a QPT.

The Completion Time of 10 hours is reasonable based on the need to limit the potentially adverse xenon redistribution, the low probability of an accident occurring while operating with QPT limits not met, and the number of steps required to complete the Required Action. The second Completion Time of 10 hours after the last performance of SR 3.2.5.1 is based on the same reasoning and is provided in the event the operators cannot or choose not to continue to perform SR 3.2.5.1 as required by Required Action A.1.1.

A.2

Although the actions directed by Required Action A.1.2.1 restore thermal margins, if the source of the QPT is not established and corrected, it is prudent to establish increased margins. A required Completion Time of 24 hours to reduce QPT to less than the steady state limit is a reasonable time for investigation and corrective measures.

ACTIONS (continued)

B.1

If the Required Actions and associated Completion Times of Condition A are not met, a further power reduction is required. Power reduction to < 60% of ALLOWABLE THERMAL POWER provides conservative protection from increased peaking due to xenon redistribution. The required Completion Time of 2 hours is reasonable to allow the operator to reduce THERMAL POWER to < 60% of ALLOWABLE THERMAL POWER without challenging unit systems.

B.2

Reduction of the nuclear overpower trip setpoint to ≤ 65.5% of ALLOWABLE THERMAL POWER after THERMAL POWER has been reduced to < 60% of ALLOWABLE THERMAL POWER maintains both core protection and OPERABILITY margin at reduced power similar to that at full power. The required Completion Time of 10 hours allows the operator sufficient time to reset the trip setpoint and is reasonable based on operating experience.

C.1

If the Required Actions and associated Completion Times of Condition B are not met, then the reactor will continue in power operation with significant QPT. Either the power level has not been reduced to comply with the Required Action or the nuclear overpower trip setpoint has not been reduced within the required Completion Time. To preclude risk of fuel damage in any of these conditions, THERMAL POWER is reduced further. Operation below 20% RTP allows the operator to investigate the cause of the QPT and to correct it. Local LHRs with a large QPT do not violate the fuel design limits at or below 20% RTP. The required Completion Time of 4 hours is acceptable based on limiting the potential increase in local LHRs that could occur due to xenon redistribution with the QPT out of specification.

D.1

QPT in excess of the maximum setpoint specified in the COLR can be an indication of a severe power distribution anomaly, and a power reduction to at most 20% RTP ensures local LHRs do not exceed allowable limits while the cause is being determined and corrected.

The required Completion Time of 4 hours is reasonable to allow the operator to reduce THERMAL POWER to ≤ 20% RTP without challenging unit systems.

SURVEILLANCE REQUIREMENTS

QPT can be monitored by both the Incore and Excore Detector systems. The QPT setpoints are derived from their corresponding measurement system independent limits by adjustment for system observability errors and instrumentation errors. Although they may be based on the same measurement system independent limit, the limits for the different systems are not identical because of differences in the errors applicable for these systems. For QPT measurements using the Incore Detector System, the Minimum Incore Detector System consists of OPERABLE detectors configured as follows:

a. Two sets of four detectors shall lie in each core half. Each set of detectors shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane.

b. Detectors in the same plane shall have quarter core radial symmetry.

Figure B 3.2.4-1 (Minimum Incore Detector System for QPT Measurement) depicts an example of this configuration. The symmetric full Incore Detector System for QPT uses the Incore Detector System as described above and is configured such that at least 75% of the detectors in each core quadrant are OPERABLE.

SR 3.2.4.1

Checking the QPT indication every 7 days ensures that the operator can determine whether the plant computer software and Incore Detector System inputs for monitoring QPT are functioning properly, and takes into account other information and alarms available to the operator in the control room. This procedure allows the QPT mechanisms, such as xenon redistribution, burnup gradients, and CONTROL ROD drive mechanism malfunctions, which can cause slow development of a QPT, to be detected. Operating experience has confirmed the acceptability of a Surveillance Frequency of 7 days.

Following restoration of the QPT to within the setpoint, operation at ≥ 95% RTP may proceed provided the QPT is determined to remain within the setpoint at the increased THERMAL POWER level. In case QPT exceeds the setpoint for more than 24 hours (Condition A), the potential for xenon redistribution is greater. Therefore, the QPT is monitored for 12 consecutive hourly intervals to determine whether the period of any oscillation due to xenon redistribution causes the QPT to exceed the setpoint again.

REFERENCES

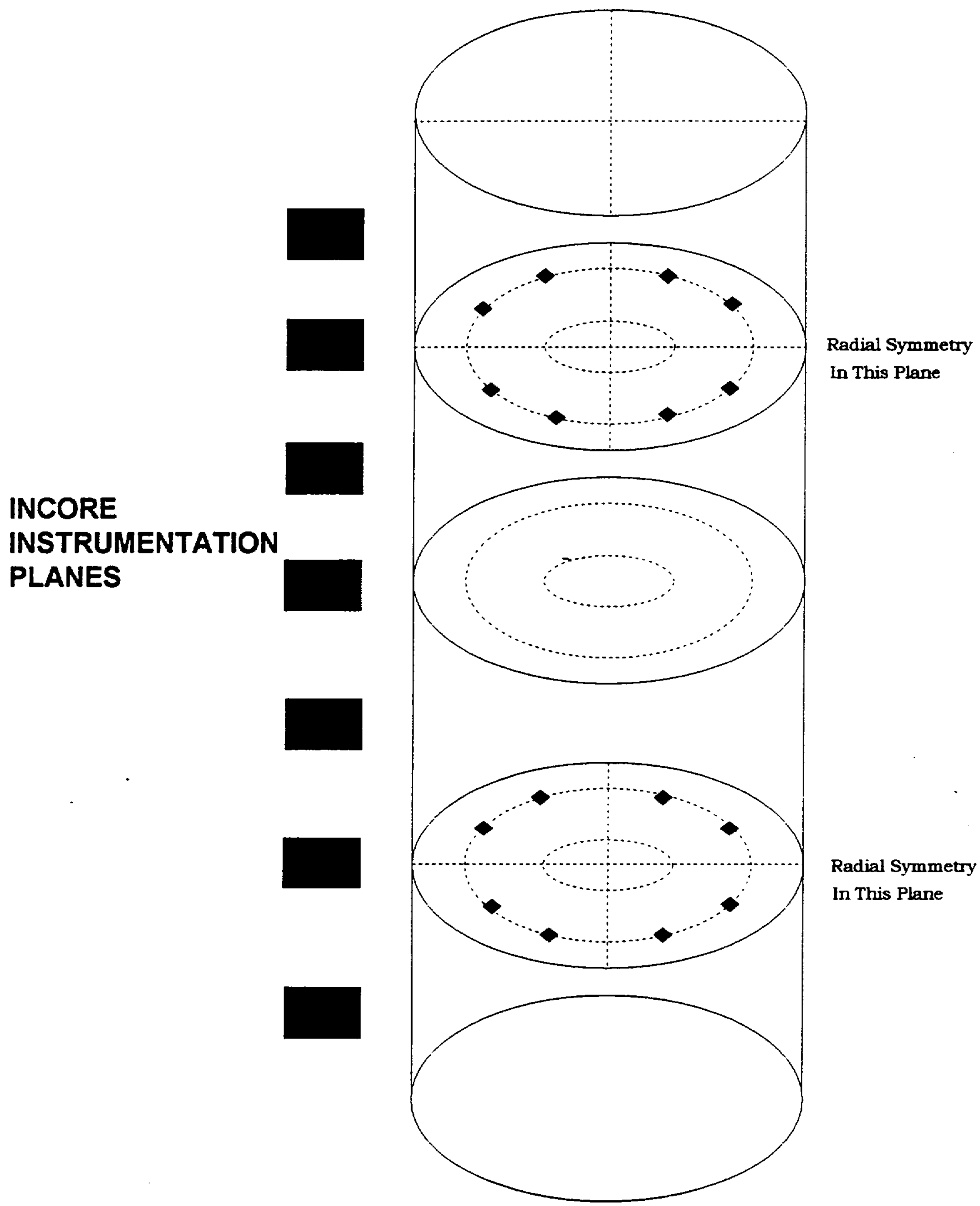
1. 10 CFR 50.46

2. BAW 10122A, "Normal Operating Controls," Rev. 1, May 1984.

3. 10 CFR 50.36

Figure B 3.2.4-1

Minimum Incore System for QUADRANT POWER Tilt Measurement



B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 Power Peaking

BASES

BACKGROUND

The purpose of this LCO is to establish limits that constrain the core power distribution within design limits during normal operation, during abnormalities and such that accident initial condition protection criteria are preserved. The accident initial condition criteria are preserved by bounding operation within specified acceptable fuel design limits. This is accomplished by limiting the local linear heat rate (LHR) to three general constraints: 1) the LHR may not exceed a value that results in fuel centerline melt, 2) the LHR may not exceed a value that would result in peak cladding temperatures of greater than 2200°F during a loss of coolant accident (LOCA), and 3) the LHR may not exceed a value that would result in the minimum departure from nucleate boiling ratio (DNBR) dropping below the specified acceptable fuel design limits in the event of the limiting loss of flow transient.

The LOCA-limited LHR is a specified acceptable fuel design limit that preserves the initial conditions for the Emergency Core Cooling Systems (ECCS) analysis. The LOCA-limited LHR is dependent upon core axial location and fuel batch design. The LOCA-limited LHR may be designated as LHR in units kW/ft or as a power peaking factor. When expressed as a power peaking factor, the LOCA-limited LHR is designated as $F_Q(Z)$. $F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions. Operation within the limits given by the LOCA LHR figure in the COLR prevents power generation rates that would exceed the LOCA-limited LHR limits derived from the analysis of the ECCS.

The LOCA-limited LHR bounds the fuel centerline melt LHR limit. Thus, compliance with the LOCA-limited LHR ensures compliance with the fuel centerline melt LHR.

The DNBR-limited LHR is a specified acceptable fuel design limit that preserves the initial conditions for the limiting loss of flow transient. DNBR is defined as the ratio of the heat flux that would cause departure from nucleate boiling (DNB) at a particular core location to the actual heat flux at that core location. The DNBR-limited LHR represents the linear power generation rate along the fuel rod on which the minimum DNBR occurs. Compliance with this LHR value may be accomplished: 1) by correlating the LHR at the limiting location to the critical heat flux (expressed as a LHR) for the limiting location, 2) by correlating the LHR to DNBR or DNB margin for the limiting location, or 3) by correlating the LHR to a power peaking factor (designated as $F^N_{\Delta H}$) for the limiting location.

The relationship between the observable parameters of neutron power, reactor coolant flow, temperature and pressure and the critical heat flux, DNBR or DNB margin is provided through use of a critical heat flux correlation. The critical heat

BACKGROUND (continued)

flux correlations used to determine the critical heat flux for uniform and non-uniform heat flux distributions are described in the Bases for SL 2.1.1. $F^N_{\Delta H}$ is defined as the ratio of the integral of linear power along the fuel rod on which the minimum DNBR occurs to the average integrated rod power. Operation within the DNBR-limited LHR limit prevents DNB during a postulated loss of forced reactor coolant flow accident.

Measurement of the core core peaking factors using the Incore Detector System to obtain a three dimensional power distribution map provides direct confirmation that LHRs are within their limits and may be used to verify that the core local LHRs remain bounded when one or more normal operating parameters exceed their limits.

APPLICABLE SAFETY ANALYSES

The LOCA-limited LHR limits are determined by the ECCS analysis in order to limit peak cladding temperatures to 2200°F during a LOCA. The maximum acceptable cladding temperature is specified by 10 CFR 50.46 (Ref. 1). Higher cladding temperatures could cause severe cladding failure by oxidation due to a Zircaloy water reaction. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, peak cladding temperature is usually most limiting.

The DNBR-limited LHR limits provide protection from DNB during a limiting loss of flow transient. Proximity to the DNB condition is expressed by the DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux at that core location. The minimum DNBR value during both normal operation and anticipated transients is limited to the DNBR correlation limit for the particular fuel design in use, and is accepted as an appropriate margin to DNB. The DNBR correlation limit ensures that there is at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB. The critical heat flux correlations used to determine the critical heat flux for uniform and non-uniform heat flux distributions are described in the Bases for SL 2.1.1.

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break LOCA, peak cladding temperature must not exceed 2200°F (Ref. 1).

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition.

APPLICABLE SAFETY ANALYSES (continued)

The reload safety evaluation analysis determines limits on global core parameters that characterize the core power distribution. The primary parameters used to monitor and control the core power distribution are the regulating rod position, the APSR position, the AXIAL POWER IMBALANCE, and the QPT. These parameters are normally used to monitor and control the core power distribution because their measurements are continuously observable. Limits are placed on these parameters to ensure that the core power peaking factors remain bounded during operation in MODE 1 with THERMAL POWER greater than 20% RTP. Nuclear design model calculational uncertainty, manufacturing tolerances (e.g., the engineering hot channel factor), effects of fuel densification and rod bow, and modeling simplifications (such as treatment of the spacer grid effects) are accommodated as necessary through use of peaking augmentation factors in the reload safety evaluation analysis (Ref. 2).

LHR limitations satisfy Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

This LCO for power peaking ensures that the core operates within the LHR bounds assumed for the ECCS and thermal hydraulic analyses. Verification that LHR is within the limits of this LCO as specified in the COLR allows continued operation when the Required Actions of LCO 3.1.4, "CONTROL ROD Group Alignment Limits," LCO 3.2.1, "Regulating Rod Group Insertion Limits," LCO 3.2.2, "APSR Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT," are entered. Conservative THERMAL POWER reductions are required if the limits on LHR are exceeded. Verification that LHR is within the limits is also required during MODE 1 PHYSICS TESTS per LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 1."

Measurement uncertainties are applied when LHR is determined using the Incore Detector System. The measurement uncertainties applied to the measured values account for uncertainties in observability and instrument string signal processing.

APPLICABILITY

In MODE 1 with THERMAL POWER > 20% RTP, the limits on LHR must be maintained in order to prevent the core power distribution from exceeding the limits assumed in the analyses of the LOCA and loss of forced reactor coolant flow accidents. In MODE 1 with THERMAL POWER ≤ 20% RTP and in MODES 2, 3, 4, 5, and 6, this LCO is not applicable because the reactor has insufficient stored energy in the fuel or energy being transferred to the coolant to require a limit on the distribution of core power.

APPLICABILITY (continued)

The minimum THERMAL POWER level of 20% RTP was chosen based on the ability of the incore detection system to satisfactorily obtain meaningful power distribution data.

ACTIONS

The operator must take care in interpreting the relationship of the LHRs, DNBRs, and power peaking factors to their limits. Limiting values may be expressed as an LHR, DNBR, margin to DNB or as power peaking factors. When expressed as power peaking factors, the value must be adjusted in inverse proportion to the THERMAL POWER level of the core as the power is reduced from RTP. Thus, the allowable peaking factors will increase as THERMAL POWER decreases.

A.1

When the LHR is determined not to be within its specified limit as determined by a three dimensional power distribution map, a THERMAL POWER reduction is taken to reduce the limiting LHR in the core. The Completion Time of 2 hours provides an acceptable time to reduce power in an orderly manner and without allowing the unit to remain in an unacceptable condition for an extended period of time.

B.1

If the Required Action and associated Completion Time for Condition A are not met, then THERMAL POWER operation should be reduced. The reactor is placed in MODE 1 with THERMAL POWER less than or equal to 20% RTP where this LCO does not apply. The required Completion Time of 4 hours is a reasonable amount of time for the operator to reduce THERMAL POWER in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1

Core power distribution monitoring is performed using the Incore Detector System to obtain a three dimensional power distribution map. Maximum LHR values obtained from this map may then be compared with the limits in the COLR to verify that the limits have not been exceeded. Minimum DNBR values or DNB margins determined from the core power distribution mapping may also be compared to their limits or correlated to LHR values to verify that the limits have not been exceeded. Measurement of the core power distribution in this manner may be used to verify that the measured LHR values remain within their specified limits when one or more

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.5.1 (continued)

of the limits specified by LCO 3.1.4, LCO 3.1.6, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, or LCO 3.2.4 is exceeded, or when LCO 3.1.8 is applicable. If the local LHRs remain within their limits when one or more of these parameters exceed their limits, operation at THERMAL POWER may continue because the true initial conditions (the core power distribution) remain within their specified limits.

Because the limits on LHR are preserved when the parameters specified by LCO 3.1.4, LCO 3.1.6, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, or LCO 3.2.4 are within their limits, a Note is provided in the SR to indicate that monitoring core local LHRs is required only when complying with the Required Actions of these LCOs and when LCO 3.1.8 is applicable.

Frequencies for monitoring of the core local LHRs are specified in the Action statements of the individual LCOs. These Frequencies are reasonable based on the low probability of a limiting event occurring simultaneously with LHR exceeding its limit, and they provide sufficient time for the operator to obtain a power distribution map from the Incore Detector System. Indefinite THERMAL POWER operation in a Required Action of LCO 3.1.4, LCO 3.1.6, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, or LCO 3.2.4 is permitted, because the core local LHRs assumed in the accident analyses are within analyzed core power distributions and spatial xenon distributions.

REFERENCES

1. 10 CFR 50.46.

2. BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," Rev. 2, October 1997.

3. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Protection System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a reactor trip, if necessary, to protect against violating the core fuel design limits and the Reactor Coolant System (RCS) pressure boundary during abnormalities. By tripping the reactor, the RPS also assists the Engineered Safety Feature (ESF) Systems in mitigating accidents.

The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by identifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as the LCOs and administrative controls on other parameters and equipment performance.

The LSSS, defined in this Specification as the Allowable Value, in conjunction with the LCOs and administrative controls, establishes the threshold for protective system action to prevent exceeding specified acceptable limits during Design Basis Accidents (DBAs). Acceptable consequences for accidents are that the offsite dose shall be maintained within 10 CFR 100 limits or other limits approved by the NRC. During abnormalities, one or more of the following limits is maintained:

a. For accidents other than locked rotor, the departure from nucleate boiling ratio (DNBR) shall be maintained above the Safety Limit (SL) value. For the locked rotor accident, the minimum DNBR shall not be less than the applicable critical heat flux correlation limit, or fuel cladding shall be shown to experience no significant temperature excursions;

b. Fuel centerline temperature shall be maintained below the SL value;

c. The RCS pressure SL of 2750 psig shall not be exceeded; and

d. Reactor power shall not exceed 112% RTP.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 100 criteria during abnormalities.

<u>RPS Overview</u>

The RPS consists of four separate redundant protection channels that receive inputs of neutron flux, RCS pressure, RCS flow, reactor outlet temperature, RCS pump status, reactor building (RB) pressure, main feedwater (MFW) pump turbine status, and main turbine status.

BACKGROUND (continued)

Figure 7.1, SAR, Chapter 7 (Ref. 1), shows the arrangement of the RPS protection channels. A protection channel is composed of measurement channels, a manual trip channel, a reactor trip module (RTM), and control rod drive (CRD) trip devices. LCO 3.3.1 provides requirements for the individual measurement channels. These channels encompass all equipment and electronics from the point at which the measured parameter is sensed through the bistable relay contacts in the trip string. LCO 3.3.2, "Reactor Protection System (RPS) Manual Reactor Trip," LCO 3.3.3, "Reactor Protection System (RPS) - Reactor Trip Module (RTM)," and LCO 3.3.4, "Control Rod Drive (CRD) Trip Devices," discuss the remaining RPS elements.

The RPS instrumentation measures critical unit parameters and compares these to predetermined setpoints. If the setpoint is exceeded, a channel trip signal is generated. The generation of trip signals in any two of the four RPS channels will result in the trip of the reactor.

The Reactor Trip System (RTS) contains multiple CRD trip devices, two AC trip breakers, and two DC trip breaker pairs that provide a path for power to the CRD System. In addition to the safety rods, the power for the regulating rods and APSRs may be interrupted by the electronic trip assembly (ETA) relays. The system has two separate paths (or channels), with each path having either two breakers in series or a breaker and an ETA relay controlled silicon controlled rectifier (SCR) in series. Each path provides independent power to the CRDs. Either path can provide sufficient power to operate all CRDs. Two separate power paths to the CRDs ensure that a single failure that opens one path will not cause an unwanted reactor trip.

The RPS consists of four independent protection channels, each containing an RTM. Each RTM receives signals from its own measurement channels that indicate a protection channel trip is required. The RTM transmits this signal to its own two-out-of-four trip logic and to the two-out-of-four logic of the RTMs in the other three RPS channels. Whenever any two RPS channels trip, the RTM in each channel actuates to remove 120 VAC power from its associated CRD trip breaker.

The reactor is tripped by opening circuit breakers and de-energizing ETA relays that interrupt the control power supply to the CRDs. Six breakers are installed to increase reliability and allow testing of the trip system. A one-out-of-two taken twice logic is used to interrupt power to the rods.

The RPS has two manual bypasses: a shutdown bypass and a channel bypass. Shutdown bypass allows the withdrawal of safety rods to provide the availability of rapidly insertable negative reactivity during unit cooldowns or heatups. Channel bypass is typically used for maintenance and testing. Test circuits in the trip strings allow testing of RPS trip Functions. Also, an automatic bypass is provided at low power levels for the Main Turbine Trip and the Loss of Main Feedwater Pump Functions.

BACKGROUND (continued)

The RPS receives input from the instrumentation channels discussed next. The specific relationship between measurement channels and protection channels differs from parameter to parameter.

These arrangements and the relationship of instrumentation channels to trip Functions are discussed below to assist in understanding the overall effect of instrumentation channel failure.

<u>Power Range Nuclear Instrumentation</u>

Power Range Nuclear Instrumentation channels provide inputs to the following RPS trip Functions:

1. Nuclear Overpower

   a. Nuclear Overpower - High Setpoint;

   b. Nuclear Overpower - Low Setpoint;

7. Reactor Coolant Pump to Power;

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE (Power Imbalance Flow);

9. Main Turbine Trip (Oil Pressure); and

10. Loss of Main Feedwater Pumps (Control Oil Pressure).

The Main Turbine Trip and Loss of Main Feedwater Pumps Functions utilize the Power Range Nuclear Instrumentation only for enabling/disabling the operating bypass at low power levels.

The power range instrumentation has four linear channels, one for each core quadrant. Each channel feeds one RPS protection channel. Each channel originates in a detector assembly containing two uncompensated ion chambers. The ion chambers are positioned to represent the top half and bottom half of the core. The individual currents from the chambers are fed to individual linear amplifiers. The summation of the top and bottom is the total reactor power. The top minus the bottom neutron signal is the measured AXIAL POWER IMBALANCE of the reactor core.

<u>Reactor Outlet Temperature</u>

The Reactor Outlet Temperature provides input to the following Functions:

2. Reactor Outlet High Temperature; and

5. RCS Variable Low Pressure.

BACKGROUND (continued)

The Reactor Outlet Temperature is measured by two resistance elements in each hot leg, for a total of four. One temperature detector is associated with each protection channel.

Reactor Coolant System Pressure

The Reactor Coolant System Pressure provides input to the following Functions:

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure; and

11. Shutdown Bypass RCS High Pressure.

The RPS inputs of reactor coolant pressure are provided by two pressure transmitters in each hot leg, for a total of four. One sensor is associated with each protection channel.

Reactor Building Pressure

The Reactor Building Pressure measurements provide input only to the Reactor Building High Pressure trip, Function 6. There are four RB High Pressure sensors, one associated with each protection channel.

Reactor Coolant Pump Power Monitoring

Reactor coolant pump power monitors are inputs to the Reactor Coolant Pump to Power trip, Function 7. Each RCP's operating current is measured by a current transformer providing the current input to the associated RCP underpower relay, and the bus voltage is measured by a potential transformer providing the voltage input to the associated RCP underpower relays. Each RCP underpower relay provides individual RCP status to each protection channel.

Reactor Coolant System Flow

The Reactor Coolant System Flow measurements are an input to the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip, Function 8. The reactor coolant flow inputs to the RPS are provided by eight differential pressure transmitters, four on each loop, which measure flow through calibrated flow tubes. One flow input in each loop is associated with each protection channel.

BACKGROUND (continued)

Main Turbine Automatic Stop Oil Pressure

Main Turbine Automatic Stop Oil Pressure is an input to the Main Turbine Trip (Oil Pressure) reactor trip, Function 9. Each of the four protection channels receives turbine status information from one of four pressure switches monitoring main turbine automatic stop oil pressure. Contact buffers in each protection channel continuously monitor the status of the contact inputs and initiate an RPS trip when a main turbine trip is indicated.

Feedwater Pump Control Oil Pressure

Feedwater Pump Control Oil Pressure is an input to the Loss of Main Feedwater Pumps (Control Oil Pressure) trip, Function 10. Control oil pressure is measured by four switches on each feedwater pump. One switch on each pump is associated with each protection channel.

RPS Bypasses

The RPS is designed with two types of manual bypasses: channel bypass and shutdown bypass.

Channel bypass provides a method of placing all Functions in one RPS protection channel in a bypassed condition, and shutdown bypass provides a method of leaving the safety rods withdrawn during cooldown and depressurization of the RCS. Each bypass is discussed next.

Channel Bypass

A channel bypass provision is provided to allow for maintenance and testing of the RPS. The use of channel bypass keeps the protection channel trip relay energized regardless of the status of the instrumentation channel bistable relay contacts. To place a protection channel in channel bypass, the key switch must be operated, and the other three channels must not be in channel bypass. This is ensured by contacts from the other channels being in series with the channel bypass relay. If any contact is open, the second channel cannot be bypassed. When the bypass relay is energized, the bypass contact closes, maintaining the channel trip relay in an energized condition. An indicator light remains lit while the channel is in bypass. All RPS trips are reduced to a two-out-of-three logic in channel bypass. Only one channel bypass key is accessible for use in the control room.

Shutdown Bypass

During unit cooldown, it is allowable to leave some safety rods withdrawn to provide shutdown capabilities in the event of unusual positive reactivity additions (moderator dilution, etc.).

BACKGROUND (continued)

However, the unit is also depressurized as coolant temperature is decreased. If the safety rods are withdrawn and coolant pressure is decreased, an RCS Low Pressure trip will occur at 1800 psig and the rods will fall into the core. To avoid this, the protection system allows the operator to bypass the low pressure trip and maintain shutdown capabilities. During the cooldown and depressurization, the safety rods are inserted prior to the low pressure trip of 1800 psig. The RCS pressure is decreased to less than 1720 psig, then each RPS channel is placed in shutdown bypass.

When an RPS channel is placed in shutdown bypass, the RCS Low Pressure trip, Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip, Reactor Coolant Pump to Power trip, and the RCS Variable Low Pressure trip, are bypassed and a RCS High Pressure, ≤ 1720 psig trip and a Nuclear Overpower Low Setpoint trip, ≤ 5% RTP, are inserted. The operator can now withdraw the safety rods for additional rapidly insertable negative reactivity.

The insertion of the high pressure trip with a trip setpoint of ≤ 1720 psig prevents operation at normal system pressure, approximately 2155 psig, with a portion of the RPS bypassed, and ensures that the bypass is removed prior to normal operation. When the RCS pressure is increased during a unit heatup, the safety rods are inserted prior to reaching 1720 psig. The shutdown bypass is removed, which returns the RPS to normal, and system pressure is increased to greater than 1800 psig. All or some of the safety rods are then withdrawn and normally remain at the full out condition for the rest of the heatup.

The insertion of the Nuclear Overpower Low Setpoint Trip provides a backup to the Shutdown Bypass RCS High Pressure trip while preventing the generation of any significant amount of power.

Module Interlock and Test Trip Relay

Each channel and each trip module is capable of being individually tested. When a module is placed into the test mode, it causes the test trip relay to open and to indicate an RPS channel trip. Under normal conditions, the channel to be tested is placed in bypass before a module is tested.

Trip Setpoints/Allowable Value

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

The trip setpoints used in the bistables are based on the analytical limits used in the safety analysis described in SAR, Chapter 14 and Chapter 3A (Ref. 2). The selection of these trip setpoints is such that adequate protection is provided when appropriate sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and

BACKGROUND (continued)

environment errors, the Allowable Values specified in Table 3.3.1-1 are equal to or conservatively adjusted with respect to the analytical limits. Guidance used to calculate the uncertainty associated with the trip setpoints is provided in Instrument Loop Error Analysis and Setpoint Methodology Manual, Design Guide, IDG-001 (Ref. 3). The explicit uncertainties are addressed in the individual design calculations as required. The trip setpoint entered into the bistable may be more conservative than that specified by the Allowable Value to account for changes in instrument error detectable by a CHANNEL FUNCTIONAL TEST. A channel is inoperable if its as-found trip setpoint is not within its required Allowable Value.

Setpoints in accordance with the Allowable Value ensure that the limits of Chapter 2.0, "Safety Limits," in the Technical Specifications are not violated during abnormalities and that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the abnormality or DBA and the equipment functions as analyzed. Note that in LCO 3.3.1 the Allowable Values listed in Table 3.3.1-1 are the LSSS.

Each channel can be tested online to verify that the signal and trip setpoint are within the specified allowance requirements of approved calibration procedures. Once a designated channel is taken out of service for testing, a simulated signal may be injected in place of the field instrument signal. The process equipment for the channel may then be tested, verified, and calibrated.

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

Analyzed accidents and transients can be detected by one or more RPS Functions. The accident analysis contained in the SAR, Chapter 14 and Chapter 3A (Ref. 2), takes credit for most RPS trip Functions. Functions not specifically credited in the accident analysis were qualitatively credited in the NRC staff approved licensing basis for the unit. These Functions are high RB pressure, high RCS temperature, turbine trip, loss of main feedwater, the shutdown bypass nuclear overpower low setpoint, and shutdown bypass high pressure. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions also serve as backups to Functions that were credited in the safety analysis.

The LCO requires all instrumentation performing an RPS Function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable. The four channels of each Function in Table 3.3.1-1 of the RPS instrumentation shall be OPERABLE during its specified Applicability to ensure that a reactor trip will be actuated if needed. Additionally, during shutdown bypass with any CRD trip breaker closed, the applicable RPS Functions must also be OPERABLE. This ensures the capability to trip the withdrawn CONTROL RODS exists at all times that rod motion is possible. The trip Function channels specified in Table 3.3.1-1 are

APPLICABLE SAFETY ANALYSES (continued)

considered OPERABLE when all channel components necessary to provide a reactor trip are functional and in service for the required MODE or Other Specified Condition listed in Table 3.3.1-1.

Required Actions allow maintenance (protection channel) bypass of individual channels, but the bypass activates interlocks that prevent operation with a second channel bypass. Bypass effectively places the unit in a two-out-of-three logic configuration that can still initiate a reactor trip, even with a single failure within the system.

Only the Allowable Values are specified for each RPS trip Function in the LCO. Trip setpoints are specified in the setpoint calculations or calibration procedures. The setpoints are selected such that the setpoint measured by CHANNEL FUNCTIONAL TESTS is not expected to exceed the Allowable Value if the bistable is performing as required.

For most RPS Functions, the Allowable Value is to ensure that the departure from nucleate boiling (DNB) or RCS pressure SLs are not challenged. Cycle specific figures for use during operation are contained in the COLR.

Certain RPS trips function to indirectly protect the SLs by detecting specific conditions that do not immediately challenge SLs but will eventually lead to challenge if no action is taken. These trips function to minimize the consequences of unit transients caused by the specific conditions. The Allowable Value for these Functions is selected at the specified deviation from normal values that will indicate the condition, without risking spurious trips due to normal fluctuations in the measured parameter.

The Allowable Values for bypass removal Functions are stated in the Applicable MODE or Other Specified Condition column of Table 3.3.1-1.

The safety analyses applicable to each RPS Function are discussed next.

1. Nuclear Overpower

   a. Nuclear Overpower - High Setpoint

      The Nuclear Overpower - High Setpoint trip provides protection for the design thermal overpower condition based on the measured out of core fast neutron leakage flux.

      The Nuclear Overpower - High Setpoint trip initiates a reactor trip when the neutron power reaches a predefined setpoint at the design overpower limit. Because THERMAL POWER lags the neutron power, tripping when the neutron power reaches the design overpower will limit THERMAL POWER to a maximum value of the design overpower.

APPLICABLE SAFETY ANALYSES (continued)

Thus, the Nuclear Overpower - High Setpoint trip protects against violation of the DNBR and fuel centerline melt SLs. However, the RCS Variable Low Pressure, and Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE, also provide protection. The role of the Nuclear Overpower - High Setpoint trip is to limit reactor THERMAL POWER below the highest power at which the other two trips are known to provide protection.

The Nuclear Overpower - High Setpoint trip also provides transient protection for rapid positive reactivity excursions during power operations. These events include the rod withdrawal accident, the rod ejection accident, and the steam line break accident. By providing a trip during these events, the Nuclear Overpower - High Setpoint trip protects the unit from excessive power levels and also serves to reduce reactor power to prevent violation of the RCS pressure SL.

Rod withdrawal accident analyses cover a large spectrum of reactivity insertion rates (rod worths), which exhibit slow and rapid rates of power increases. At high reactivity insertion rates, the Nuclear Overpower - High Setpoint trip provides the primary protection. At low reactivity insertion rates, the high pressure trip provides primary protection.

The specified Allowable Value is selected to initiate a trip at or before reactor power exceeds the highest point at which the RCS Variable Low Pressure and the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trips are analyzed to provide protection against DNB and fuel centerline melt. The Allowable Value does not account for harsh environment induced errors, because the trip will actuate prior to degraded environmental conditions being reached.

b. <u>Nuclear Overpower - Low Setpoint</u>

While in shutdown bypass, the Nuclear Overpower - Low Setpoint is instated with a trip setpoint of $\leq$ 5% RTP. The low power setpoint, in conjunction with the Shutdown Bypass RCS High Pressure setpoint, protect the unit from excessive power conditions when other RPS trips are bypassed.

The Allowable Value was chosen to be as low as practical and still lie within the range of the out of core instrumentation.

2. <u>Reactor Outlet High Temperature</u>

The Reactor Outlet High Temperature trip, in conjunction with the RCS Low Pressure and RCS Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the reactor outlet temperature approaches the conditions necessary for DNB. Portions of each Reactor

APPLICABLE SAFETY ANALYSES (continued)

Outlet High Temperature trip channel are common with the RCS Variable Low Pressure trip. The Reactor Outlet High Temperature trip provides steady state protection for the DNBR SL.

The Reactor Outlet High Temperature trip limits the maximum RCS temperature to below the highest value for which DNB protection by the Variable Low Pressure trip is ensured. The trip setpoint Allowable Value is selected to initiate a trip before hot leg temperatures reach the point beyond which the RCS Low Pressure and Variable Low Pressure trips are analyzed. Above the high temperature trip, the variable low pressure trip need not provide protection, because the unit would have tripped already. The Allowable Value does not reflect errors induced by harsh environmental conditions that the equipment is expected to experience because the trip will actuate prior to degraded environmental conditions being reached.

3. RCS High Pressure

The RCS High Pressure trip works in conjunction with the pressurizer safety valves to prevent RCS overpressurization, thereby protecting the RCS High Pressure SL.

The RCS High Pressure trip has been credited in the accident analysis calculations for slow positive reactivity insertion transients (rod withdrawal accidents and moderator dilution). The rod withdrawal accidents cover a large spectrum of reactivity insertion rates and rod worths that exhibit slow and rapid rates of power increases. At high reactivity insertion rates, the Nuclear Overpower - High Setpoint trip provides the primary protection. At low reactivity insertion rates, the RCS High Pressure trip provides the primary protection.

The Allowable Value is selected such that the RCS High Pressure SL is not exceeded during steady state operation or slow power increasing transients. The Allowable Value does not reflect errors induced by harsh environmental conditions because the trip will actuate prior to degraded environmental conditions being reached.

4. RCS Low Pressure

The RCS Low Pressure trip, in conjunction with the Reactor Outlet High Temperature and Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated prior to reactor outlet temperature exceeding the conditions necessary for DNB. The RCS Low Pressure trip provides the DNB low pressure limit for the RCS Variable Low Pressure trip.

The RCS Low Pressure Allowable Value is selected to initiate a reactor trip before RCS pressure is reduced below the lowest point at which the RCS Variable Low Pressure trip is analyzed. The RCS Low Pressure trip provides protection for primary system depressurization events and has been credited in

APPLICABLE SAFETY ANALYSES (continued)

the accident analysis calculations for small break loss of coolant accidents (LOCAs). Consequently, harsh RB conditions created by small break LOCAs can affect performance of the RCS pressure sensors and transmitters. Therefore, degraded environmental conditions are considered in the Allowable Value determination.

5. <u>RCS Variable Low Pressure</u>

The RCS Variable Low Pressure trip, in conjunction with the Reactor Outlet High Temperature and RCS Low Pressure trips, provides protection for the DNBR SL. A trip is initiated prior to the system parameters of pressure and temperature exceeding the conditions necessary for DNB. The RCS Variable Low Pressure trip provides a floating low pressure trip based on the reactor outlet temperature expressed in degrees Fahrenheit within the range specified by the Reactor Outlet High Temperature and RCS Low Pressure trips.

The RCS Variable Low Pressure Allowable Value is selected to initiate a trip prior to temperature and pressure exceeding the conditions necessary for DNB while operating in a temperature pressure region constrained by the low pressure and high temperature trips. The RCS Variable Low Pressure trip is not assumed for transient protection in the unit safety analysis. Therefore, the Allowable Value does not account for errors induced by a harsh RB environment.

6. <u>Reactor Building High Pressure</u>

The Reactor Building High Pressure trip provides an early indication of a high energy line break (HELB) inside the RB. By detecting changes in the RB pressure, the RPS can provide a reactor trip before the other system parameters have varied significantly. Thus, this trip acts to minimize accident consequences.

The Allowable Value for RB High Pressure trip is set at the lowest value consistent with avoiding spurious trips during normal operation. Even in the case where this trip is a backup for other RPS trips for LOCA or MSLB, it is assumed to occur before degraded building conditions have an appreciable effect on RB High Pressure trip components. Therefore, determination of the Allowable Value does not account for errors induced by a harsh environment.

7. <u>Reactor Coolant Pump to Power</u>

The Reactor Coolant Pump to Power trip provides protection for changes in the reactor coolant flow due to the loss of multiple RCPs. Because the flow reduction lags loss of power indications due to the inertia of the RCPs, the trip initiates protective action earlier than a trip based on a measured flow signal.

APPLICABLE SAFETY ANALYSES (continued)

The trip also prevents operation with both pumps in either coolant loop tripped. Under these conditions, core flow and core fluid mixing may be insufficient for adequate heat transfer. Thus, the Reactor Coolant Pump to Power trip functions to protect the DNBR and fuel centerline temperature SLs.

The Reactor Coolant Pump to Power trip has been credited in the accident analysis calculations for the loss of four RCPs. The trip also provides protection for the loss of a pump or pumps which would result in both pumps in a single steam generator loop being tripped.

The Allowable Value for the Reactor Coolant Pump to Power trip setpoint is selected to prevent normal power operation unless at least one RCP is operating in each loop. RCP status is monitored by power transducers associated with each pump. These relays indicate a loss of an RCP on underpower. The underpower Allowable Value is selected to reliably trip on loss of voltage to the RCPs. Neither the reactor power nor the pump power Allowable Value account for instrumentation errors caused by harsh environments because the trip Function is not required to respond to events that could create harsh environments around the equipment.

8. <u>Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE</u>

The Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip provides steady state protection for the reactor core SLs. A reactor trip is initiated prior to the core power, AXIAL POWER IMBALANCE, and reactor coolant flow conditions exceeding the DNB or fuel centerline temperature limits.

This trip supplements the protection provided by the Reactor Coolant Pump to Power trip, through the power to flow ratio, for loss of reactor coolant flow events. The power to flow ratio provides direct protection for the limiting loss of flow transient which is the loss of two RCPs from four pump operation. The imbalance portion of the trip is credited for steady state protection only.

The power to flow ratio of the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip also provides steady state protection to prevent reactor power from exceeding the allowable power when the primary system is operating with two or three pump flow. Thus, the power to flow ratio prevents overpower conditions similar to the Nuclear Overpower trip. This protection ensures that during reduced flow conditions the core power is maintained below that required to begin DNB.

The Allowable Value is selected to ensure that a trip occurs prior to core power, axial power peaking, and reactor coolant flow conditions reaching DNB or fuel centerline temperature limits. The Allowable Value for this Function is given in the unit COLR because the cycle specific core peaking changes affect the Allowable Value.

APPLICABLE SAFETY ANALYSES (continued)

9. Main Turbine Trip (Oil Pressure)

The Main Turbine Trip Function trips the reactor when the main turbine is tripped at high power levels. The Main Turbine Trip Function provides an early reactor trip in anticipation of the loss of heat sink associated with a turbine trip. The Main Turbine Trip Function was added to the B&W designed units in accordance with NUREG-0737 (Ref. 4) following the Three Mile Island Unit 2 accident. The trip lowers the probability of an RCS electromatic relief valve (ERV) actuation for turbine trip cases.

Each of the four turbine oil pressure switches feeds one of the four protection channels through a buffer that continuously monitors the status of the contacts. Therefore, failure of any pressure switch affects only one protection channel.

For the Main Turbine Trip (Oil Pressure) bistable, the Allowable Value of ≥ 40.5 psig is selected to provide a trip whenever main turbine oil pressure drops below the normal operating range. The reactor power bypass is designed to automatically remove the turbine oil pressure trip function from the bypassed condition at < 45% RTP. Alarms are available to alert operators when the bypass function is enabled. Should the automatic bypass removal function fail such that the channel remains in the bypassed state, the channel must be considered inoperable at power levels of ≥ 45% RTP and the appropriate condition is entered. Failure of the automatic bypass removal feature alone or the inability to place the channel in a bypassed state when < 45% RTP does not constitute channel inoperability. The automatic bypass removal feature is tested to ensure its continued availability during the monthly CHANNEL FUNCTIONAL TEST. The turbine trip is not required to protect against events that can create a harsh environment in the turbine building. Therefore, errors induced by harsh environments are not included in the determination of the setpoint Allowable Value.

10. Loss of Main Feedwater Pumps (Control Oil Pressure)

The Loss of Main Feedwater Pumps (Control Oil Pressure) trip provides a reactor trip at high power levels when both MFW pumps are tripped. The trip provides an early reactor trip in anticipation of the loss of heat sink associated with a loss of main feedwater. This trip was added in accordance with NUREG-0737 (Ref. 4) following the Three Mile Island Unit 2 accident. This trip provides a reactor trip for a loss of main feedwater to minimize challenges to the ERV.

For the feedwater pump control oil pressure bistable, the Allowable Value of ≥ 55.5 psig is selected to provide a trip whenever feedwater pump control oil pressure drops below the normal operating range. The reactor power bypass is designed to automatically remove the main feedwater pump oil pressure trip function from the bypassed condition at < 10% RTP. Alarms are available to

APPLICABLE SAFETY ANALYSES (continued)

alert operators when the bypass function is enabled. Should the automatic bypass removal function fail such that the channel remains in the bypassed state, the channel must be considered inoperable at power levels of ≥ 10% RTP and the appropriate condition is entered. Failure of the automatic bypass removal feature alone or the inability to place the channel in a bypassed state when < 10% RTP does not constitute channel inoperability. The automatic bypass removal feature is tested to ensure its continued availability during the monthly CHANNEL FUNCTIONAL TEST. The Loss of Main Feedwater Pumps (Control Oil Pressure) trip is not required to protect against events that can create a harsh environment in the turbine building. Therefore, errors caused by harsh environments are not included in the determination of the setpoint Allowable Value.

11. Shutdown Bypass RCS High Pressure

The RPS Shutdown Bypass is provided to allow for withdrawing the CONTROL RODS while operating below the normal RCS Low Pressure trip setpoint. The shutdown bypass allows the operator to withdraw the safety groups of CONTROL RODS. This makes their negative reactivity available to terminate inadvertent reactivity excursions. Because the shutdown bypass high pressure trip setpoint is below the normal RCS low pressure trip setpoint, the reactor must be tripped while passing between these two setpoints. This ensures that RPS trips cannot be bypassed unless the CONTROL RODS are all inserted.

Accidents analyzed in the SAR, Chapter 14 and Chapter 3A (Ref. 2), do not include events that occur during shutdown bypass operation.

During shutdown bypass operation with the Shutdown Bypass RCS High Pressure trip active with a setpoint of ≤ 1720 psig and the Nuclear Overpower - Low Setpoint active with a setpoint of ≤ 5% RTP, the trips listed below are bypassed.

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE.

The Shutdown Bypass Nuclear Overpower - Low Setpoint Allowable Value is selected to initiate a trip before producing significant THERMAL POWER.

APPLICABLE SAFETY ANALYSES (continued)

General Discussion

In MODES 1 and 2, the RPS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 5). In MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal, the RPS satisfies Criterion 4 of 10 CFR 50.36.

In MODE 1; in MODE 2, when not operating in shutdown bypass; and in MODE 3, when not operating in shutdown bypass but with any CRD trip breaker in the closed position and the CRD system capable of rod withdrawal, the following trips are required to be OPERABLE. These trips function to ensure that any withdrawn CONTROL RODS can be automatically inserted to make or maintain the reactor subcritical.

1.a. Nuclear Overpower-High Setpoint; and

3. RCS High Pressure.

In MODES 1 and 2, the following trips are required to the OPERABLE. These trips function as primary or as back-up trips to ensure that any withdrawn CONTROL RODS can be automatically inserted to make or maintain the reactor subcritical.

2. Reactor Outlet High Temperature; and

6. Reactor Building High Pressure.

In addition, Function 6, Reactor Building High Pressure, is required to be OPERABLE in MODE 3, whenever any CRD trip breaker is closed and the CRD system is capable of rod withdrawal. In this MODE, this Function serves purely as a back-up to other required Functions.

In MODE 1 and in MODE 2, when not in shutdown bypass operation, the following trips are required to be OPERABLE. These Functions operate to ensure that any withdrawn CONTROL RODS can be automatically inserted to make or maintain the reactor subcritical. These functions are all bypassed when the channel is placed in a shutdown bypass condition. Therefore, they are not required to be OPERABLE during shutdown bypass operation.

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE.

Two Functions are required to be OPERABLE only during portions of MODE 1. These are the Main Turbine Trip (Oil Pressure) and the Loss of Main Feedwater Pumps (Control Oil Pressure) trip. These Functions are required to be OPERABLE

APPLICABLE SAFETY ANALYSES (continued)

at ≥ 45% RTP and ≥ 10% RTP, respectively. Analyses presented in BAW-1893 (Ref. 6) have shown that for operation below these power levels, these trips are not necessary to minimize challenges to the ERV as required by NUREG-0737 (Ref. 4).

Because the safety function of the RPS is to trip the CONTROL RODS, the RPS is not required to be OPERABLE in MODE 3, 4, or 5, if either the reactor trip breakers are open, or the CRD System is incapable of rod withdrawal. Similarly, the RPS is not required to be OPERABLE in MODE 6 because the CONTROL RODS are normally decoupled from the CRDs.

However, during shutdown bypass operation, in MODE 2, 3, 4, or 5, the Shutdown Bypass RCS High Pressure and Nuclear Overpower - Low setpoint trips are required to be OPERABLE if the CRD trip breakers are closed and the CRD System is capable of rod withdrawal. Under these conditions, the Shutdown Bypass RCS High Pressure and Nuclear Overpower - Low setpoint trips sufficiently reduce the potential for conditions that could challenge SLs.

ACTIONS

Conditions A, B, and C are applicable to all RPS protection Functions. If a channel's trip setpoint is found nonconservative with respect to the required Allowable Value in Table 3.3.1-1, or the transmitter, instrument loop, signal processing electronics or bistable is found inoperable, the channel must be declared inoperable and all applicable Conditions entered immediately.

A.1 and A.2

If one or more Functions in one protection channel become inoperable, the affected protection channel must be placed in bypass or trip, or the bypass of the remaining channels prevented. If the channel is bypassed, all RPS Functions are placed in a two-out-of-three logic configuration and the bypass of any other channel is prevented. In this configuration, the RPS can still perform its safety function in the presence of a random failure of any single channel. Alternatively, the inoperable channel can be placed in trip. Tripping the affected protection channel places all RPS Functions in a one-out-of-three configuration.

Another option is to maintain the channel, which contains one or more inoperable Functions, in an untripped and unbypassed state. In this case, bypass of the remaining three channels must be prevented. This is accomplished by tagging them, under administrative controls, to prevent their being bypassed. This option assumes that the inoperability of the Function(s) does not require the channel containing the inoperable Function(s) to remain in a tripped condition, and that the channel contains other Functions which remain OPERABLE.

ACTIONS (continued)

A.1 and A.2 (continued)

By maintaining the channel in an untripped and unbypassed state, the inoperable Function (s) are in a two-out-of-three logic configuration. This configuration is equivalent to bypassing the channel. However, by maintaining the channel in an untripped and unbypassed condition, the OPERABLE Functions within that channel remain in service in a normal two-out-of-four logic configuration.

Operation in these configurations may continue indefinitely because the RPS is capable of performing its trip Function in the presence of any single random failure. The 1 hour Completion Time is sufficient to perform Required Action A.1 or Required Action A.2.

B.1, B.2.1, and B.2.2

For Required Action B.1 and Required Action B.2, if one or more Functions in two protection channels become inoperable, one of two inoperable protection channels must be placed in trip. The second inoperable channel may be bypassed or may be maintained in an untripped and unbypassed condition. If the channel is not bypassed, bypass of the remaining channels must be prevented. This is accomplished by tagging them, under administrative controls, to prevent their being bypassed. This option assumes that the inoperability of the Function(s) in the second channel does not require that channel to remain in a tripped condition, and that the channel contains one or more Function(s) which remains OPERABLE. These Required Actions place all RPS Functions in either a one-out-of-two or one-out-of-three logic configuration. In either of these configurations, the RPS can still perform its safety functions in the presence of a random failure of any single channel. The 1 hour Completion Time is sufficient time to perform Required Action B.1, Required Action B.2.1, and Required Action B.2.2.

C.1

Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.1-1. The applicable Condition referenced in the table is Function dependent. If the Required Action and associated Completion Time of Condition A or B are not met or if more than two channels are inoperable, Condition C is entered to provide for transfer to the appropriate subsequent Condition.

D.1 and D.2

If Required Action C.1 and Table 3.3.1-1 direct entry into Condition D, the unit must be brought to a MODE in which the specified RPS trip Functions are not required to be OPERABLE. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and to open all CRD trip breakers without challenging unit systems.

ACTIONS (continued)

E.1

If Required Action C.1 and Table 3.3.1-1 direct entry into Condition E, the unit must be brought to a MODE in which the specified RPS trip Functions are not required to be OPERABLE. To achieve this status, all CRD trip breakers must be opened. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open CRD trip breakers without challenging unit systems.

F.1

If Required Action C.1 and Table 3.3.1-1 direct entry into Condition F, the unit must be brought to a MODE in which the specified RPS trip Function is not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced to < 45% RTP. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach 45% RTP from full power conditions in an orderly manner without challenging unit systems.

G.1

If Required Action C.1 and Table 3.3.1-1 direct entry into Condition G, the unit must be brought to a MODE in which the specified RPS trip Function is not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced to < 10% RTP. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach 10% RTP from full power conditions in an orderly manner without challenging unit systems.

SURVEILLANCE REQUIREMENTS

The SRs for each RPS Function are identified by the SRs column of Table 3.3.1-1 for that Function. Most Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION testing.

The SRs are modified by a Note which directs the reader to Table 3.3.1-1 to determine the correct SRs to perform for each RPS Function.

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours provides reasonable assurance of prompt identification of a gross failure of instrumentation. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to the same 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 two instrument channels could be

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1 (continued)

an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of factors including channel instrument uncertainties. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are, where practical, verified to be reading at the bottom of the range and not failed downscale.

The agreement criteria includes an expectation of one decade of overlap when transitioning between neutron flux instrumentation. For example, during a power increase near the top of the scale for the intermediate range monitors, a power range monitor reading is expected with at least one decade overlap. Without such an overlap, the power range monitors are considered inoperable unless it is clear that an intermediate range monitor inoperability is responsible for the lack of the expected overlap.

The Frequency is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal but more frequent checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels.

For Functions that trip on a combination of several measurements, such as the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE Function, the CHANNEL CHECK must be performed on each input.

SR 3.3.1.2

This SR is the performance of a heat balance calibration for the power range channels every 96 hours and once within 24 hours after a THERMAL POWER change of ≥ 10% RTP in one direction, when reactor power is ≥ 20% RTP. The heat balance calibration consists of a comparison of the results of the calorimetric with the power range channel output. The outputs of the power range channels are calibrated to the calorimetric. Note 1 to the SR states if the absolute difference between the calorimetric and the Nuclear Instrumentation System (NIS) channel is

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.2 (continued)

> 2% RTP, the NIS channel is not declared inoperable but must be adjusted. If the NIS channel cannot be properly adjusted, the channel is declared inoperable. Note 2 clarifies that this Surveillance is required only if reactor power is ≥ 20% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 20% RTP.

Two calorimetric calculations are routinely performed. One relies upon primary system parameters and the other relies upon secondary system parameters. The primary calorimetric is generally less accurate than the secondary calorimetric at higher power levels and more accurate at lower power levels. For comparison to the nuclear instrumentation, between 0 and 15% power, only the primary calorimetric (heat balance) is considered. From 15 to 100% power the calorimetric is weighted linearly with only the secondary heat balance being considered at 100% power.

The power range channel's output shall be adjusted consistent with the calorimetric results if the absolute difference between the calorimetric and the power range channel's output is > 2% RTP. The value of 2% is adequate because this value is assumed in the safety analyses of SAR, Chapter 14 (Ref. 2). These checks and, if necessary, the adjustment of the power range channels ensure that channel accuracy is maintained within the analyzed error margins. The 96 hour Frequency is adequate, based on unit operating experience, which demonstrates the change in the difference between the power range indication and the calorimetric results rarely exceeds 2% in any 96 hour period. Furthermore, the control room operators monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

A comparison of power range nuclear instrumentation channels against incore detectors shall be performed at a 31 day Frequency when reactor power is ≥ 20% RTP. The SR is modified by two Notes. Note 2 clarifies that 24 hours is allowed for performing the first Surveillance after reaching 20% RTP. Note 1 states if the absolute difference between the power range and incore AXIAL POWER IMBALANCE measurements is ≥ 2% RTP, the power range channel is not inoperable, but an adjustment of the measured imbalance to agree with the incore measurements is necessary. If the power range channel cannot be properly recalibrated, the channel is declared inoperable. The calculation of the Allowable Value envelope assumes a difference in out of core to incore AXIAL POWER IMBALANCE measurements of 2.5%. Additional inaccuracies beyond those that are measured are also included in the setpoint envelope calculation. The 31 day Frequency is adequate, considering that long term drift of the excore linear amplifiers is small and burnup of the detectors is slow. Also, the excore readings are a strong function of the power produced in the peripheral fuel bundles, and do not represent an integrated reading across the core. The slow changes in neutron flux during the fuel cycle can also be detected at this interval.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.4

A CHANNEL FUNCTIONAL TEST is performed to ensure that the entire channel will perform the intended function. Setpoints must be found within the Allowable Values specified in Table 3.3.1-1. Any setpoint adjustment shall be consistent with the assumptions of the current setpoint analysis.

The Frequency of 31 days is based on operating experience, which has demonstrated through high reliability of the instrumentation, that failure of more than one channel, of a given Function, in any 31 day interval is rare.

Testing in accordance with this SR is normally performed on a rotational basis, with one channel being tested each week. Testing one channel each week reduces the probability of an undetected failure existing within the system and minimizes the likelihood of the same systematic test errors being introduced into each redundant channel. The automatic bypass removal feature is verified for the turbine oil pressure trip and the main feedwater pump oil pressure trip functions during the CHANNEL FUNCTIONAL TEST.

SR 3.3.1.5

A Note to the Surveillance indicates that neutron detectors are excluded from CHANNEL CALIBRATION. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that instrument errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATION must be performed consistent with the assumptions of the setpoint analysis. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature (RTD) sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

The Frequency is justified by the assumption of at least an 18 month calibration interval in the determination of the allowable magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. SAR, Chapter 7.
2. SAR, Chapter 14 and Chapter 3A.
3. Instrument Loop Error Analysis and Setpoint Methodology Manual, Design Guide, IDG-001.
4. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.
5. 10 CFR 50.36.
6. BAW-1893, "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," October 1985.

B 3.3 INSTRUMENTATION

B 3.3.2 Reactor Protection System (RPS) Manual Reactor Trip

BASES

BACKGROUND

The RPS Manual Reactor Trip provides the operator with the capability to trip the reactor from the control room in the absence of, or coincident with, any other trip condition. Manual trip is provided by a trip push button on the main control board. This push button operates four electrically independent switches. This trip is independent of the automatic trip system. As shown in Figure 7.1, SAR, Chapter 7 (Ref. 1), control power for the control rod drive (CRD) breakers and electronic trip assembly (ETA) relays comes from the reactor trip modules (RTMs). The manual trip switches are located between the RTM output and the breaker undervoltage coils, breaker undervoltage relays, and ETA relays. The switches also initiate actuation of the breaker shunt trip mechanisms. These are separate switches which are actuated through a mechanical linkage from a single push button. Opening of the switches opens the circuits to the breakers, tripping them.

APPLICABLE SAFETY ANALYSES

The Manual Reactor Trip ensures that the control room operator can initiate a reactor trip at any time. The Manual Reactor Trip Function is required as a backup to the automatic trip functions.

Operating experience has shown the Manual Reactor Trip Function to be significant to public health and safety, and therefore satisfy Criterion 4 of 10 CFR 50.36 (Ref. 2).

LCO

The LCO on the RPS Manual Reactor Trip requires that the trip shall be OPERABLE whenever the reactor is critical or any time any CRD breaker is closed and rods are capable of being withdrawn, including shutdown bypass. This enables the operator to terminate any reactivity excursion that in the operator's judgment requires protective action, even if no automatic trip condition exists.

The Manual Reactor Trip Function is composed of four electrically independent trip switches sharing a common mechanical push button.

APPLICABILITY

The Manual Reactor Trip Function is required to be OPERABLE in MODES 1 and 2. It is also required to be OPERABLE in MODES 3, 4, and 5 if any CRD trip breaker is in the closed position and if the CRD System is capable of rod withdrawal. The primary safety function of the RPS is to trip the CONTROL RODS; therefore, the Manual Reactor Trip Function is not needed in MODE 3, 4, or 5 if the reactor trip breakers are open or if the CRD System is incapable of rod withdrawal. Similarly, the RPS Manual Reactor Trip is not needed in MODE 6 because the CONTROL RODS are normally decoupled from the CRDs.

ACTIONS

A.1

Condition A applies when the Manual Reactor Trip Function is found inoperable. One hour is allowed to restore the Function to OPERABLE status. The automatic functions and various alternative manual trip methods, such as removing power to the RTMs, are still available. The 1 hour Completion Time is sufficient time to correct minor problems.

B.1 and B.2

If the Required Action and associated Completion Time are not met in MODE 1, 2, or 3, the unit must be placed in a MODE in which manual trip is not required. Required Action B.1 and Required Action B.2 place the unit in at least MODE 3 with all CRD trip breakers open within 6 hours. The allowed 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 unit systems.

C.1

If the Required Action and associated Completion Time are not met in MODE 4 or 5, the unit must be placed in a MODE in which manual trip is not required. To achieve this status, all CRD trip breakers must be opened. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the Manual Reactor Trip Function. This test verifies the OPERABILITY of the Manual Reactor Trip by actuation of the CRD trip breakers. The Frequency shall be once prior to each reactor startup if not performed within the preceding 7 days to ensure the OPERABILITY of the Manual Reactor Trip Function prior to achieving criticality. The Frequency was developed in consideration that this Surveillance is only performed during a unit outage.

REFERENCES

1. SAR, Chapter 7.
2. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.3 Reactor Protection System (RPS) - Reactor Trip Module (RTM)

BASES

BACKGROUND

The RPS consists of four independent protection channels, each containing an RTM. Figure 7.1, SAR, Chapter 7 (Ref. 1), shows a typical RPS protection channel and the relationship of the RTM to the RPS instrumentation, manual trip, and control rod drive (CRD) trip devices. The RTM receives bistable trip signals from the functions in its own channel and channel trip signals from the other three RPS - RTMs. The RTM provides these signals to its own two-out-of-four trip logic and transmits its own channel trip signal to the two-out-of-four logic of the RTMs in the other three RPS channels. Whenever any two RPS channels transmit channel trip signals, the RTM logic in each channel actuates to remove 120 VAC power from its associated CRD trip device.

The RPS trip scheme consists of series contacts that are operated by bistables. During normal unit operations, all contacts are closed and the RTM channel trip relay remains energized. However, if any trip parameter exceeds its setpoint, its associated contact opens, which de-energizes the channel trip relay.

When an RTM channel trip relay de-energizes, several things occur:

a. Each of the four (4) output logic relays "informs" its associated RPS channel that a reactor trip signal has occurred in the tripped RPS channel;

b. The contacts in the trip device circuitry, powered by the tripped channel, open, but the trip device remains energized through the closed contacts from the other RTMs. (This condition exists in each RPS - RTM. Each RPS - RTM controls power to a trip device.); and

c. The contact in parallel with the channel reset switch opens and the trip is sealed in. To re-energize the channel trip relay, the channel reset switch must be depressed after the trip condition has cleared.

When the second RPS channel senses a reactor trip condition, the output logic relays for the second channel de-energize and open contacts that supply power to the trip devices. With contacts opened by two separate RPS channels, power to the trip devices is interrupted and the CONTROL RODS fall into the core.

A minimum of two out of four RTMs must sense a trip condition to cause a reactor trip. Also, two channel trips caused by different trip functions can result in a reactor trip.

APPLICABLE SAFETY ANALYSES

Accident analyses rely on a reactor trip for protection of reactor core integrity, reactor coolant pressure boundary integrity, and reactor building OPERABILITY. A reactor trip must occur when needed to prevent accident conditions from exceeding those calculated in the accident analyses. More detailed descriptions of the applicable accident analyses are found in the bases for each of the RPS trip Functions in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation."

In MODES 1 and 2, the RTMs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2). In MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal, the RTMs satisfy Criterion 4 of 10 CFR 50.36.

LCO

The RTM LCO requires all four RTMs to be OPERABLE. Failure of any RTM renders a portion of the RPS inoperable.

To be considered OPERABLE, an RTM must be able to receive and interpret trip signals from its own and other OPERABLE RPS channels and to open its associated trip device.

The requirement for four channels to be OPERABLE ensures that a minimum of two RPS channels will remain OPERABLE if a single failure has occurred in one channel and if a second channel has been bypassed. This two-out-of-four trip logic also ensures that a single RPS channel failure will not cause an unwanted reactor trip. Violation of this LCO could result in a trip signal not causing a reactor trip when needed.

APPLICABILITY

The RTMs are required to be OPERABLE in MODES 1 and 2. They are also required to be OPERABLE in MODES 3, 4, and 5 if any CRD trip breakers are in the closed position and the CRD System is capable of rod withdrawal. The RTMs are designed to ensure a reactor trip would occur, if needed. This need may exist in any of these MODES; therefore, the RTMs must be OPERABLE.

ACTIONS

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

When an RTM is inoperable, the associated CRD trip breaker must then be placed in a condition that is equivalent to a tripped condition for the RTM. Required Action A.1.1 or Required Action A.1.2 requires this either by opening (tripping) the

ACTIONS (continued)

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

CRD trip breaker or by removing power to the CRD trip device. Tripping one RTM or removing power opens one set of CRD trip devices. Power to hold up CONTROL RODS is still provided via the parallel CRD trip device(s). Therefore, a reactor trip will not occur until a second protection channel trips.

To ensure the trip signal is registered in the other channels, Required Action A.2 requires that the inoperable RTM be removed from the cabinet. This action causes the electrical interlocks to indicate a tripped channel in the remaining three RTMs. Operation in this condition is allowed indefinitely because the actions put the RPS into a one-out-of-three configuration. The 1 hour Completion Time is sufficient time to perform the Required Actions.

B.1, B.2.1, and B.2.2

Condition B applies if two or more RTMs are inoperable in MODE 1, 2, or 3, or if the Required Actions and associated Completion Time of Condition A are not met in MODE 1, 2, or 3. In this case, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 with all CRD trip breakers open or with power to all CRD trip breakers removed within 6 hours. The allowed 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 unit systems.

C.1 and C.2

Condition C applies if two or more RTMs are inoperable in MODE 4 or 5, or if the Required Actions and associated Completion Times are not met in MODE 4 or 5. In this case, the unit must be placed in a MODE in which the LCO does not apply. This is done by opening all CRD trip breakers or removing power from all CRD trip breakers. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers or remove all power to the CRD System without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1

The SRs include performance of a CHANNEL FUNCTIONAL TEST every 92 days. This test shall verify the OPERABILITY of the RTM and its ability to receive and properly respond to channel trip and reactor trip signals.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.1 (continued)

The Frequency of 92 days is based on operating experience, which has demonstrated through high reliability of the instrumentation, that failure of more than one RTM in any 92 day interval is rare (Ref. 3).

Testing in accordance with this SR is normally performed on a rotational basis, with one RTM being tested each 23 days. Testing one RTM each 23 days reduces the probability of an undetected failure existing within the system and minimizes the likelihood of the same systematic test errors being introduced into each redundant RTM.

REFERENCES

1. SAR, Chapter 7.
2. 10 CFR 50.36.
3. BAW-10167A, "Justification for Increasing the Reactor Trip System On-Line Test Intervals," Supplement 3, "Justification for Increasing the Trip Device Test Interval," February 1998.

B 3.3 INSTRUMENTATION

B 3.3.4 Control Rod Drive (CRD) Trip Devices

BASES

BACKGROUND

The Reactor Protection System (RPS) contains multiple CRD trip devices: two AC trip breakers, two DC trip breaker pairs, and ten electronic trip assembly (ETA) relays. The system has two separate paths (or channels), with each path having one AC breaker either in series with a pair of DC breakers or functionally in series with five ETA relays in parallel. Each path provides independent power to the CRDs. Either path can provide sufficient power to operate the entire CRD System.

Figure 7-10 SAR, Chapter 7 (Ref. 1), illustrates the configuration of CRD trip devices. To trip the reactor, power to the CRDs must be removed. Loss of power causes the CRD's mechanisms to release the CONTROL RODS, which then fall by gravity into the core.

Power to CRDs is supplied from two separate unit sources through the AC trip circuit breakers. These breakers are designated A and B, and their undervoltage and shunt trip coils are controlled by RPS channels A and B, respectively. From the circuit breakers, the CRD power travels through voltage regulators and stepdown transformers. These devices in turn supply redundant buses that feed the DC holding power supplies and the regulating rod, APSR and auxiliary power supplies.

The DC holding power supplies rectify the AC input and supply power to hold the safety rods in their fully withdrawn position. One of the redundant power sources supplies phase A; the other, phase CC. Either phase being energized is sufficient to hold the rod. Two breakers are located on the output of each power supply. Each breaker controls half of the power to two of the four safety rod groups. The undervoltage and shunt trip coils on the two circuit breakers on the output of one of the power supplies is controlled by RPS channel C. The other two breakers are controlled by RPS channel D.

In addition to the DC holding power supplies, the redundant buses also supply power to the regulating rod, APSR and auxiliary power supplies. These power supplies contain silicon controlled rectifiers (SCRs), which are gated on and off to provide power to, and remove power from, the phases of the CRD mechanisms. The gating control signal for these SCRs is supplied through the closed contacts of the ETA relays. These contacts are referred to as E and F contactors, and are controlled by the C and D RPS channels, respectively.

The AC breaker and DC breakers, or gated SCRs, are in series in one of the power supplies; whereas, the redundant AC breaker and DC breakers or gated SCRs are in series in the other power supply to the CONTROL RODS. The logic required to

BACKGROUND (continued)

cause a reactor trip is the opening of a circuit breaker or ETA relay in each of the redundant power supplies. (The pair of DC circuit breakers on the output of the power supply are treated as one breaker.) This is known as a one-out-of-two taken twice logic. The following examples illustrate the operation of the reactor trip circuit breakers.

a. If the A AC circuit breaker opens:

   1. the input power to associated DC power supply is lost, and

   2. the SCR supply from the associated power source is lost.

b. If the D DC circuit breaker(s) and F contactors open:

   1. the output of the redundant DC power supply is lost and the safety rods de-energize, and

   2. when the F contactor opens, SCR gating power is lost and the regulating rods will be de-energized.

c. The combination of (a) and (b) causes a reactor trip.

Any other combination of at least one circuit breaker opening in each power supply will cause a reactor trip.

In summary, two tripped RPS channels will cause a reactor trip. For example, a reactor trip occurs if RPS channel B senses a low Reactor Coolant System (RCS) pressure condition and if RPS channel C senses a variable low RCS pressure condition. When the channel B bistable relay de-energizes, the channel trip relay de-energizes and opens its associated contacts. The same thing occurs in channel C, except the variable low pressure bistable relay de-energizes the channel C trip relay. When the output logic relays in channels B and C de-energize, the B and C contacts in the trip logic of each channel's reactor trip module (RTM) open causing an undervoltage to each trip breaker. All trip breakers and the ETA relay contactors open, and power is removed from all CRD mechanisms. All rods fall into the core, resulting in a reactor trip.

APPLICABLE SAFETY ANALYSES

Accident analyses rely on a reactor trip for protection of reactor core integrity, reactor coolant pressure boundary integrity, and reactor building OPERABILITY. A reactor trip must occur when needed to prevent accident consequences from exceeding those calculated in the accident analyses. The control rod insertion limits ensure that adequate rod worth is available upon reactor trip to shut down the reactor to the required SDM. Further, OPERABILITY of the CRD trip devices

APPLICABLE SAFETY ANALYSES (continued)

ensures that all CONTROL RODS will trip when required. More detailed descriptions of the applicable accident analyses are found in the Bases for each of the individual RPS trip Functions in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation."

In MODES 1 and 2, the CRD trip devices satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2). In MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal, the CRD trip devices satisfy Criterion 4 of 10 CFR 50.36.

LCO

The LCO requires all of the specified CRD trip devices to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when any CRD trip breaker is in the closed position and the CRD System is capable of rod withdrawal. Failure of any required CRD trip device renders a portion of the RPS inoperable. Without reliable CRD reactor trip circuit breakers and associated support circuitry, a reactor trip may not occur when initiated either automatically or manually.

All required CRD trip devices shall be OPERABLE to ensure that the reactor remains capable of being tripped any time it is critical. OPERABILITY is defined as the CRD trip device being able to receive a reactor trip signal and to respond to this trip signal by interrupting power to the CRDs. Both of a CRD trip breaker's diverse trip devices and the breaker itself must be functioning properly for the breaker to be OPERABLE.

Both ETA relays associated with each of the three regulating rod groups and the two ETA relays associated with the auxiliary power supply must be OPERABLE to satisfy the LCO. The ETA relays associated with the APSR power supply are not required to be OPERABLE because the APSRs are not designed to fall into the core upon initiation of a reactor trip.

Requiring all breakers and ETA relays to be OPERABLE ensures that at least one device in each of the two power paths to the CRDs will remain OPERABLE even with a single failure.

APPLICABILITY

The CRD trip devices are required to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when any CRD trip breaker is in the closed position and the CRD System is capable of rod withdrawal.

The CRD trip devices are designed to ensure that a reactor trip would occur if needed. Since a trip may be required in all of these MODES, the CRD trip devices must be OPERABLE.

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each CRD trip device.

A.1 and A.2

Condition A represents reduced redundancy in the CRD trip Function. Condition A applies when:

- One diverse trip Function (undervoltage or shunt trip device) is inoperable in one or more CRD trip breaker(s) or breaker pair; or
- One diverse trip Function is inoperable in both DC trip breakers associated with one protection channel. In this case, the inoperable trip Function does not need to be the same for both breakers.

If one of the diverse trip Functions on a CRD trip breaker or breaker pair becomes inoperable, actions must be taken to preclude the inoperable CRD trip device from preventing a reactor trip when needed. This is done by manually opening the inoperable CRD trip breaker or by removing power from the inoperable CRD trip breaker. Either of these actions places the affected CRDs in a one-out-of-two trip configuration, which precludes a single failure, which in turn could prevent tripping of the reactor. The 48 hour Completion Time has been shown to be acceptable through operating experience.

B.1 and B.2

Condition B represents a loss of redundancy for the CRD trip Function. Condition B applies when both diverse trip Functions are inoperable in one or more trip breaker(s) or breaker pairs.

Required Action B.1 and Required Action B.2 are the same as Required Action A.1 and Required Action A.2, but the Completion Time is shortened. The 1 hour Completion Time allowed to open or remove power from the CRD trip breaker allows the operator to take all the appropriate actions for the inoperable breaker and still ensures that the risk involved is acceptable.

C.1, C.2, C.3, and C.4

Condition C represents a loss of redundancy for the CRD trip Function. Condition C applies when one or more ETA relays are inoperable. The preferred action is to restore the ETA relay to OPERABLE status. If this cannot be done, the operator can perform one of four actions to eliminate reliance on the failed ETA relay. The first option is to switch the affected CONTROL ROD group to an alternate power supply which has two OPERABLE or one OPERABLE and one open ETA relay.

ACTIONS (continued)

C.1, C.2, C.3, and C.4 (continued)

This removes the failed ETA relay from the trip sequence, and the unit can operate indefinitely. The second option is to transfer the affected CONTROL ROD group to a DC holding power supply. This option is only available if the affected group is a safety rod group and the affected power supply is the auxiliary power supply. The third option is to open the inoperable ETA contacts. This option results in the safety function being performed. The fourth option is to open the corresponding AC CRD trip breaker. This also results in the safety function being performed, thereby eliminating the failed ETA relay from the trip sequence.

The 1 hour Completion Time is sufficient to perform the Required Action.

D.1, D.2.1, and D.2.2

If the Required Actions and associated Completion Times of Condition A, B, or C are not met in MODE 1, 2, or 3, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3, with all CRD trip breakers open or with power to all CRD trip breakers removed within 6 hours. The allowed 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 unit systems.

E.1 and E.2

If the Required Actions and associated Completion Times of Condition A, B, or C are not met in MODE 4 or 5, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, all CRD trip breakers must be opened or power to all CRD trip breakers removed within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers or remove power from all CRD trip breakers without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1

SR 3.3.4.1 is to perform a CHANNEL FUNCTIONAL TEST every 92 days. This test verifies the OPERABILITY of the trip devices by actuation of the end devices. Also, this test independently verifies the undervoltage and shunt trip mechanisms of the trip breakers. The Frequency of 92 days is based on operating experience, which has demonstrated that failure of more than one channel of a given function in any 92 day interval is a rare event (Ref. 3).

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.1 (continued)

Testing in accordance with this SR is normally performed on a rotational basis with one channel being tested each 23 days. Testing one channel each 23 days reduces the probability of an undetected failure existing within the system and minimizes the likelihood of the same systematic test errors being introduced into each redundant trip device.

REFERENCES

1. SAR, Chapter 7.

2. 10 CFR 50.36.

3. BAW-10167A, "Justification for Increasing the Reactor Trip System On-Line Test Intervals," Supplement 3, "Justification for Increasing the Trip Device Test Interval," February 1998.

B 3.3 INSTRUMENTATION

B 3.3.5 Engineered Safeguards Actuation System (ESAS) Instrumentation

BASES

BACKGROUND

The ESAS initiates necessary safety systems, based on the values of selected unit Parameters, to protect against violating core design limits and to mitigate accidents.

The ESAS operates in a distributed manner to initiate the appropriate systems. The ESAS does this by determining the need for actuation in each of three analog instrument channels monitoring each actuation Parameter. Once the need for actuation is determined, the condition is transmitted to digital actuation logic channels, which perform the two-out-of-three logic to determine the actuation of each end device.

Three Parameters are used for actuation:

- Low Reactor Coolant System (RCS) Pressure;
- High Reactor Building (RB) Pressure; and
- High High RB Pressure.

LCO 3.3.5 covers only the analog instrument channels that measure these Parameters. These channels include the equipment necessary to produce an actuation signal input to the digital actuation logic channels. This includes sensors, bistable devices, operational bypass circuitry, and logic buffer modules. LCO 3.3.6, "Engineered Safeguards Actuation System (ESAS) Manual Initiation," and LCO 3.3.7, "Engineered Safeguards Actuation System (ESAS) Actuation Logic," provide requirements on the manual initiation and digital actuation logic Functions.

The ESAS monitors three parameters via analog instrument channels. Each analog instrument channel provides input to the appropriate digital actuation logic channels that initiate equipment with a two-out-of-three coincidence logic on each digital channel. Each digital actuation logic channel includes bistable inputs from all three analog instrument channels of one parameter, i.e., either Low RCS Pressure, High RB Pressure, or High High RB Pressure. The digital actuation logic combines the analog instrument channel trips to actuate the individual Engineered Safeguards (ES) components needed to initiate each ES System. Figure 7.6, SAR, Chapter 7 (Ref. 1), also illustrates how analog instrument channel trips combine to cause digital actuation logic channel trips.

The ESAS is divided into five Functions actuated by ten digital actuation logic channels.

BACKGROUND (continued)

The ESAS High Pressure Injection (HPI) Function is actuated by ESAS digital actuation logic channels 1 and 2 and includes the following system actuations: HPI, a subset of RB isolation valves, diesel generators (DGs), and ES electrical alignment. Digital actuation logic channels 1 and 2 are actuated by two-out-of-three RCS Pressure—Low analog instrument channels, or two-out-of-three RB Pressure—High analog instrument channels.

The ESAS Low Pressure Injection (LPI) Function is actuated by ESAS digital actuation logic channels 3 and 4 and includes the following system actuations: LPI, a subset of RB isolation valves, and emergency feedwater (EFW) through an ESAS signal provided to the Emergency Feedwater Initiation and Control (EFIC) Instrumentation System. Digital actuation logic channels 3 and 4 are actuated by two-out-of-three RCS Pressure—Low analog instrument channels, or two-out-of-three RB Pressure—High analog instrument channels.

The ESAS RB Cooling Function is actuated by ESAS digital actuation logic channels 5 and 6 and includes the following system actuations: RB cooling, a subset of RB isolation valves, and RB penetration room ventilation system. Digital actuation logic channels 5 and 6 are actuated by two-out-of-three RB Pressure—High analog instrument channels.

The ESAS RB Spray Function is actuated by ESAS digital actuation logic channels 7 and 8 and includes the following system actuations: RB spray. Digital actuation logic channels 7 and 8 are actuated by two-out-of-three RB Pressure—High High analog instrument channels.

The ESAS Spray Additive Function is actuated by ESAS digital actuation logic channels 9 and 10 and includes the following system actuations: spray additive. Digital actuation logic channels 9 and 10 are actuated by two-out-of-three RB Pressure—High High analog instrument channels.

The following matrix identifies the ESAS digital actuation logic channels and the systems actuated by each Parameter.

ESAS Digital Actuation Logic Channels	Actuated Systems	Parameter		
		RCS Press. Low	RB Press. High	RB Press. High High
1 and 2	Subset of RB Isolation, ES Electrical Alignment, HPI, and DG Start	X	X	
3 and 4	Subset of RB isolation, LPI, and EFIC EFW	X	X	
5 and 6	Subset of RB Isolation, RB Cooling, and Penetration Room Ventilation		X	
7 and 8	RB Spray			X
9 and 10	Spray Additive			X

BACKGROUND (continued)

The ES equipment is divided between the two redundant actuation trains. The division of the equipment between the two actuation trains is based on the equipment redundancy and function and is accomplished in such a manner that the failure of one of the digital actuation logic channels and the related safeguards equipment will not inhibit the overall ES Functions. Redundant ES pumps are controlled from separate and independent digital actuation logic channels.

The actuation of ES equipment is also available by manual actuation switches located on the control room console.

The ESAS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate Design Basis Accidents (DBAs), specifically the loss of coolant accident (LOCA), RB DBA, and as a backup to mitigate the steam line break (SLB) event (Ref. 2). The ESAS relies on the OPERABILITY of the digital actuation logic channels to perform the actuation of the selected systems.

Engineered Safeguards Actuation System Bypasses

No provisions are made for maintenance bypass of ESAS instrumentation channels. Operational bypass of certain channels is necessary to allow accident recovery actions to continue and, for some channels, to allow reactor shutdown without ESAS actuation.

The ESAS RCS pressure analog instrument channels include permissive bistables that allow manual bypass when reactor pressure is below the point at which the low pressure trip is required to be OPERABLE. Once permissive conditions are sensed, the RCS pressure trips may be manually bypassed. Bypasses are automatically removed when bypass permissive conditions are exceeded. Failure of the automatic bypass removal feature or the inability to bypass the RCS pressure function when below 1750 psig does not constitute channel inoperability. However, a channel that remains bypassed when pressure is raised above 1750 psig will be considered inoperable and appropriate conditions will be entered.

This bypass provides an operational provision only outside the Applicability for this Parameter, and provides no safety function. The automatic bypass removal feature is verified during the monthly CHANNEL FUNCTIONAL TEST.

Reactor Coolant System Pressure

The RCS pressure is monitored by three independent pressure transmitters located in the RB. These transmitters are separate from the transmitters that feed the Reactor Protection System (RPS). Each of the pressure signals generated by these transmitters is monitored by two bistables to provide a trip signal at $\geq$ 1585 psig and a bypass permissive signal at $\leq$ 1750 psig.

BACKGROUND (continued)

The outputs of the three low RCS pressure trip bistables drive relays in two sets of identical and independent digital instrument channels. These two sets of channels each use two-out-of-three coincidence digital logic for actuation.

Each analog channel can be tested online to verify that the signal and trip setpoint are within the specified allowance requirements of approved calibration procedures. The built-in test facilities permit an electrical trip test of each analog instrument string by the substitution of signals at the buffer amplifiers. When an analog instrument string is placed in test, all associated analog subsystem outputs go to the trip state. This assures that all protective action cannot be defeated by placing analog instrument strings in test.

Reactor Building Pressure

The RB pressure is monitored by three independent pressure transmitters located inside the RB. These transmitters are separate from the transmitters that feed the Reactor Protection System (RPS). Each of the pressure signals generated by these transmitters is monitored by two bistables to provide trip signals. The outputs of the bistables, associated with the RB Pressure—High and RB Pressure—High High trips, drive relays in two sets of identical and independent digital instrument channels. These two sets of channels each use two-out-of-three coincidence digital logic for automatic actuation.

Each channel can be tested online to verify that the signal and trip setpoint are within the specified allowance requirements of approved calibration procedures. The built-in test facilities permit an electrical trip test of each analog instrument string by the substitution of signals at the buffer amplifiers. When an analog instrument string is placed in test, all associated analog subsystem outputs go to the trip state. This assures that all protective action cannot be defeated by placing analog instrument strings in test.

Trip Setpoints and Allowable Values

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

The trip setpoints used in the bistables are based on the analytical limits used in the safety analysis described in SAR, Chapter 14 and Chapter 3A (Ref. 2). The selection of these trip setpoints is such that adequate protection is provided when appropriate sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and environment induced errors, the Allowable Values specified in Table 3.3.5-1 are equal to or conservatively adjusted with respect to the analytical limits. Guidance used to calculate uncertainties associated with the trip setpoints is provided in the Instrument Loop Error Analysis and Setpoint Methodology Manual, Design Guide,

BACKGROUND (continued)

IDG-001 (Ref. 4). The trip setpoint entered into the bistable may be more conservative than that specified by the Allowable Value to account for changes in instrument error detectable by a CHANNEL FUNCTIONAL TEST. A channel is inoperable if its as-found trip setpoint is not within its required Allowable Value.

Setpoints, in accordance with the Allowable Values, ensure that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the DBA and the equipment functions as analyzed.

Each channel can be tested online to verify that the trip setpoint is within the specified allowance requirements of approved calibration procedures. Once a designated channel is taken out of service for testing, a simulated signal may be injected in place of the field instrument signal. The process equipment for the channel may then be tested, verified, and calibrated.

APPLICABLE SAFETY ANALYSES

The following ESAS Functions have been assumed within the accident analyses.

High Pressure Injection

The ESAS actuation of HPI has been assumed for core cooling in the LOCA analysis and is available for boron addition in the SLB analysis.

Low Pressure Injection

The ESAS actuation of LPI has been assumed for large break LOCAs.

Reactor Building Spray, Reactor Building Cooling, and Reactor Building Isolation

The ESAS actuation of the RB coolers and RB Spray have been credited in the RB analysis for LOCAs. Accident dose calculations have credited RB Penetration Room Ventilation, RB Isolation, and RB Spray. The MSLB analysis also credits ESAS actuation of the RB Cooling and RB Spray.

Emergency Power

The ESAS initiated DG Start and ES electrical equipment alignment have been included in the design to ensure that emergency power is available throughout the limiting LOCA scenarios.

The small and large break LOCA analyses (Ref. 2) assume a conservative delay time for the actuation of HPI and LPI. This delay time includes allowances for DG starting, DG loading, Emergency Core Cooling Systems (ECCS) pump starts, and valve alignment. Similarly, the RB Cooling, RB Isolation, and RB Spray have been analyzed with delays appropriate for the entire system analyzed.

APPLICABLE SAFETY ANALYSES (continued)

Accident analyses rely on automatic ESAS actuation for protection of the core temperature and containment pressure limits and for limiting off site dose levels following an accident. These include LOCA, SLB, and other events that result in RCS inventory reduction or severe loss of RCS cooling.

The ESAS instrumentation satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3) for operation in MODE 1. There are no specific safety analyses for operation in MODES 2, 3 and 4. However, industry operating experience has identified the ESAS instrumentation as significant to public health and safety during these operating conditions. Therefore, the ESAS instrumentation satisfies Criterion 4 of 10 CFR 50.36 for operation in MODES 2, 3 and 4.

LCO

The LCO requires three ESAS analog instrument channels for each Parameter in Table 3.3.5-1 to be OPERABLE. Failure of any instrument renders the affected analog instrument channel(s) inoperable and reduces the reliability of the affected Functions.

Only the Allowable Value is specified for each ESAS Function in the LCO. Trip setpoints are provided in the calibration procedures. The trip setpoints are selected to ensure the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the Allowable Value if the bistable is performing as required. Each Allowable Value specified is equal to or more conservative than any analytical limit assumed in the safety analysis to account for instrument uncertainties appropriate to the trip Parameter. Guidance used to calculate the uncertainties associated with the trip setpoints is contained in Instrument Loop Error Analysis Setpoint Methodology, Design Guide, IDG-001 (Ref. 3).

The values for bypass removal functions are stated in the Applicable MODES or Other Specified Condition column of Table 3.3.5-1.

Three ESAS analog instrument channels shall be OPERABLE to ensure that a single failure in one channel will not result in loss of the ability to automatically actuate the required safety systems.

Reactor Coolant System Pressure

Three channels of RCS Pressure - Low are required OPERABLE. Each channel includes a sensor, trip bistable, bypass bistable, bypass relays, and output relays. Failure of a bypass bistable or bypass circuitry, such that an analog instrument channel cannot be bypassed, does not render the channel inoperable since the channel is still capable of performing its safety function, i.e., this is not a safety related bypass function.

LCO (continued)

The trip setpoints are the nominal values at which the bistables are set. For the RCS Pressure—Low, the limiting safety analysis assumes the HPI, LPI, EFIC EFW, ES electrical alignment, and two subsets of RB isolation actuate at ≥ 1520 psig (≥ 1535 psia). The Allowable Value of ≥ 1585 psig includes considerations for instrumentation error and an allowance for margin. Allowances for instrument drift and additional margin are included in the trip setpoint.

Guidance used to calculate the uncertainties associated with the trip setpoints is provided in Instrument Loop Error Analysis and Setpoint Methodology Manual, Design Guide, IDG-001 (Ref. 4). The explicit uncertainties associated with each setpoint are addressed in the individual design calculations or calibration procedures. Setpoints in accordance with the Allowable Value in conjunction with the LCOs and administrative controls ensure that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the DBA and the equipment functions as analyzed. An analog instrument channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Reactor Building Pressure

Three channels of RB Pressure—High and RB Pressure—High High are required to be OPERABLE. Each channel includes a pressure switch, bypass relays, and output relays.

The trip setpoints are the nominal values at which the bistables are set. Credit is taken in the safety analyses for RB Pressure—High trip for the actuation of selected systems. The safety analyses for reactor building performance and equipment environmental qualification (pressure and temperature envelope definition) conservatively assume the RB cooling is not initiated until well beyond the expected actual automatic actuation time frame. Therefore, no additional consideration of the instrumentation uncertainties is warranted.

Credit is taken in the safety analyses for RB Pressure—High High trip for the actuation of selected systems. The safety analyses for reactor building performance and equipment environmental qualification (pressure and temperature envelope definition) conservatively assumes the RB spray is not initiated until well beyond the expected actual automatic actuation time frame. Therefore, no additional consideration of the instrumentation uncertainties is warranted.

Therefore, the bistable is considered to be properly adjusted when the "as left" value is consistent with the identified Allowable Value, i.e., for this parameter the trip setpoint and the Allowable Value are the same. Guidance used to calculate the uncertainties associated with the trip setpoints is provided in Instrument Loop Error Analysis and Setpoint Methodology Manual, Design Guide, IDG-001 (Ref. 4). Setpoints in accordance with the Allowable Value ensure that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the DBA and the equipment functions as analyzed. An analog instrument channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

APPLICABILITY

Three ESAS analog instrument channels for each of the following Parameters shall be OPERABLE.

1. Reactor Coolant System Pressure - Low Setpoint

The RCS Pressure - Low Setpoint actuation Parameter shall be OPERABLE during operation at or above 1750 psig. This requirement ensures the capability to automatically actuate safety systems and components during conditions indicative of a LOCA or secondary unit overcooling. Below 1750 psig, the low RCS Pressure actuation Parameter can be bypassed to avoid actuation during normal unit cooldowns when safety systems actuations are not required.

The allowance for the bypass is consistent with the transition of the unit to a lower energy state, where there is more margin to safety limits. The unit response to any event, given that the reactor is already tripped, will be less severe and allows more time for operator action to provide manual safety system actuations than in higher energy states. This is even more appropriate during unit heatups when the primary system and core energy content is low, prior to power operation.

In MODES 5 and 6, there is more time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident than in higher MODES. RCS pressure and temperature are very low, and many ES components are administratively controlled or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

2., 3. Reactor Building Pressure - High Setpoint and Reactor Building Pressure - High High Setpoint

The RB Pressure - High and RB Pressure - High High actuation Functions of ESAS shall be OPERABLE in MODES 1, 2, 3, and 4 when the potential for a HELB exists. In MODES 5 and 6, there is more time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident than in higher MODES. Plant pressure and temperature are very low and many ES components are administratively controlled or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

ACTIONS

Required Actions A and B apply to the ESAS instrumentation Parameters listed in Table 3.3.5-1.

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each Parameter.

If an analog instrument channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or ESAS bistable is found inoperable, then all affected functions provided by that channel should be declared inoperable and the unit must enter the Conditions for the particular protection Parameter affected.

A.1

Condition A applies when one analog instrument channel becomes inoperable in one or more Parameters. If one ESAS analog instrument channel is inoperable, placing it in a tripped condition leaves the system in a one-out-of-two condition for actuation. Thus, if another analog instrument channel were to fail, the ESAS instrumentation could still perform its actuation functions. This action is completed when all of the affected output relays are tripped. This can normally be accomplished by tripping the affected bistables.

The 1 hour Completion Time is sufficient time to perform the Required Action.

B.1, B.2, and B.3

Condition B applies when Required Action A.1 and its associated Completion Time are not met, or when one or more parameters have more than one analog instrument channel inoperable. If Condition B applies, the unit must be brought to a condition in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours. Additionally, for the RCS Pressure—Low parameter, the unit must be brought to < 1750 psig within 36 hours, and for the RB Pressure—High and High High parameters, the unit must be brought to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

The ESAS Parameters listed in Table 3.3.5-1 are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION.

SR 3.3.5.1

Performance of the CHANNEL CHECK every 12 hours provides reasonable assurance for prompt identification of a gross failure of instrumentation. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between CHANNEL CALIBRATIONs.

Agreement criteria are determined by the unit staff, based on a combination of factors including channel instrument uncertainties. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but potentially more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels.

SR 3.3.5.2

A CHANNEL FUNCTIONAL TEST is performed on each required ESAS analog instrument channel to ensure the entire channel will perform the intended functions. Any setpoint adjustment shall be consistent with the assumptions of the setpoint calculations.

The Frequency of 31 days is based on unit operating experience, with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any 31 day interval is a rare event. The RCS low pressure automatic bypass removal feature is verified during its CHANNEL FUNCTIONAL TEST.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.3

CHANNEL CALIBRATION is a complete check of the analog instrument channel, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the analog instrument channel remains OPERABLE between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint calculations. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint calculations.

This Frequency is justified by the assumption of at least an 18 month calibration interval to determine the magnitude of equipment drift in the setpoint calculations.

REFERENCES

1. SAR, Chapter 7.
2. SAR, Chapter 14 and Chapter 3A.
3. 10 CFR 50.36.
4. Instrument Loop Error Analysis and Setpoint Methodology Manual, Design Guide, IDG-001.

B 3.3 INSTRUMENTATION

B 3.3.6 Engineered Safeguards Actuation System (ESAS) Manual Initiation

BASES

BACKGROUND

The ESAS manual initiation capability allows the operator to actuate ESAS Functions from the control room in the absence of any other initiation condition. This ESAS manual initiation capability is provided in the event the operator determines that an ESAS Function is needed and has not been automatically actuated. Furthermore, the ESAS manual initiation capability allows operators to rapidly initiate Engineered Safeguards (ES) Functions if the trend of unit parameters indicates that ES actuation will be needed.

LCO 3.3.6 covers only the system level manual initiation of these Functions. LCO 3.3.5, "Engineered Safeguards Actuation System (ESAS) Instrumentation," and LCO 3.3.7, "Engineered Safeguards Actuation System (ESAS) Actuation Logic," provide requirements on the portions of the ESAS that automatically initiate the Functions described earlier.

The ESAS manual initiation Function relies on the OPERABILITY of the digital actuation logic channels (LCO 3.3.7) to perform the actuation of the systems. A manual trip push button is provided on a control room console for each of the digital actuation logic channels. Operation of the push button energizes relays whose contacts perform a logical "<u>OR</u>" function with the automatic actuation.

The ESAS manual initiation channel is defined as the console switch and the instrumentation from the console switch to, but not including, the digital actuation logic channels, which actuate the end devices. Other means of manual initiation, such as controls for individual ES devices, may be available in the control room and other unit locations. These alternative means are not required by this LCO, nor may they be credited to fulfill the requirements of this LCO.

APPLICABLE SAFETY ANALYSES

The ESAS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate Design Basis Accidents, specifically, the loss of coolant accident (LOCA), RB DBA and as a backup to mitigate the steam line break event.

The ESAS manual initiation ensures that the control room operator can rapidly initiate ES Functions. The manual initiation trip Function is required as a backup to automatic trip functions and allows operators to initiate ESAS whenever any parameter is rapidly trending toward its trip setpoint.

APPLICABLE SAFETY ANALYSES (continued)

Operating experience has shown the ESAS manual initiation function to be significant to public health and safety, and therefore satisfy Criterion 4 of 10 CFR 50.36 (Ref. 1).

LCO

Two ESAS manual initiation channels of each ESAS Function shall be OPERABLE whenever conditions exist that could require ES protection of the reactor or RB. Two OPERABLE channels ensure that no single random failure will prevent system level manual initiation of any ESAS Function. The ESAS manual initiation Function allows the operator to initiate protective action prior to automatic initiation or in the event the automatic initiation does not occur.

The ESAS is divided into five Functions actuated by ten manual initiation channels as indicated in the following table:

Function	Associated Channels
High Pressure Injection	1 & 2
Low Pressure Injection	3 & 4
RB Cooling	5 & 6
RB Spray	7 & 8
Spray Additive	9 & 10

The ESAS High Pressure Injection (HPI) Function is actuated by ESAS Manual Initiation channels 1 and 2 and includes the following system actuations: HPI, a subset of reactor building (RB) isolation valves, diesel generators, and ES electrical alignment.

The ESAS Low Pressure Injection (LPI) Function is actuated by ESAS Manual Initiation channels 3 and 4 and includes the following system actuations: LPI, a subset of RB isolation valves, and emergency feedwater (EFW) through an ESAS signal provided to the Emergency Feedwater Isolation and Control (EFIC) System.

The ESAS RB Cooling Function is actuated by ESAS Manual Initiation channels 5 and 6 and includes the following system actuations: RB cooling, a subset of RB isolation valves, and RB penetration room ventilation system.

The ESAS RB Spray Function is actuated by ESAS Manual Initiation channels 7 and 8 and includes the following system actuations: RB spray.

The ESAS Spray Additive Function is actuated by ESAS Manual Initiation channels 9 and 10 and includes the following system actuations: spray additive.

APPLICABILITY

The ESAS manual initiation Functions shall be OPERABLE in MODES 1 and 2, and in MODES 3 and 4 when the associated ES equipment is required to be OPERABLE. The manual initiation channels are required because ES Functions are designed to provide protection in these MODES. ESAS initiates systems that are either reconfigured for decay heat removal operation or disabled while in MODES 5 and 6. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components. Time is available to evaluate unit conditions and to respond by manually operating the ES components, if required.

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each ESAS manual initiation Function.

A.1

Condition A applies when one manual initiation channel of one or more ESAS Functions becomes inoperable. Required Action A.1 must be taken to restore the channel to OPERABLE status within the next 72 hours. The Completion Time of 72 hours is based on unit operating experience and administrative controls, which provide alternative means of ESAS Function initiation via individual component controls. The 72 hour Completion Time is generally consistent with the allowed outage time for the safety systems actuated by ESAS.

B.1 and B.2

If Required Action A.1 and the associated Completion Time are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the ESAS manual initiation. This test verifies that the initiating circuitry is OPERABLE and will actuate the digital actuation logic channels. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency is demonstrated to be sufficient, based on operating experience, which shows these components usually pass the Surveillance when performed on the 18 month Frequency.

REFERENCES

1. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.7 Engineered Safeguards Actuation System (ESAS) Actuation Logic

BASES

BACKGROUND

The digital actuation logic channels of ESAS are defined as the instrumentation between, but not including, the buffers of the analog instrument channels and the unit controls that actuate ESAS equipment. Each of the components actuated by the ESAS Functions is associated with one or more digital actuation logic channels. If two-out-of-three ESAS analog instrument channels indicate a trip, or if channel level manual initiation occurs, the digital actuation logic channel is activated and the associated equipment is actuated.

The purpose of requiring OPERABILITY of the ESAS digital actuation logic channels is to ensure that the Functions of the ESAS can be automatically initiated in the event of an accident. Automatic actuation of some Functions is necessary to prevent the unit from exceeding the Emergency Core Cooling Systems (ECCS) limits in 10 CFR 50.46 (Ref. 1). It should be noted that OPERABLE digital actuation logic channels alone will not ensure that each Function can be activated; the analog instrument channels and actuated equipment associated with each Function must also be OPERABLE to ensure that the Functions can be automatically initiated during an accident.

LCO 3.3.7 covers only the digital actuation logic channels that initiate these Functions. LCO 3.3.5, "Engineered Safeguards Actuation System (ESAS) Instrumentation," and LCO 3.3.6, "Engineered Safeguards Actuation System (ESAS) Manual Initiation," provide requirements on the analog instrument and manual initiation channels that input to the digital actuation logic channels.

The ESAS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate Design Basis Accidents (DBAs), specifically, the loss of coolant accident (LOCA) and steam line break (SLB) events. The ESAS relies on the OPERABILITY of the digital actuation logic for each component to perform the actuation of the selected systems.

The small and large break LOCA analyses assume a conservative delay time for the actuation of high pressure injection (HPI) and low pressure injection (LPI) in BAW-10103A, Rev. 3 (Ref. 2). This delay time includes allowances for diesel generator (DG) starts, DG loading, ECCS pump starts, and valve alignment. Similarly, the reactor building (RB) Cooling, RB Isolation, and RB Spray have been analyzed with delays appropriate for the entire system.

BACKGROUND (continued)

The ESAS automatic initiation of Engineered Safeguards (ES) Functions to mitigate accident conditions is assumed in the DBA analysis and is required to ensure that consequences of analyzed events do not exceed the accident analysis predictions. Automatically actuated features include HPI, LPI, RB Cooling, RB Spray, RB Spray Additive, and RB Isolation.

APPLICABLE SAFETY ANALYSES

Accident analyses rely on automatic ESAS actuation for protection of the core and RB and for limiting off site dose levels following an accident. The digital actuation logic is an integral part of the ESAS.

The ESAS actuation logic satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3) for operation in MODE 1. There are no specific safety analyses for operation in MODES 2, 3 and 4. However, industry operating experience has identified the ESAS actuation logic as significant to public health and safety during these operating conditions. Therefore, the ESAS actuation logic satisfies Criterion 4 of 10 CFR 50.36 for operation in MODES 2, 3 and 4.

LCO

The digital actuation logic channels are required to be OPERABLE whenever conditions exist that could require ES protection of the reactor or the RB. This ensures automatic initiation of the ES required to mitigate the consequences of accidents.

The ESAS is divided into five Functions actuated by ten digital actuation logic channels as indicated in the following table:

Function	Associated Channels
High Pressure Injection	1 & 2
Low Pressure Injection	3 & 4
RB Cooling	5 & 6
RB Spray	7 & 8
Spray Additive	9 & 10

The ESAS HPI Function is actuated by ESAS digital actuation logic channels 1 and 2 and includes the following system actuations: HPI, a subset of RB isolation valves, DGs, and ES electrical alignment. Digital actuation logic channels 1 and 2 are actuated by two-out-of-three RCS Pressure—Low analog instrument channels, or two-out-of-three RB Pressure—High analog instrument channels.

LCO (continued)

The ESAS LPI Function is actuated by ESAS digital actuation logic channels 3 and 4 and includes the following system actuations: LPI, a subset of RB isolation valves, and EFW through an ESAS signal provided to EFIC. Digital actuation logic channels 3 and 4 are actuated by two-out-of-three RCS Pressure—Low analog instrument channels, or two-out-of-three RB Pressure—High analog instrument channels.

The ESAS RB Isolation and Cooling Function is actuated by ESAS digital actuation logic channels 5 and 6 and includes the following system actuations: RB cooling, a subset of RB isolation valves, and RB penetration room ventilation system. Digital actuation logic channels 5 and 6 are actuated by two-out-of-three RB Pressure—High analog instrument channels.

The ESAS RB Spray Function is actuated by ESAS digital actuation logic channels 7 and 8 and includes the following system actuations: RB spray. Digital actuation logic channels 7 and 8 are actuated by two-out-of-three RB Pressure—High High analog instrument channels.

The ESAS Spray Additive Function is actuated by ESAS digital actuation logic channels 9 and 10 and includes the following system actuations: spray additive. Digital actuation logic channels 9 and 10 are actuated by two-out-of-three RB Pressure—High High analog instrument channels.

APPLICABILITY

The digital actuation logic channels shall be OPERABLE in MODES 1 and 2, and in MODES 3 and 4 when the associated ES equipment is required to be OPERABLE, because ES Functions are designed to provide protection in these MODES. Automatic actuation in MODE 5 or 6 is not required because the systems initiated by the ESAS are either reconfigured for decay heat removal operation or disabled. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components. Time is available to evaluate unit conditions and respond by manually operating the ES components, if required.

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each ESAS digital actuation logic channel.

ACTIONS (continued)

A.1 and A.2

When one or more digital actuation logic channel(s) are inoperable, the associated component(s) can be placed in their ES configuration. Required Action A.1 is equivalent to the digital actuation logic channel performing its safety function ahead of time. In some cases, placing the component in its ES configuration would violate unit safety or operational considerations. In these cases, the component status should not be changed, but the supported system component must be declared inoperable. Conditions which would preclude the placing of a component in its ES configuration include, but are not limited to, violation of system separation, activation of fluid systems that could lead to thermal shock, isolation of fluid systems that are normally functioning, and actuation of components which would not return to their actuated condition upon restoration of electrical power. The Completion Time of 1 hour is based on operating experience and reflects the urgency associated with the inoperability of a safety system component.

Required Action A.2 requires entry into the Required Actions of the affected supported systems, since the true effect of digital actuation logic channel failure is inoperability of the supported system. The Completion Time of 1 hour is based on operating experience and reflects the urgency associated with the inoperability of a safety system component. A combination of Required Actions A.1 and A.2 may be used for different components associated with an inoperable ESAS digital actuation logic channel.

SURVEILLANCE REQUIREMENTS

SR 3.3.7.1

SR 3.3.7.1 is the performance of a CHANNEL FUNCTIONAL TEST on a 31 day Frequency. The test demonstrates that each digital actuation logic channel successfully performs the two-out-of-three logic combinations every 31 days. The test simulates the required one-out-of-three inputs to the logic circuit and verifies the successful operation of the digital actuation logic. The Frequency is based on operating experience that demonstrates the rarity of more than one channel failing within the same 31 day interval. The CHANNEL FUNCTIONAL TEST performed for the Reactor Building Spray System Logic Channels shall include testing of the associated spray pump, spray valves, and chemical additive valve logic channels.

REFERENCES

1. 10 CFR 50.46.
2. BAW-10103A, Rev. 3, July 1977.
3. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.8 Diesel Generator (DG) Loss of Power Start (LOPS)

BASES

BACKGROUND

The DGs provide a source of emergency power when offsite power is either unavailable or is insufficiently stable to allow operation of safety related loads. Undervoltage protection will generate a LOPS in the event a loss of voltage or degraded voltage condition occurs on unit vital buses. There are two LOPS Functions for each 4.16 kV vital bus.

Two undervoltage relays with inverse voltage time characteristics are provided on each 4.16 kV Class 1E bus for the purpose of detecting a loss of bus voltage. The relay Allowable Values are based on a maximum setting, which is below the lowest allowed motor terminal momentary voltage of 75% of motor voltage rating of 4000 V. The Allowable Values are adjusted to include channel uncertainties and calibration tolerances. Upon loss of power to either of these relays, in approximately 1.0 second, load shedding and starting of the associated DG are initiated. Isolation of the safety related buses is delayed approximately 2.0 seconds to allow an automatic transfer to offsite power. The safety related bus is isolated only if the transfer is unsuccessful.

Two definite time undervoltage relays are provided on each safety related 480 V load center bus with a coincident trip logic (2 out of 2) for the purpose of detecting a sustained undervoltage condition. The undervoltage relay Allowable Values on the 480 V bus are based on long term motor voltage requirements plus the maximum feeder voltage drop allowance resulting in a nominal setting of 92% of the motor rated voltage of 460 V. The Allowable Values are adjusted to include channel uncertainties and calibration tolerances. Upon voltage degradation to 92% of 460 V and after a delay of 8 seconds, both relays must operate to isolate the associated safety related 4.16 kV bus from offsite power, and start and connect the associated DG. The relays are delayed 8.0 seconds to prevent spurious operation of the relays when large motors start on the safety related 4.16 kV and 480 V buses. The LOPS is further described in SAR, Section 8.3.1 (Ref. 1).

Trip Setpoints

The Allowable Values associated with the relays are consistent with the analytical limits presented in SAR, Section 8.3.1 (Ref. 1). The selection of these values is such that adequate protection is provided when all sensor and processing time delays are taken into account. A channel is inoperable if its actuation trip setpoint is not within its required Allowable Value.

BACKGROUND (continued)

A complete loss of offsite power will result in approximately a 1 second delay in LOPS actuation. The DG starts and is available to accept loads within a 15 second time interval on actuation by the Engineered Safeguards Actuation System (ESAS) or LOPS. Emergency power is established within the maximum time delay assumed for each event analyzed in the accident analysis in which a loss of offsite power is assumed (Ref. 2).

The DG LOPS protection channels conform to the single failure criteria of IEEE-279-1971 as discussed in Ref. 1.

APPLICABLE SAFETY ANALYSES

The DG LOPS is required for the Engineered Safeguards (ES) to function in any accident which assumes a loss of offsite power.

Accident analyses credit the loading of the DG, based on the loss of offsite power, during a loss of coolant accident (LOCA). The actual DG start has historically been associated with the ESAS actuation. The diesel loading has been included in the assumed delay time associated with each safety system component requiring DG supplied power following a loss of offsite power. The analysis assumes a nonmechanistic DG loading, which does not explicitly account for each individual component of the loss of power detection and subsequent actions. The total assumed actuation time for the limiting systems, high pressure injection, and low pressure injection includes contributions from the DG Start, DG loading, and safety injection system component actuation. The response of the DG to a loss of power must be demonstrated to fall within this analysis response time when including the contributions of all portions of the delay.

The required channels of LOPS, in conjunction with the ES systems powered from the DGs, provide unit protection for the analyzed accidents in which a loss of offsite power is assumed.

The delay times assumed in the safety analysis for the ES equipment include the 15 second DG start delay and, if applicable, the appropriate sequencing delay. The assumed response times for ESAS actuated equipment in LCO 3.3.5, "Engineered Safeguards Actuation System (ESAS) Instrumentation," include the appropriate DG loading and sequencing delay.

In MODE 1, the DG LOPS channels satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3). There are no specific safety analyses for operation in MODES 2, 3, and 4. However, industry operating experience has identified DG LOPS as significant to public health and safety during these operating conditions. Therefore, in MODES 2, 3 and 4, the DG LOPS channels satisfy Criterion 4 of 10 CFR 50.36.

LCO

The LCO for the DG LOPS requires that two relays per DG (DG1 and DG2) of the loss of voltage instrumentation Function shall be OPERABLE and two relays per DG of the degraded voltage instrumentation Function shall be OPERABLE to ensure that the automatic 4.16 kV bus isolation capability and automatic start of the DG is available when needed. The degraded voltage relays may be bypassed for ≤ 30 seconds during reactor coolant pump start to prevent such starts from initiating spurious DG LOPS, separation of the ES busses from offsite power, and subsequent loading of the DG. Therefore, the automatic bypass and associated alarms are required functions for OPERABILITY of the DG LOPS instrumentation.

Loss of either DG LOPS function could result in the delay of safety systems initiation when required. This could lead to unacceptable consequences during accidents.

The Allowable Values must be met for each Function to be considered OPERABLE. Allowable Values are specified in the specifications. The setpoints are selected to ensure that the setpoint measured by CHANNEL FUNCTIONAL TESTS does not exceed the Allowable Value if the relay is performing as required. Each Allowable Value is more conservative than any analytical limit assumed in the transient and accident analysis and the trip setpoint is equal to or more conservative to the Allowable Value, accounting for instrument uncertainties appropriate to the trip function. Guidance used to calculate the uncertainties associated with the relay settings is contained in the ANO-1 Design Guide, IDG-001, "Instrument Loop Error Analysis and Setpoint Methodology Manual" (Ref. 4).

The LOPS relay Allowable Values are based on the short term starting voltage protection as well as long term running voltage protection. The 4.16 kV undervoltage relay Allowable Values are based on a maximum setting, which is below the lowest allowed motor terminal momentary voltage of 75% of motor rated voltage of 4000 V. The 480 V undervoltage relay Allowable Values are based on long term motor voltage requirements plus the maximum feeder voltage drop allowance resulting in a nominal 92% setting of the motor rated voltage of 460 V. The Allowable Values for both the 4.16 kV and the 480 V relays include adjustments for channel uncertainties and calibration tolerances.

APPLICABILITY

The DG LOPS actuation Function shall be OPERABLE in MODES 1, 2, 3, and 4 because ES Functions are required to be OPERABLE in these MODES. Automatic actuation is not required in MODES 5 or 6 since there is no automatic protective function on a loss of power or degraded power to the vital bus.

ACTIONS

A Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each Function.

If a relay's trip setpoint is found nonconservative with respect to the Allowable Value, or the relay is found inoperable, then the function that the relay provides must be declared inoperable and the LCO Condition entered for the particular protection function affected. Since the required relay Functions are specified on a per DG basis, the Condition may be entered separately for each DG.

A.1

With one or more relays in one or more Functions for one or more DGs inoperable, Required Action A.1 requires the inoperable relay(s) to be restored to OPERABLE status within 1 hour. With a relay of a Function inoperable, the logic is not capable of providing an automatic DG LOPS signal for valid conditions for the associated DG. The 1 hour Completion Time is reasonable to evaluate and to take action by correcting the degraded condition in an orderly manner and takes into account the low probability of an event requiring LOPS occurring during this interval.

B.1

Condition B applies if the Required Action and associated Completion Time of Condition A are not met.

Required Action B.1 ensures that Required Actions for affected diesel generator inoperabilities are initiated. Depending on the DG(s) affected, the appropriate Actions specified in LCO 3.8.1, "AC Sources - Operating," are required immediately.

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1

Performance of the CHANNEL CHECK once every 7 days provides reasonable assurance for prompt identification of a gross failure of instrumentation. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

The Frequency is based on operating experience that demonstrates channel failure is rare. Since the probability of random failure in any 7 day period is low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of this instrumentation.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.8.2

The Note allows channel bypass for testing of the loss of voltage Function without entering the associated Conditions and Required Actions, although during this time period it cannot actuate a diesel start. This allowance is based on the assumption that 4 hours is the average time required to perform channel Surveillance. The 4 hour testing allowance does not significantly reduce the probability that the DG will start when necessary. It is not acceptable to remove channels from service for more than 4 hours to perform required Surveillance testing without declaring the channel inoperable.

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The setpoints and the response to a loss of voltage and a degraded voltage test shall include a single point verification that the trip occurs within the required delay time. CHANNEL CALIBRATION shall verify that setpoints are within the required ranges.

The Frequency is based on the reliability of the components, on operating experience which demonstrates channel failure is rare, and on consistency with the typical industry refueling cycle, and is justified by the assumption of at least an 18 month calibration interval in the determination of equipment drift.

REFERENCES

1. SAR, Section 8.3.1.
2. SAR, Chapter 6 and 14.
3. 10 CFR 50.36.
4. ANO-1 Design Guide, IDG-001, "Instrument Loop Error Analysis and Setpoint Methodology Manual."

B 3.3 INSTRUMENTATION

B 3.3.9 Source Range Neutron Flux

BASES

BACKGROUND

The source range neutron flux channels provide the operator with an indication of the approach to criticality. These channels also provide the operator with a flux indication that reveals changes in reactivity.

The source range instrumentation has two redundant count rate channels originating in two high sensitivity fission chambers. Two source range detectors are externally located on opposite sides of the core. These channels are used over a counting range of 0.1 cps to 1E5 cps and are displayed on the operator's control console in terms of log count rate. The channels also measure the rate of change of the neutron flux level, which is displayed for the operator in terms of startup rate from -1 decades to +7 decades per minute. An interlock provides a control rod withdraw "inhibit" on a high startup rate of +2 decades per minute in either channel. This interlock is bypassed when the intermediate range neutron flux channels reach 1E-9 amps or power range neutron flux channels reach 10% RTP.

APPLICABLE SAFETY ANALYSES

The source range neutron flux channels are necessary to monitor core reactivity changes. They are the primary means for detecting reactivity changes and triggering operator actions to respond to reactivity transients initiated from conditions in which the Reactor Protection System (RPS) is not required to be OPERABLE. They also trigger operator actions to anticipate RPS actuation in the event of reactivity transients starting from shutdown or low power conditions.

The source range neutron flux channels satisfy Criterion 4 of 10 CFR 50.36 (Ref. 1).

LCO

One source range neutron flux channel shall be OPERABLE to provide the operator with source range neutron instrumentation. The source range instrumentation provides the primary power indication at ≤ 1E-10 amp on intermediate range instrumentation and must remain OPERABLE for the operator to continue increasing power.

APPLICABILITY

One source range neutron flux channel shall be OPERABLE in MODE 2 to provide indication during an approach to criticality. Neutron flux level is sufficient for monitoring on the intermediate range and on the power range instrumentation prior to entering MODE 1; therefore, source range instrumentation is not required in MODE 1.

In MODES 3, 4, and 5, source range neutron flux instrumentation shall be OPERABLE to provide the operator with a means of monitoring neutron flux and to provide an early indication of reactivity changes.

The requirements for source range neutron flux instrumentation during MODE 6 refueling operations are addressed in LCO 3.9.2, "Nuclear Instrumentation."

ACTIONS

A.1, A.2, A.3, and A.4

With the required source range neutron flux channel inoperable with ≤ 1E-10 amp on the intermediate range neutron flux instrumentation, the operators must take actions to limit the possibilities for adding positive reactivity. This is done by immediately suspending positive reactivity additions, initiating action to insert all CONTROL RODS, and opening the control rod drive trip breakers within 1 hour. RCS temperature changes are permitted, however, provided the effects of such temperature changes are accounted for in the SDM calculations. Periodic SDM verification is then required to provide a means for detecting the slow reactivity changes that could be caused by mechanisms other than CONTROL ROD withdrawal or operations involving positive reactivity changes. Since the source range instrumentation provides the only reliable direct indication of power in these MODES, the operators must continue to verify the SDM every 12 hours until at least one channel of the source range instrumentation is returned to OPERABLE status. Required Action A.1, Required Action A.2, and Required Action A.3 preclude rapid positive reactivity additions. The 1 hour Completion Time for Required Action A.3 and Required Action A.4 provides sufficient time for operators to accomplish the actions. The 12 hour Frequency for performing the SDM verification provides reasonable assurance that the reactivity changes possible with CONTROL RODS inserted are detected before SDM limits are challenged.

If no indication of intermediate range flux is available, these Required Actions are also appropriate.

ACTIONS (continued)

B.1

With > 1E-10 amp in MODE 2, 3, 4, or 5 on the intermediate range neutron flux instrumentation, continued operation is allowed with the required source range neutron flux channel inoperable. The ability to continue operation is justified because the instrumentation does not provide a safety function during high power operation. However, actions are initiated within 1 hour to restore the required channel to OPERABLE status for future availability. The Completion Time of 1 hour is sufficient to initiate the action. The action must continue until the channel is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1

Performance of the CHANNEL CHECK once every 12 hours provides reasonable assurance of prompt identification of a gross failure of instrumentation. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to the same parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of factors including channel instrument uncertainties. If a channel is outside the criteria, it may be an indication that the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction.

The agreement criteria includes an expectation of one decade of overlap when transitioning between neutron flux instrumentation. For example, during a power reduction near the bottom of the scale for the intermediate range monitors, a source range monitor reading is expected with at least one decade overlap. Without such an overlap, the source range monitors are considered inoperable unless it is clear that an intermediate range monitor inoperability is responsible for the lack of the expected overlap.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.9.1 (continued)

The Frequency is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channel. When operating with only one channel OPERABLE, CHANNEL CHECK is still required. However, in this condition, a redundant source range may not be available for comparison. CHANNEL CHECK may still be performed via comparison with intermediate range detectors, if available, and verification that the OPERABLE source range channel is energized and indicating a value consistent with current unit status.

SR 3.3.9.2

For a source range neutron flux channel, CHANNEL CALIBRATION is a complete check and readjustment of the channel from the preamplifier input to the indicator. This test verifies the channel responds to measured parameters within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel at a setpoint which accounts for instrument drift to ensure that the instrument channel remains operational between successive tests.

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult, and there is no adjustment that can be made to the detectors. Furthermore, adjustment of the detectors is unnecessary because they are passive devices with minimal drift. Finally, the detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output.

The Frequency of 18 months is based on demonstrated instrument CHANNEL CALIBRATION reliability over an 18 month interval, such that the instrument is not adversely affected by drift.

REFERENCES

1. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.10 Intermediate Range Neutron Flux

BASES

BACKGROUND

The intermediate range neutron flux channels provide the operator with an indication of reactor power at higher power levels than the source range instrumentation and lower power levels than the power range instrumentation.

The intermediate range instrumentation has two channels originating in two gamma compensated ion chambers. Each channel provides eight decades of flux level information in terms of the log of ion chamber current from 1E-11 amp to 1E-3 amp. The channels also measure the rate of change of the neutron flux level, which is displayed for the operator in terms of startup rate from -0.5 decades to +5 decades per minute. A high startup rate of +3 decades per minute in either channel will initiate a control rod withdrawal inhibit while below 10% RTP.

The intermediate range compensated ion chambers are of the electrically adjustable gamma compensating type. Each detector has a separate adjustable high voltage power supply and an adjustable compensating voltage supply.

APPLICABLE SAFETY ANALYSES

Intermediate range neutron flux channels are necessary to monitor core reactivity changes and provide the primary indication to trigger operator actions to anticipate Reactor Protection System actuation in the event of reactivity transients starting from low power conditions.

The intermediate range neutron flux channels satisfy Criterion 4 of 10 CFR 50.36 (Ref. 1).

LCO

One intermediate range neutron flux instrumentation channel shall be OPERABLE to provide the operator with neutron flux indication. This enables operators to control the increase in power and to detect neutron flux transients. This indication is used until the power range instrumentation is on scale. Violation of this requirement could prevent the operator from detecting and controlling neutron flux transients that could result in reactor trip during power escalation.

APPLICABILITY

The required intermediate range neutron flux channel shall be OPERABLE in MODE 2 and in MODES 3, 4, and 5 with any control rod drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

The intermediate range instrumentation is designed to detect power changes when the power range and source range instrumentation cannot provide reliable indications, e.g., during initial criticality and power escalation. Since those conditions can exist in, or propagate from, all of these MODES, the intermediate range instrumentation must be OPERABLE.

ACTIONS

A.1 and A.2

With the required intermediate range neutron flux channel inoperable when THERMAL POWER is ≤ 5% RTP, the operators must place the reactor in the next lowest condition for which the intermediate range instrumentation is not required. This involves providing power level indication on the source range instrumentation by immediately suspending operations involving positive reactivity changes and, within 1 hour, placing the reactor in the tripped condition with the CRD trip breakers open. RCS temperature changes are permitted provided the effects of such changes are accounted for in the SDM caculations. The Completion Times are based on unit operating experience and allow the operators sufficient time to manually insert the CONTROL RODS prior to opening the CRD breakers.

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1

Performance of the CHANNEL CHECK once every 12 hours provides reasonable assurance of prompt identification that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to the same parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.10.1 (continued)

Agreement criteria are determined by the unit staff based on a combination of factors including channel instrument uncertainties. If a channel is outside the criteria, it may be an indication that the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. Off scale low current loop channels are verified, where practical to be reading at the bottom of the range and not failed low.

The agreement criteria includes an expectation of one decade of overlap when transitioning between neutron flux instrumentation. For example, during a power increase near the top of the scale for the source range monitors, an intermediate range monitor reading is expected with at least one decade overlap. Without such an overlap, the intermediate range monitors are considered inoperable unless it is clear that a source range monitor inoperability is responsible for the lack of the expected overlap. Further, during a power reduction near the bottom of the scale for the power range monitors, an intermediate range monitor reading is expected with at least one decade overlap. Without such an overlap, the intermediate range monitors are considered inoperable unless it is clear that a power range monitor inoperability is responsible for the lack of the expected overlap.

The Frequency, about once every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channel.

When operating with only one channel OPERABLE, CHANNEL CHECK is still required. However, in this condition, a redundant intermediate range is not available for comparison. CHANNEL CHECK may still be performed via comparison with power or source range detectors, if available, and verification that the OPERABLE intermediate range channel is energized and indicates a value consistent with current unit status.

SR 3.3.10.2

A CHANNEL FUNCTIONAL TEST, of the required intermediate range instrument channel, verifies proper operation of the channel each 31 days. Monthly testing provides reasonable assurance that the instrument channel will function, if required, to provide indication during MODE 2 and during unanticipated reactivity excursions from MODES 3, 4, or 5.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.10.3

For intermediate range neutron flux channels, CHANNEL CALIBRATION is a complete check and readjustment of the channels, from the preamplifier input to the indicators. This test verifies the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel at a setpoint which accounts for instrument drift to ensure that the instrument channel remains operational between successive tests.

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult. In addition, the detectors are of simple construction, and any failures in the detectors will be apparent as a change in channel output. The Frequency is based on operating experience and consistency with the typical industry refueling cycle and is justified by demonstrated instrument reliability over an 18 month interval such that the instrument is not adversely affected by drift.

REFERENCES

1. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.11 Emergency Feedwater Initiation and Control (EFIC) Instrumentation

BASES

BACKGROUND

The EFIC System instrumentation is designed to protect against the consequences of a simultaneous blowdown of both steam generators. Steam generator (SG) isolation is actuated to protect the core during an overcooling condition upon a main steam or feedwater line rupture. The Emergency Feedwater (EFW) System is actuated to protect the core during an overheating condition upon a loss of main feedwater or a loss of primary side forced circulation (loss of all four reactor coolant pumps). In addition, EFIC controls the EFW flow rate to the SG(s) to control SG level and minimize overcooling. EFIC also selects the appropriate SG(s) under conditions of steam line break or main feedwater or emergency feedwater line break downstream of the last check valve, and provides for isolation of the main steam and main feedwater lines of a depressurized steam generator. The EFIC Functions that are supported and the parameters that are needed for each of these Functions are described next.

The EFIC instrumentation contains devices and circuitry that generate the following signals when monitored variables reach levels that are indicative of conditions requiring protective actions.

a. EFW Initiation;

b. EFW Vector Valve Control; and

c. Main Steam Line Isolation.

EFW is initiated to restore a source of cooling water to the secondary system when conditions indicate that the normal source of feedwater is insufficient to continue heat removal. The two indications used for this are the loss of both MFW pumps and a low level in the steam generator (SG). Also, EFW is initiated when action is being taken to isolate the MFW from the SG during conditions of uncontrolled depressurizations. This is done by initiating EFW when steam pressure reaches the low SG pressure setpoint. Also, EFW is initiated when the primary system experiences a total loss of forced circulation. This initiation, on the loss of all reactor coolant pumps (RCPs), ensures the EFW is available to raise SG levels to promote natural circulation cooling.

The EFIC System initiates EFW when an Engineered Safeguards Actuation System (ESAS) signal is initiated on low RCS pressure or high reactor building pressure (ESAS Channels 3 and 4) in order to support heat removal following Emergency Core Cooling System (ECCS) actuation. This is a digital signal provided by the

BACKGROUND (continued)

ESAS Automatic Actuation Logic. Refer to the Bases for LCO 3.3.5, "Engineered Safeguards Actuation System (ESAS) Instrumentation," and LCO 3.3.7, "Engineered Safeguards Actuation System (ESAS) Automatic Actuation Logic," for additional discussion.

The EFIC System also initiates EFW on loss of main feedwater flow as part of the Diverse Reactor Overpressure Protection System (DROPS) which is the system provided for ANO-1 to comply with requirements to reduce risk from an anticipated transient without scram (ATWS). The DROPS consists of the Diverse Scram System (DSS) and the ATWS Mitigation System Actuation Circuitry (AMSAC) . EFW initiation for ATWS prevention and mitigation is not required by this Specification.

The EFIC System also isolates main steam and MFW to an SG that has experienced an uncontrolled depressurization. With the uncontrolled depressurization, the heat sink temperature control is lost and the heat removal rate cannot be controlled. The main steam and MFW are isolated to an SG when the steam pressure reaches a low setpoint below the normal operating point of the secondary system.

EFW initiation also enables EFIC vector logic which performs an EFW control function to preclude the delivery of fluid to a depressurized SG, thereby avoiding an uncontrolled cooling condition as long as the other SG remains pressurized. When both of the SGs are depressurized, the EFIC vector logic provides EFW flow to both SGs until a significant pressure difference between the two SGs is developed, thereby ensuring that core cooling is maintained.

Trip Setpoints and Allowable Values

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

The trip setpoints used in the bistables are based on the analytical limits stated in SAR, Chapters 7 and 14 (Refs. 2 and 3). The selection of these trip setpoints is such that adequate protection is provided when appropriate sensor and processing time delays are taken into account. The Allowable Values are conservatively adjusted with respect to the analytical limits to allow for calibration tolerances, instrumentation uncertainties, instrument drift, and environmental errors as required.

Guidance used to calculate the uncertainties associated with the trip setpoints is provided in Instrument Loop Error Analysis and Setpoint Methodology Manual Design Guide, IDG-001 (Ref. 4). The explicit uncertainties are addressed in the design calculations as required. The trip setpoint entered into the bistable may be more conservative than that specified by the Allowable Value to account for changes in instrument error detectable by a CHANNEL FUNCTIONAL TEST. A channel is inoperable if its as-found trip setpoint is not within its required Allowable Value.

BACKGROUND (continued)

Setpoints in accordance with the Allowable Value in conjunction with the LCOs and administrative controls ensure that the consequences of Design Basis Accidents (DBAs) are acceptable, providing the unit is operated from within the LCOs at the onset of the DBA, and that the equipment functions as analyzed.

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

Actuation Logic

SAR, Section 7.1.4 (Ref. 2), describes the EFIC EFW Initiation logic operation.

Each EFIC train actuates on a one-out-of-two taken twice combination of trip signals from the instrumentation channels. Each EFIC channel can issue an initiate command, but an EFIC actuation will take place only if at least two channels issue initiate commands. For the EFW Initiation and Main Steam Line Isolation functions, the one-out-of-two taken twice logic combinations are transposed between trains so that failure of two channels prevents actuation of, at most, one train.

More detailed descriptions of the EFIC instrumentation are provided below.

1. EFW Initiation

   Figure 10-2, Sheet 4, SAR, Chapter 10 (Ref. 5), illustrates each channel of the EFIC EFW Initiation Function. The individual instrumentation channels that serve EFIC EFW Initiation Function are discussed next.

   a. Loss of MFW Pumps (Control Oil Pressure)

      Loss of both MFW Pumps is one of the six parameters within the EFIC System that automatically initiates EFW. The MFW Pump status instrumentation, and associated bypasses, are internal to the Reactor Protection System (RPS). For RPS, loss of MFW Pumps is detected by MFW Pump turbine control oil pressure. Each RPS channel receives MFW Pump status information from one of four pressure switches per pump. If both switches in a single channel trip (one from each pump), the associated RPS channel trips. Each RPS channel provides a contact input into its associated EFIC channel representative of both MFW Pumps tripped. At least two EFIC channels in trip are required for EFW Initiation. This Function is automatically bypassed when THERMAL POWER is < 10% RTP and the bypass is automatically removed when THERMAL POWER is ≥ 10% RTP. The bypass functions occur internal to the RPS, i.e., prior to input to the EFIC System. This parameter value (i.e., 10% RTP) is a nominal value consistent with the requirements of LCO 3.3.1, "RPS Instrumentation."

BACKGROUND (continued)

Loss of both MFW Pumps was chosen as an EFW automatic initiating parameter because it is a direct and immediate indicator of loss of MFW.

b. SG Level – Low

Four EFIC dedicated low range level transmitters per SG are used to generate the signals used for detection for low level conditions for EFW actuation. There is one transmitter for each of the four channels A, B, C, and D. At least two channels are required to initiate EFW. SG Level - Low was chosen as an EFW automatic initiating parameter because it indicates that the normal feedwater source may be insufficient to meet the heat removal requirements.

Signals from channels A and B are also used to control SG level at approximately 31 inches when one or more RCPs are operating. This parameter is referenced to the top of the lower tube sheet.

c. SG Pressure – Low

Four transmitters per SG (one transmitter per channel) provide the EFIC System with channels A through D of SG Pressure - Low. These are the same transmitters used by the Main Steam Line Isolation Function. When the SG pressure at the transmitter drops below the bistable Allowable Value of 584.2 psig on a given channel, an EFW Initiation signal is sent to the automatic actuation logic. At least two channels are required to initiate EFW and main steam line isolation. The Allowable Value of ≥ 584.2 psig includes consideration for instrumentation error and an allowance for margin. Allowances for instrument drift and additional margin are included in the trip setpoint. The low pressure Function may be manually bypassed when either SG is less than 750 psig. If both SG pressure inputs exceed 750 psig, the EFIC channel bypass is automatically removed. The low pressure operational bypass allows for normal cooldown without EFIC actuation. The parameter value (i.e., 750 psig) is a nominal value. Should the channel remain bypassed above 750 psig, the channel is considered inoperable and appropriate conditions are entered. Failure of the automatic bypass removal feature alone or the inability to bypass a channel when below 750 psig does not constitute channel inoperability. The automatic bypass removal feature is verified during the monthly CHANNEL FUNCTIONAL TEST.

SG Pressure - Low is a primary indication and actuation signal for a steam line break or feedwater line break (non-design basis transient). For a small break, which does not depressurize the SG or takes a long time to depressurize the SG, automatic actuation is not required. The operator has time to diagnose the problem and take the appropriate actions.

BACKGROUND (continued)

d. RCP Status

A loss of power to all four RCPs is an indication of a pending loss of forced flow in the Reactor Coolant System. These signals are input into the four channels of EFIC.

When at least two channels issue initiate commands based on loss of all RCPs, the EFIC System will automatically actuate EFW and control the level at approximately 312 inches in the SG. This higher level provides a thermal center in the SG at a higher elevation than that of the reactor to enhance natural circulation of the reactor coolant. This parameter is referenced to the top of the lower tube sheet.

To allow heatup and cooldown operations without actuation, a bypass permissive of 10% RTP is used. The 10% bypass permissive was chosen because it was an available, qualified Class 1E signal at the time the EFIC System was designed. When the first RCP is started, the "loss of four RCPs" initiation signal may be manually reset. If the bypass is not manually reset, it will be automatically reset when the unit reaches 10% power. Should the channel remain bypassed when ≥ 10% RTP, the channel is considered inoperable and appropriate conditions are entered. Failure of the automatic bypass removal feature alone or the inability to bypass a channel when below 10% RTP does not constitute channel inoperability. The automatic bypass removal feature is verified during the monthly CHANNEL FUNCTIONAL TEST.

During cooldown, the bypass may be inserted at any time the power has been reduced below 10% RTP. However, for most operating conditions, this trip function remains active until after the Decay Heat Removal System has been initiated and the system is ready for the last RCP to be tripped. This trip function must be bypassed prior to stopping the last RCP. This parameter value (i.e., 10% RTP) is a nominal value consistent with the requirements of LCO 3.3.1, "RPS Instrumentation."

e. ESAS

The EFIC System initiates EFW when an ESAS signal is initiated on low RCS pressure or high reactor building pressure (ESAS Channels 3 and 4) in order to support heat removal following ECCS actuation. This is a digital signal provided by the ESAS Automatic Actuation Logic. Refer to the Bases for LCO 3.3.5, "Engineered Safeguards Actuation System (ESAS) Instrumentation," and LCO 3.3.7, "Engineered Safeguards Actuation System (ESAS) Automatic Actuation Logic," for additional discussion.

BACKGROUND (continued)

f. DROPS

The EFIC System also initiates EFW on loss of main feedwater flow as part of the DROPS which is the system provided for ANO-1 to comply with requirements to reduce risk from an ATWS. The DROPS consists of the Diverse Scram System (DSS) and the ATWS Mitigation System Actuation Circuitry (AMSAC) . EFW initiation for ATWS prevention and mitigation is not required by this Specification.

2. EFW Vector Valve Control

Figure 10-2, Sheet 4, SAR, Chapter 10 (Ref. 5), illustrates the EFIC EFW Vector Valve Control inputs to the EFIC Vector Logic (See Bases for LCO 3.3.14, "EFIC Vector Logic"). The function of the EFW vector logic is to determine whether EFW should not be fed to one or the other SG once enabled by the EFW Initiation Function. This is to preclude the continued addition of EFW to a depressurized SG and, thus, to minimize the overcooling effects.

Each set of vector logic receives SG pressure information from bistables located in the input logic of the same EFIC channel. The pressure information received is:

a. SG A pressure less than 584.2 psig;

b. SG B pressure less than 584.2 psig;

c. SG A pressure 100 psid greater than SG B pressure; and

d. SG B pressure 100 psid greater than SG A pressure.

The Allowable Value of ≥ 584.2 psig includes consideration for instrumentation error and an allowance for margin. Allowances for instrument drift and additional margin are included in the trip setpoint. The 100 psid value is considered to be a nominal value.

The vector logic outputs are in a neutral state until enabled by the train A or B trip logics. When enabled, the vector logic can issue close commands to the EFW control valves and open or closed commands to the EFW isolation valves per the selected channel assignments. The level control module provides input to the flow controllers which control the position of the EFW control valves.

Each vector logic may isolate EFW to one SG or the other, never both.

The valve open or close commands are determined by the relative values of SG pressures as discussed in the Bases for LCO 3.3.14.

BACKGROUND (continued)

3. Main Steam Line Isolation

SAR, Section 7.1.4 (Ref. 2) describes one channel of the EFIC Main Steam Line Isolation logic. Four pressure transmitters (one transmitter per channel) per SG provide EFIC with channels A through D logic of SG pressure. The channels are as described for EFW Initiation mentioned earlier.

Bypass

One of the four initiation channels can be put into "maintenance bypass." Bypassing one initiation channel isolates that channel's signal to the functions fed from initiation channel but does not bypass the trip logic within the actuation train. An interlock feature prevents bypassing more than one channel at a time. In addition, since the EFIC System receives signals from the RPS, the maintenance bypass from the RPS is interlocked with the EFIC System. If one channel of the RPS is in maintenance bypass, only the corresponding channel of the EFIC may be bypassed (e.g., channel A, RPS, and channel A, EFIC). This ensures that only the corresponding channels of the EFIC and RPS are placed in maintenance bypass at the same time.

EFIC channel maintenance bypass does not bypass EFW Initiation from ESAS. The EFIC EFW initiation from ESAS is, however, bypassed when its associated ESAS channel is bypassed.

The operational bypass provisions were discussed as part of the individual Functions described earlier.

The EFIC System is designed to perform its intended EFW Initiation and Main Steam Line Isolation function with one channel in maintenance bypass (in effect, inoperable) concurrent with a postulated single failure in any one of the remaining channels. This is in compliance with IEEE-279-1971 (Ref. 6).

APPLICABLE SAFETY ANALYSES

1. EFW Initiation

Although loss of both MFW pumps is a direct and immediate indicator of loss of MFW, other scenarios such as valve closures could potentially cause loss of feedwater. As part of the post-TMI review, a loss of main feedwater was analyzed (Ref. 3). The EFIC System response for a loss of MFW conservatively assumes the actuation of EFW on low SG level. If the loss of feedwater is due to loss of MFW pumps, EFW will be actuated earlier than it would on low SG level, which will increase the SG heat transfer capability and will lessen the severity of the transient.

APPLICABLE SAFETY ANALYSES (continued)

The basis for initiation of the EFW systems is a loss of MFW. For this analysis, SG Level - Low is the most conservative parameter from which to automatically initiate EFW since this yields the least SG inventory available for heat removal. SG Level - Low would be an indicator of any event involving a loss of SG secondary side inventory heat removal capability.

SG Pressure - Low is a primary indication and provides an actuation signal for a SLB. In the SLB analyses, SAR Section 14.2.2.1 (Ref. 3), EFIC initiation occurs; however, no EFW flow occurred because level did not reach the SG Level - Low setpoint.

Loss of four RCPs is a primary indicator of the need for emergency feedwater (EFW) for the loss of electric power analysis, SAR Section 14.1.2.8 (Ref. 3).

The SAR SBLOCA analyses, SAR Section 14.2.2.5 (Ref. 3), assume initiation of EFW based on concurrent loss of offsite power and the resultant loss of four RCPs. Initiation of EFW would also occur when an ESAS signal is generated on low RCS pressure or high reactor building pressure (ESAS Channels 3 or 4) in order to support heat removal following ECCS actuation, however, these are considered backup initiation responses.

2. <u>EFW Vector Valve Control</u>

The SAR SLB analyses, SAR Section 14.2.2.1 (Ref. 3), consider isolation of the affected SG as a function automatically performed by the EFIC System. The EFIC Vector Logic utilizes the EFW Vector Valve Control Functions (i.e., SG Pressure - Low and SG Differential Pressure - High) to determine which steam generator is associated with the rupture and provide appropriate isolation.

3. <u>Main Steam Line Isolation</u>

The SAR SLB analyses, SAR Section 14.2.2.1 (Ref. 3), assume actuation of the Main Steam Line Isolation on SG Pressure - Low, initiating closure of the main steam isolation valves and the main feedwater isolation valves. The steam generator in the steam loop associated with the rupture blows dry after feedwater isolation. EFW flow is available to the unaffected steam generator to preserve the availability of an RCS heat sink.

In MODE 1, the EFIC System satisfies Criterion 3 of 10 CFR 50.36 (Ref. 7). In MODES 2 and 3, the EFIC System satisfies Criterion 4 of 10 CFR 50.36 since there are no specific safety analyses that credit the EFIC system for operation at less than full rated power.

LCO

All instrumentation performing an EFIC System Function in Table 3.3.11-1 shall be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable.

Four channels are required OPERABLE for all EFIC Functions. Each EFIC instrumentation channel is considered to include the sensors and measurement channels for each Function, the operational bypass switches, and permissives. Failures that disable the capability to place a channel in operational bypass, but which do not disable the trip Function, do not render the protection channel inoperable.

EFIC initiation function values for the bypass removal functions are specified in terms of applicability limits (i.e., identified in the Applicable MODES or Other Specified Conditions column of Table 3.3.11-1) for the associated trip Function. Trip setpoints are specified in the setpoint calculations or calibration procedures. The trip setpoints are selected to ensure the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the Allowable Value if the bistable is performing as required. Guidance used to calculate the uncertainties associated with the trip setpoints is provided in Reference 4.

The Bases for the LCO requirements of each specific EFIC Function are discussed next.

<u>Loss of MFW Pumps</u>

Four EFIC channels for Loss of MFW Pumps shall be OPERABLE. This ensures that upon the loss of both MFW pumps, EFW will be automatically initiated. This Function is provided as a direct digital input from the RPS and includes a bypass enable and removal function.

<u>SG Level – Low</u>

Four EFIC dedicated low range level transmitters per SG shall be OPERABLE with an SG Level - Low actuation Allowable Value of ≥ 11.1 inches, to generate the signals used for detection for low level conditions for EFW Initiation. This parameter is referenced to the top of the lower tube sheet and includes consideration for instrumentation error and an allowance for margin. Allowances for instrument drift and additional margin are included in the trip setpoint. There is one transmitter for each of the four channels A, B, C, and D. The signals are also used after EFW is actuated to control level at approximately 31 inches when one or more RCPs are in operation. In the determination of the low level setpoint, it is desired to place the setpoint as low as possible, considering instrument errors, to give the maximum operational margin between the integrated control system setpoint and the EFW Initiation setpoint. This will minimize spurious or unwanted initiation of EFW. Credit is only taken for low level actuation for those transients which do not involve a degraded environment. Therefore, normal environment errors only are used for determining the SG Level - Low level setpoint. This parameter is referenced to the top of the lower tube sheet.

LCO (continued)

SG Pressure - Low

Four EFIC channels per SG shall be OPERABLE with an SG low pressure actuation Allowable Value of ≥ 584.2 psig. The setpoint is chosen to avoid actuation under transient conditions not requiring secondary system isolation, preferring to maintain a steaming path to the condenser, if possible. Small break LOCA analyses have indicated minimum secondary system pressures of greater than the above setpoint. The SG Pressure - Low Function includes a bypass enable and removal function. The bypass removal value is chosen to allow sufficient operating margin for the operator to bypass when cooling down. The above Allowable Value (i.e., 584.2 psig) includes consideration for instrumentation error and an allowance for margin. Allowances for instrument drift and additional margin are included in the trip setpoint.

SG Differential Pressure - High

Four EFIC channels for SG differential pressure shall be OPERABLE. This Function ensures that automatic EFW isolation to a depressurized SG occurs. The MSLB analysis assumes the depressurized SG is isolated when a differential pressure of 150 psid is detected. The in-plant setpoint is conservatively chosen to protect the MSLB assumptions.

RCP Status

Four EFIC channels for RCP status shall be OPERABLE. This ensures that upon the loss of four RCPs, EFW will be automatically initiated with the EFW control level automatically raised to approximately 312 inches, providing a higher SG level for establishing and maintaining natural circulation conditions. No setpoint is specified since the status indication as used by EFIC is binary in nature. The RCP Status Function includes a bypass enable and removal function from the RPS. The above parameter value (i.e., 312 inches) does contain an allowance for instrument error. This parameter is referenced to the top of the lower tube sheet.

APPLICABILITY

The EFIC System instrumentation Functions shall be OPERABLE in accordance with Table 3.3.11-1. Each Function has its own requirements that are conservative with respect to the specific accidents and conditions for which it is designed to mitigate the consequences. The parameter values provided as part of the Applicability do contain an allowance for instrument error.

The initiation of EFW on the Loss of MFW Pumps shall only be required when the unit is ≥ 10% RTP. Below 10% RTP, the EFW Initiation on low SG level will mitigate primary system overheating.

APPLICABILITY (continued)

EFW Initiation on low SG level shall be OPERABLE in MODES 1, 2, and 3 which are conditions during which the SG is required for heat removal.

To avoid automatic actuation of the EFW pumps during normal heatup and cooldown transients, the low SG pressure Function can be bypassed at or below a secondary pressure of 750 psig during MODE 3 operation.

The EFW System Initiation on loss of all RCPs Function shall be operable at ≥ 10% RTP. It is possible to bypass the Function below 10% RTP; however, for most cases, the Function is kept in service until the unit is placed on the Decay Heat Removal System. To prevent inadvertent actuation of the EFW pumps, it must be bypassed prior to stopping the last RCP.

The Main Steam Line Isolation and EFW Vector Valve Control Functions shall be OPERABLE in MODES 1 and 2, and MODE 3 with SG pressure ≥ 750 psig because the SG inventory can contribute significantly to the reactor building peak pressure with a secondary side break. Both the normal feedwater and the EFW must be able to be isolated on each SG to limit overcooling of the primary and to limit mass and energy releases to the reactor building. Once the SG pressures have decreased below 750 psig the energy level is low and the secondary side feedwater flow rate is low or nonexistent. Also, the primary system temperatures are typically too low to allow the SGs to effectively remove energy, or are sufficiently low to allow for operator action. Therefore, EFIC instrumentation is not required to be OPERABLE.

ACTIONS

If a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or any of the transmitter, signal processing electronics, or EFIC channel cabinet modules are found inoperable, then all affected Functions provided by that channel must be declared inoperable and the unit must enter the Conditions for the particular protection Function affected.

A Note has been added to the ACTIONS indicating that a separate Condition entry is allowed for each Function.

A.1

Condition A applies to failures of a single EFW Initiation or Main Steam Line Isolation instrumentation channel. This includes failure of a common instrumentation channel in any combination of the Functions.

ACTIONS (continued)

A.1 (continued)

With one channel inoperable in one or more EFW Initiation or Main Steam Line Isolation Functions listed in Table 3.3.11-1, the channel(s) must be placed in bypass or trip within 1 hour. This Condition applies to failures that occur in a single channel, e.g., channel A, which when bypassed will remove initiate Functions within the channel from service. Since the RPS and EFIC channels are interlocked, only the corresponding channel in each system may be bypassed at any time. This feature is ensured by an electrical interlock. If testing of another channel in either the EFIC or RPS is required, the EFIC channel must be placed in trip to allow the other channel to be bypassed. With the channel in trip, the resultant logic is one-out-of-two. The Completion Time of 1 hour is adequate to perform Required Action A.1.

B.1 and B.2

Condition B applies to a situation where two instrumentation channels of the same protection functions of EFW Initiation or Main Steam Line Isolation instrumentation are inoperable. For example, Condition B applies if channel A and B of the EFW Initiation Function are inoperable.

With two EFW Initiation or Main Steam Line Isolation protection channels inoperable, one channel must be placed in bypass (Required Action B.1). Bypassing one of the remaining OPERABLE channels is not possible due to system interlocks. Therefore, the second channel must be tripped (Required Action B.2) to prevent a single failure from causing loss of the EFIC Function. The Completion Times of 1 hour are adequate to perform the Required Actions.

C.1

The function of the EFW Vector Valve Control is to meet the single-failure criterion while being able to provide EFW on demand and isolate an SG when required. These conflicting requirements result in the necessity for two valves in series, in parallel with two valves in series, and a four channel valve command system. Refer to LCO 3.3.14, "Emergency Feedwater Initiation and Control (EFIC) Vector Logic."

With one EFW Vector Valve Control channel inoperable, the system cannot meet the single-failure criterion and still meet the dual functional criteria described earlier. This condition is analogous to having one EFW train inoperable. Therefore, when one vector valve control channel is inoperable, the channel must be restored to OPERABLE status (Required Action C.1) within 72 hours, which is consistent with the Completion Time associated with the loss of one train of EFW.

ACTIONS (continued)

D.1, D.2, E.1, F.1, F.2

If the Required Actions and associated Completion Times are not met, the unit must be placed in a MODE or condition in which the requirement does not apply. This is done by placing the unit in a nonapplicable MODE for the particular Function. The nonapplicable MODE is less than 10% RTP for Functions 1.a and 1.d, MODE 4 for Function 1.b, and MODE 3 with SG pressure less than 750 psig for all other Functions. In addition, for Function 3.a, once the unit is in MODE 3, a nonapplicable condition may be achieved by closing and deactivating the valves associated with the Main Steam Line Isolation Function. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

A Note indicates that the SRs for each EFIC instrumentation Function are identified in the SRs column of Table 3.3.11-1. Individual EFIC subgroup relays must also be tested, one at a time, to verify the individual EFIC components will actuate when required. Some components cannot be tested at power since their actuation might lead to unit trip or equipment damage. These are specifically identified and must be tested when shut down. The various SRs account for individual functional differences and for test frequencies applicable specifically to the Functions listed in Table 3.3.11-1. The operational bypasses associated with each EFIC instrumentation channel are also subject to these SRs to ensure OPERABILITY of the EFIC instrumentation channel.

SR 3.3.11.1

Performance of the CHANNEL CHECK once every 12 hours provides reasonable assurance for prompt identification of a gross failure of instrumentation.
A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of factors including channel instrument uncertainties. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.11.1 (continued)

indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are verified, where practical, to be reading at the bottom of the range and not failed downscale.

The Frequency is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.

SR 3.3.11.2

A CHANNEL FUNCTIONAL TEST verifies the function of the automatic bypass removal feature, required trip, interlock, and alarm functions of the channel. Setpoints for trip functions must be found within the Allowable Value. (Note that the values for the bypass removal functions are identified in the Applicable MODES or Other Specified Condition column of Table 3.3.11-1 as limits on applicability for the trip Functions.) Any setpoint adjustment shall be consistent with the assumptions of the current setpoint analysis.

The Frequency of 31 days is based on unit operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any 31 day interval is a rare event.

SR 3.3.11.3

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The test verifies the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channels adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis.

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

REFERENCES

1. 10 CFR 50.62.

2. SAR, Chapter 7.

3. SAR, Chapter 14.

4. Instrument Loop Error Analysis and Setpoint Methodology Manual, Design Guide, IDG-001.

5. SAR, Chapter 10, Figure 10-2, Sheet 4.

6. IEEE-279-1971, April 1972.

7. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.12 Emergency Feedwater Initiation and Control (EFIC) Manual Initiation

BASES

BACKGROUND

The EFIC manual initiation capability provides the operator with the capability to actuate EFIC Functions from the control room in the absence of any other initiation condition. Manually actuated Functions include Main Steam Line Isolation for steam generator (SG) A, Main Steam Line Isolation for SG B, and Emergency Feedwater (EFW) Actuation. These Functions are provided in the event the operator determines that an EFIC Function is needed prior to automatic actuation or in the event that EFIC does not automatically actuate when required. These are backup Functions to those performed automatically by EFIC.

The manual actuation of these functions may be performed from the Remote Switch Matrix, located on the main control boards, or from the manual actuation trip switches located on the EFIC control cabinets in the control room. The required manual actuation logic within each train consists of two manual switches (one for Trip Bus 1 and one for Trip Bus 2). When one manual trip switch is depressed, a half trip occurs. When both manual trip switches are depressed, a full trip of the train actuation occurs for that particular Function. The Remote Switch Matrix and the EFIC control cabinet trip switches perform parallel functions and, therefore, any combination of switches depressed within a train that energizes both Trip Bus 1 and Trip Bus 2 for a given Function will result in an actuation of that Function. The use of two manual trip switches for each train of actuation logic allows testing without actuating the end devices and also reduces the possibility of accidental manual actuations.

The EFIC manual initiation circuitry satisfies the manual initiation and single-failure criterion requirements of IEEE-279-1971 (Ref. 1).

APPLICABLE SAFETY ANALYSES

EFIC Functions credited in the safety analysis are automatic. However, the manual initiation Functions are required by design as backups to the automatic initiation Functions and allow operators to actuate EFW or Main Steam Line Isolation whenever these Functions are needed. Furthermore, the manual initiation of EFW and Main Steam Line Isolation may be specified in unit operating procedures.

The EFIC manual initiation functions satisfy Criterion 4 of 10 CFR 50.36 (Ref. 2).

LCO

Instrumentation on the main control board performing an EFIC manual initiation Function shall be OPERABLE.

Two manual initiation switches per actuation train (Train A and Train B) of each Function (A and B Main Steam Line Isolation, and EFW Actuation) are required to be OPERABLE. This requirement may be satisfied by the manual trip switches located on the Remote Switch Matrix on the main control board, by the trip switches located on the EFIC control cabinets, or by any combination of switches located on the Remote Switch Matrix and the EFIC control cabinets such that Trip Bus 1 and Trip Bus 2 are available for each EFIC Function in each of the two EFIC trains.

APPLICABILITY

The EFIC System Manual Initiation Function shall be OPERABLE when the associated EFIC Instrumentation Main Steam Line Isolation or EFW Initiation Function is required to be OPERABLE in accordance with Table 3.3.11-1. Each Function, i.e., Main Steam Line Isolation and EFW Initiation, has its own requirements that are based on the specific accidents and conditions for which it is designed to mitigate the consequences. See Bases for LCO 3.3.11, "EFIC Instrumentation," for additional discussion of each Function.

ACTIONS

A Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each EFIC manual initiation Function.

A.1

With one required manual initiation switch of one or more EFIC Function(s) inoperable in one train, the trip bus for the associated EFIC Function(s) must be placed in the tripped condition within 72 hours. With the trip bus in the tripped condition, the single-failure criterion is met. Failure to perform Required Action A.1 could allow a single failure of another switch to prevent manual actuation of at least one of the two trains. The Completion Time allotted to trip the trip bus allows the operator to take all the appropriate actions for the failed manual initiation switch and still ensure that the risk involved in operating with the failed manual initiation switch is acceptable.

ACTIONS (continued)

B.1

With both required manual initiation switches of one or more EFIC Function(s) inoperable in one train, one manual initiation switch must be restored to OPERABLE status within 72 hours. The effect for both required switches being inoperable simultaneously is the same as for the associated EFIC components for a single train being inoperable. Therefore, the 72-hour Completion Time is appropriate since it is consistent with the Completion Times of the associated system train. The trip bus associated with the remaining inoperable manual initiation switch must be placed in the tripped condition within 72 hours (Required Action A.1). With the affected trip bus in the tripped condition, the single failure criterion is met. The Completion Time allotted to restore a trip bus or place the trip bus in the tripped condition allows the operator to take all appropriate actions for the failed manual initiation switches and still ensure that the risk involved in operating with the failed manual initiation switches is acceptable.

C.1

With one or both required manual initiation switches of one or more EFIC Function(s) inoperable in both actuation trains, one actuation train for each Function must be restored to OPERABLE status within 1 hour. With the train restored, the second train must be placed in the appropriate condition within 72 hours per Required Action A.1 or B.1, as applicable. Compliance with these actions ensures the single-failure criterion is met. The Completion Time allotted to restore the train allows the operator to take all the appropriate actions for the failed train and still ensures that the risk involved in operating with the failed train is acceptable.

D.1 and D.2

If the Required Action and the associated Completion Time is not met for any EFW Initiation Function, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

E.1, E.2.1, and E.2.2

If the Required Actions and associated Completion Times are not met for the Main Steam Line Isolation Function, the unit must be placed in a MODE or condition in which the requirement does not apply. This is initiated by placing the unit in MODE 3 within 6 hours and, either reducing SG pressure to less than 750 psig, or closing and deactivating all associated valves, i.e., the valves which EFIC would close if it were to actuate while OPERABLE . The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.12.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST to ensure that the trains can perform their intended functions. However, for Main Steam Line Isolation and EFW Initiation, the test need not include actuation of the end device. This is due to the risk of a unit transient caused by the closure of valves associated with Main Steam Line Isolation or EFW Initiation during testing at power. The Frequency of 31 days is based on operating experience with regard to channel OPERABILITY that demonstrates the rarity of more than one train failing within the same 31 day interval.

REFERENCES

1. IEEE-279-1971, April 1972.
2. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.13 Emergency Feedwater Initiation and Control (EFIC) Logic

BASES

BACKGROUND

Main Steam Line Isolation

The four emergency feedwater initiation and control (EFIC) channels sensing a steam generator (SG) low outlet pressure condition input their initiate commands to the trip logic modules. SAR, Section 7.1.4 (Ref. 1), describes the Main Steam Line Isolation Logics. The trip logic modules are identified as being part of the “A” and “B” trains and are physically located in the “A” and “B” EFIC channel cabinets. Train “A” actuation logic initiates when instrumentation channel “A” or “B” initiates and channel “C” or “D” initiates, which in simplified logic is:

Train “A” actuation = (A and C) or (A and D) or (B and C) or (B and D)

Train “B” actuation logic initiates when instrumentation channel “A” or “C” initiates and channel “B” or “D” initiates, which in simplified logic is:

Train “B” actuation = (A and B) or (A and D) or (C and B) or (C and D)

Each of the two Functions (SG A Main Steam Line Isolation, and SG B Main Steam Line Isolation) has a train “A” and a train “B” of automatic actuation logic.

Both trains "A" and "B" of the SG A Main Steam Line Isolation automatic actuation logic send closure signals to the SG A Main Steam Isolation valve.

SG B Main Steam Line Isolation automatic actuation logics respond similarly for the SG B valve.

Train “A” of the SG A Main Steam Line Isolation automatic actuation logic sends closure signals to the SG A MFW isolation valves. Similarly, Train “B” of the SG B Main Steam Line Isolation automatic actuation logic sends closure signals to the SG B MFW isolation valves.

Emergency Feedwater (EFW Initiation)

The four EFIC instrumentation channels for each of the parameters being sensed input their initiate commands to the trip logic modules. SAR, Section 7.1.4 (Ref. 1), describes the EFW initiation logic. These trip logic modules are identified as being part of the “A” and “B” trains and are physically located in the "A" and "B" EFIC channel cabinets.

BACKGROUND (continued)

EFW Initiation functions use the same actuation logic combinations as Main Steam Line Isolation. EFW initiation also occurs on Engineered Safeguards Actuation System (ESAS) actuation and on Diverse Reactor Overpressure Protection System (DROPS) actuation.

EFIC automatically initiates the EFW System when any of the following conditions exist:

a. All four reactor coolant pumps are tripped;

b. Both MFW pumps are tripped and reactor power is > 10% RTP;

c. Low level in either SG;

d. Low pressure in either SG;

e. Actuation of ESAS channels 3 or 4; or

f. Actuation of DROPS channels 1 or 2.

Vector Valve Enable Logic

The EFIC System is also responsible for sending open or close signals to the EFW control and isolation valves. SAR Section 7.1.4 (Ref. 1), describes the EFIC vector logic. The vector logic outputs are in a neutral state (neither commanding open nor close) until an enable signal is received from either train "A" or "B" of EFW Initiation. The EFIC Logic monitors the train A and B EFW Initiation logics. When an EFW Initiation occurs, the vector logic is enabled to generate open or close signals to the EFW isolation valves and close signals to the EFW control valves depending on the relative values of SG pressures. The level control module provides input to the flow controllers which control the position of the EFW control valves.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analysis discussion for the Main Steam Line Isolation and EFW Initiation Functions is discussed in the Bases for LCO 3.3.11, "EFIC Instrumentation."

LCO

Two trains each of Main Steam Line Isolation and EFW Initiation logics shall be OPERABLE. There are only two trains of automatic actuation logic per Function. Therefore, violation of this LCO could result in a complete loss of the automatic Function assuming a single failure of the other train.

LCO (continued)

To be considered OPERABLE, the Main Steam Line Isolation logic must send closure signals to the associated SG main steam and MFW isolation valves when the appropriate combinations of instrument channels indicate low SG pressure.

To be considered OPERABLE, the EFW Initiation logic must send initiation signals to the EFW System when the appropriate combinations of instrument channels indicate any of the following conditions exist:

a. All four reactor coolant pumps are tripped;

b. Both MFW pumps are tripped and reactor power is > 10% RTP;

c. Low level in either SG;

d. Low pressure in either SG; or

e. Actuation of ESAS channel 3 or 4.

APPLICABILITY

The EFIC Logic shall be OPERABLE when the associated EFIC Instrumentation Main Steam Line Isolation or EFW Initiation Function is required to be OPERABLE in accordance with Table 3.3.11-1. Each Function, i.e., Main Steam Line Isolation and EFW Initiation, has its own requirements that are based on the specific accidents and conditions for which it is designed to mitigate the consequences. See Bases for LCO 3.3.11, "EFIC Instrumentation," for additional discussion of each Function.

ACTIONS

If a train is found inoperable, then all affected logic Functions provided by that train must be declared inoperable and the appropriate Condition entered for the particular protection function affected.

For this LCO, a Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each EFIC logic Function.

A.1

Condition A applies when one or more EFIC logic Functions in a single train are inoperable (i.e., train A could be inoperable for both EFIC logic Functions and Condition A would still be applicable) with all Functions in the other train OPERABLE. This Condition is equivalent to failure of one EFW and Main Steam Line Isolation train.

ACTIONS (continued)

A.1 (continued)

With one automatic actuation logic train of one or more EFIC Functions inoperable, the associated EFIC train must be restored to OPERABLE status. Since there are only two automatic actuation logic trains per EFIC Function, the condition of one train inoperable is analogous to having one train of a two train Engineered Safeguards (ES) System inoperable. The system safety function can be accomplished; however, a single failure cannot be taken. Therefore, the failed train(s) must be restored to OPERABLE status to re-establish the system's single-failure tolerance.

Condition A can be thought of as equivalent to failure of a single train of a two train safety system (e.g., the safety function can be accomplished, but a single failure cannot be taken). Thus, the Completion Time of 72 hours has been chosen to be consistent with Completion Times for restoring one inoperable ESF System train.

The EFIC System has not been analyzed for failure of both trains of the same Function. Consequently, any combination of failures in both trains A and B is not covered by Condition A and must be addressed by entry into LCO 3.0.3.

B.1 and B.2

If Required Action A.1 and its associated Completion Time is not met for the EFW Initiation Function, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

C.1, C.2.1, and C.2.2

If the Required Actions and associated Completion Times are not met for the Main Steam Line Isolation Function, the unit must be placed in a MODE or condition in which the requirement does not apply. This is initiated by placing the unit in MODE 3 within 6 hours and, either reducing SG pressure to less than 750 psig, or closing and deactivating all associated valves, i.e., the valves which EFIC would close if it were to actuate while OPERABLE. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.13.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST to ensure that the trains can perform their intended functions. This test verifies Main Steam Line Isolation and EFW Initiation automatic actuation logics are functional. This test simulates the required inputs to the logic circuit and verifies successful operation of the automatic actuation logic. The test need not include actuation of the end device. This is due to the risk of a unit transient caused by the closure of valves associated with Main Steam Line Isolation or actuation of EFW during testing at power. The Frequency of 31 days is based on operating experience with regard to channel OPERABILITY, which has demonstrated the rarity of more than one channel failing within the same 31 day interval.

REFERENCES

1. SAR, Chapter 7.

B 3.3 INSTRUMENTATION

B 3.3.14 Emergency Feedwater Initiation and Control (EFIC) Vector Logic

BASES

BACKGROUND

The function of the EFIC vector logic is to determine whether EFW should not be fed to one or the other steam generator. This is to preclude the continued addition of EFW to a depressurized steam generator (SG) and, thus, minimize the overcooling effects. Each vector logic may isolate EFW to one SG or the other, never both.

There are four sets of vector logic; one in each channel of EFIC. Each set of vector logic receives SG pressure information from bistables located in the input logic of the same EFIC channel. The pressure information received is:

a. SG "A" pressure less than 600 psig;

b. SG "B" pressure less than 600 psig;

c. SG "A" pressure 100 psid greater than SG "B" pressure; and

d. SG "B" pressure 100 psid greater than SG "A" pressure.

These values (i.e., 600 psig and 100 psid) do contain an allowance for instrument error.

Each vector logic also receives an enable signal from both EFIC train A and train B when EFW is actuated.

The vector logic develops signals for open and close control of SG "A" and "B" EFW valves.

The vector logic outputs are in a neutral state with the valves fully open until enabled by the EFW Initiation (train A or B) trip logics. When enabled, the vector logic can issue close commands to the EFW control valves and open or close commands to the EFW isolation valves per the selected channel assignments.

The valve open/close commands are determined by the relative values of steam generator pressures as follows:

	SG VALVES	
PRESSURE STATUS	"A"	"B"
If SG "A" & SG "B" > 600 psig	Open	Open
If SG "A" > 600 psig & SG "B" < 600 psig	Open	Close
If SG "A" < 600 psig & SG "B" > 600 psig	Close	Open
If SG "A" & SG "B" < 600 psig		
AND		
• SG "A" & SG "B" within 100 psid	Open	Open
• SG "A" 100 psid > SG "B"	Open	Close
• SG "B" 100 psid > SG "A"	Close	Open

APPLICABLE SAFETY ANALYSES

The SAR SLB analyses, SAR Section 14.2.2.1 (Ref. 3), consider isolation of the affected SG as a function automatically performed by the EFIC System. The EFIC Vector Logic utilizes the EFW Vector Valve Control Functions (i.e., SG Pressure—Low and SG Differential Pressure—High) to determine which steam generator is associated with the rupture and provide appropriate isolation.

Additional information relating to EFIC initiation and logics may be referenced in the Bases for LCO 3.3.11, "EFIC Instrumentation."

LCO

Four channels of the EFIC vector logic module are required to be OPERABLE. The necessity for four channels is discussed in the BASES for ACTIONS. The 600 psig and 100 psid setpoints were chosen as discussed in Specification B 3.3.11, "EFIC Instrumentation." The feed only good generator verification study assumed a differential pressure vector value of 150 psid. A 100 psid setpoint conservatively assumes a 50 psi (25 psi per pressure channel) margin for instrument error. Failure to meet this LCO results in not being able to meet the single-failure criterion. These values (i.e., 600 psig and 100 psid) do contain an allowance for instrument error.

APPLICABILITY

The EFIC Vector Logic shall be OPERABLE when the associated EFIC Instrumentation EFW Vector Valve Control Function is required to be OPERABLE in accordance with Table 3.3.11-1. The EFW Vector Valve Control Function is required to be OPERABLE in MODES 1 and 2, and in MODE 3 with SG pressure ≥ 750 psig because the SG inventory can contribute significantly to the reactor building peak pressure with a secondary side break. Both the normal feedwater and the EFW must be able to be isolated on each SG to limit overcooling of the primary and to limit mass and energy releases to the reactor building. Once the SG pressures have decreased below 750 psig, the energy level is low and the secondary side feedwater flow rate is low or nonexistent. Also, the primary system temperatures are typically too low to allow the SGs to effectively remove energy, or are sufficiently low to allow for operator action. Therefore, EFIC Vector Logic is not required to be OPERABLE in MODE 3 below 750 psig nor in MODES 4, 5, and 6.

ACTIONS

A.1

The function of the EFW control/isolation valves and the EFIC vector logic is to meet the single-failure criterion while maintaining the capability to:

a. Provide EFW on demand; and

b. Isolate an SG when required.

These conflicting requirements result in the necessity for two valves in series, in parallel with two valves in series, and a four channel valve command system.

With one channel inoperable, the system cannot meet the single-failure criterion and still meet the dual functional criteria previously described. Therefore, when one vector valve logic channel is inoperable, the channel must be restored to OPERABLE status within 72 hours. This is analogous to having one EFW train inoperable; wherein a 72 hour Completion Time is provided by the Required Actions of LCO 3.7.5, "EFW System." As such, the Completion Time of 72 hours is based on engineering judgment.

B.1 and B.2

If Required Action A.1 cannot be met within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and SG pressure must be reduced to < 750 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.14.1

SR 3.3.14.1 is the performance of a CHANNEL FUNCTIONAL TEST every 31 days. This test demonstrates that the EFIC vector logic performs its function as desired. The Frequency is based on operating experience with respect to channel OPERABILITY that demonstrates the rarity of more than one channel failing within the same 31 day interval.

REFERENCES

None.

B 3.3 INSTRUMENTATION

B 3.3.15 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM instrumentation is to display unit variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to monitor and take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs).

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

The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed, and so that the need for and magnitude of further actions can be determined. These essential instruments are identified in SAR Table 7-11A (Ref. 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 equate to two classes of parameters identified during unit specific implementation of Regulatory Guide 1.97 as Type A and Category I variables.

Type A variables are specified because they provide the primary information that permits the control room operator to take specific manually controlled actions that are required when no automatic control is provided and that are required for safety systems to accomplish their safety functions for DBAs.
Category I variables are the key variables deemed risk significant because they are needed to:

- Determine whether systems important to safety are performing their intended functions;

- Provide information to the operators that will enable them to determine the potential for causing a gross breach of the barriers to radioactivity release; and

- 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.

These key variables are also identified in SAR Table 7-11A (Ref. 1).

BACKGROUND (continued)

The specific instrument Functions listed in Table 3.3.15-1 are discussed in the LCO Bases Section.

APPLICABLE SAFETY ANALYSES

The PAM instrumentation ensures the availability of information so that the control room operating staff can:

Perform the diagnosis specified in the abnormal and emergency operating procedures. These variables include preplanned actions for the primary success path of DBAs (e.g., loss of coolant accident (LOCA);

- Take the specified, preplanned, manually controlled actions, for which no automatic control is provided, which are required for safety systems to accomplish their safety functions;
- Determine whether systems important to safety are performing their intended functions;
- Determine the potential for causing 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 estimate the magnitude of any impending threat.

SAR Section 7.3.4 (Ref. 4) documents the results of the Regulatory Guide 1.97 analysis process which identified Type A and Category I non-Type A variables.

In MODE 1, PAM instrumentation that meets the definition of Type A in Regulatory Guide 1.97 satisfies Criterion 3 of 10 CFR 50.36 (Ref. 5). In MODES 2 and 3, Category I, non-type A, instrumentation must be retained in Technical Specifications 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 of 10 CFR 50.36 (Ref. 5).

LCO

LCO 3.3.15 requires two OPERABLE channels for all but one Function to ensure no single failure prevents the operators from being presented with the information necessary to determine the status of the unit and to bring the unit to, and maintain it in, a safe condition following that accident. Furthermore, provision of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of

LCO (continued)

displayed information. When a channel includes more than one qualified control room indication, such as both an indicator and a recorder, or an indicator and Safety Parameter Display System (SPDS) readout, etc., only one indication is required for channel OPERABILITY.

The exception to the two channel requirement is reactor building isolation valve position. In this case, the important information is the status of the reactor building penetrations. The LCO requires one position indicator for each automatic reactor building isolation valve. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the active valve and prior knowledge of the passive valve or via system boundary status. If a normally active reactor building isolation valve is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE (See Table 3.3.15-1, Note (a)).

Each of the specified instrument Functions listed in Table 3.3.15-1 are discussed below:

1. <u>Wide Range Neutron Flux</u>

   Wide Range Neutron Flux indication is a Type B, Category I variable provided to verify reactor shutdown. The Wide Range Neutron Flux channels consist of two channels of qualified fission chamber based instrumentation (Gamma-Metrics) with readout on one recorder and on the SPDS. The channels provide indication over a range of 10^{-8} to 100% full power (Ref. 1).

2. <u>Reactor Coolant System (RCS) Hot Leg Temperature</u>

   RCS Hot Leg Temperature instrumentation is a Type A Category I variable provided for verification of core cooling and long term surveillance including determining when to secure reactor coolant pumps following a LOCA. Reactor outlet temperature inputs are provided by two fast response resistance elements and associated transmitters in each loop. The two channels provide readout on one indicator and one recorder and on the SPDS. The channels provide indication over a range of 50°F to 700°F.

3. <u>RCS Hot Leg Level</u>

   RCS Hot Leg Level instrumentation is a Type B, Category I variable provided to support operator diagnosis of inadequate core cooling and tracking reactor coolant inventory. Each channel monitors level from one (1) wide range and any two (2) of four (4) narrow range transmitters per hot leg. Channel OPERABILITY requires a minimum of one wide range and any two of the narrow range transmitters in the same channel OPERABLE. In addition, reference leg temperature inputs and core exit thermocouple average temperature are used for density compensation of the level. The system is

LCO (continued)

designed to infer the water level in the hot legs during no-flow conditions. The channels provide readout on two indicators and on the SPDS. The channels provide indication over a unit elevation range of 368 feet 6 inches to 417 feet 6 inches.

4. RCS Pressure (Wide Range)

RCS Pressure (Wide Range) instrumentation is provided for verification of core cooling and RCS integrity long term surveillance.

Wide range RCS loop pressure is measured by pressure transmitters with a span of 0 psig to 3000 psig. The pressure transmitters are located inside the RB. Redundant monitoring capability is provided by two channels of instrumentation. This control room display, consisting of one indicator and one recorder, and the SPDS is the primary indication used by the operator during an accident. Therefore, the accident monitoring specification deals specifically with this portion of the instrument string.

RCS Pressure is a Type A, Category I variable because the operator uses this indication to monitor the cooldown of the RCS following a steam generator (SG) tube rupture or small break LOCA. Operator actions to maintain a controlled cooldown, such as adjusting SG pressure or level, would use this indication. In addition, high pressure injection (HPI) flow is throttled based on RCS Pressure and subcooled margin.

5. Reactor Vessel Water Level

Reactor Vessel Water Level instrumentation is a Type B, Category I variable and is provided for verification and long term surveillance of core cooling. The reactor vessel level monitoring system provides an indication of the liquid level above the fuel.

The level range extends from the top of the vessel dome down to the top of the fuel alignment plate. The response time is short enough to track the level during small break LOCA events. The resolution is sufficient to show the initial level drop, the key locations near the hot leg elevation, and the lowest levels just above the fuel. This provides the operator with adequate indication to track the progression of the accident and to detect the consequences of its mitigating actions or the functionality of automatic equipment.

The Reactor Vessel Water Level channels consist of two redundant Radcal Level Instruments (RLIs) (each containing nine (9) axially distributed level sensors and one reactor vessel head temperature thermocouple to detect reactor coolant inventory above the core), and a data acquisition system with readout on two indicators. When Reactor Coolant Pumps are running, all except the dome sensors are interlocked to read "invalid" due to flow induced

LCO (continued)

variables that may offset the sensor outputs. Channel OPERABILITY requires a minimum of three sensors in the upper plenum region and two sensors in the dome region OPERABLE. Readout for this parameter is also provided on the SPDS.

6. Reactor Building Water Level (Wide Range)

Reactor Building Water Level (Wide Range) instrumentation is a Type B, Category I variable and is provided for verification of net positive suction head (NPSH) for the recirculation phase. The Reactor Building Water Level instrumentation consists of two channels with readout on two indicators and one recorder and on the SPDS. The channels provide water level indication over a range of 0 to 144 inches.

7. Reactor Building Pressure (Wide Range)

Reactor Building Pressure (Wide Range) instrumentation is a Type B, Category I variable and is provided for verification of RCS and reactor building OPERABILITY. Reactor Building Pressure instrumentation consists of two channels with readout on two indicators and one recorder and on the SPDS. The channels provide pressure indication over a range of 0 to 210 psia (-15 to 195 psig).

8. Automatic Reactor Building Isolation Valve Position

Automatic Reactor Building Isolation Valve Position is a Type B, Category I variable and is provided for verification of the isolation status of the reactor building penetration. The LCO requires one channel of valve position indication in the control room to be OPERABLE for each automatic isolation valve in a reactor building penetration flow path, i.e., two total channels of position indication for a penetration flow path with two automatic valves. For reactor building penetrations with only one automatic valve having control room indication, Note (b) requires a single channel of valve position indication to be OPERABLE. This is sufficient to verify the isolation status of each isolable penetration via indicated status of the automatic valve, as applicable, and prior knowledge of passive valve or system boundary status. If a penetration flow path is isolated, position indication for the isolation valve(s) in the associated penetration flow path is not needed to determine status. Therefore, the position indication for valves in an isolated penetration flow path is not required to be OPERABLE. Each penetration is treated separately and each penetration flow path is considered a separate function. Therefore, separate Condition entry is allowed for each inoperable penetration flow path.

The isolation valve position PAM instrumentation consists of Class 1E position switches for each automatic reactor building isolation valve. These switches provide "closed -not closed" indication via indicating lights in the control room.

LCO (continued)

9. Reactor Building Area Radiation (High Range)

Reactor Building Area Radiation (High Range) instrumentation is a Type E, Category I variable and is provided to monitor the potential for significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans. The Reactor Building Area Radiation instrumentation consists of two channels with readout on two indicators and one recorder and on the SPDS. The channels provide high radiation indication over a range of 1 to 10^8 R/hour gamma; however, the required range is only 1 to 10^7 R/hour gamma.

10. Reactor Building Hydrogen Concentration

Reactor Building Hydrogen Concentration instrumentation is a Type A, Category I variable and is provided to detect high hydrogen concentration conditions that represent a potential for a reactor building breach and the need to initiate hydrogen control measures such as hydrogen purge. This variable is also important in verifying the adequacy of mitigating actions. The Reactor Building Hydrogen Concentration instrumentation consists of two channels with readout on two indicators and one recorder and on the SPDS. The channels provide hydrogen concentration indication over a range of 0 to 10% volume.

11. Pressurizer Level

Pressurizer Level instrumentation is a Type D, Category I variable and is used in combination with other system parameters to determine whether to terminate safety injection (SI), if still in progress, or to reinitiate SI if it has been stopped. Knowledge of pressurizer water level is also used to verify the unit conditions necessary to establish natural circulation in the RCS and to verify that the unit is maintained in a safe shutdown condition. The Pressurizer Level instrumentation consists of two channels with readout on one indicator and one recorder and on the SPDS. The channels provide level indication over a range of 87 to 407 inches (bottom to top).

12. Steam Generator Water Level

Steam Generator Water Level instrumentation is a Type A, Category I variable provided to monitor operation of RCS heat removal via the SG and to determine the affected SG for isolation following a SGTR event . The indication of SG level is provided by the low range and high range level instrumentation, covering a span of 6 inches to 500 inches above the lower tubesheet. The measured differential pressure is displayed in inches of water.

The Steam Generator Water Level instrumentation consists of two channels (A and B) for each steam generator for the low range and two channels for each steam generator for the high range with readout on four dual indicators (one

LCO (continued)

SG channel with both ranges per indicator) and on the SPDS. The Low Range channels provide level indication over a range of 6 to 156 inches of water and the High Range channels provide level indication over a range of 102 to 500 inches of water. Each range of water level instrumentation for each steam generator is considered a separate Function of PAM Instrumentation. Two additional channels (C and D) also monitor SG water level for EFIC but these channels are not required as PAM instrumentation.

SG high range level indication is used by the operator to manually raise and control SG level to establish reflux boiling (boiler condenser) heat transfer. Operator action is initiated on a loss of subcooled margin. Feedwater flow is increased until the indicated level reaches the reflux boiling (boiler condenser) setpoint.

13. <u>Steam Generator Pressure</u>

Steam Generator Pressure instrumentation is a Type A, Category I variable provided to support operator diagnosis of a design basis steam generator tube rupture to identify and isolate the affected SG. In addition, SG pressure is a key parameter used by the operator to evaluate primary-to-secondary heat transfer. For example, the operator may use this indication to control the primary system cooldown following a steam line break accident or a small break loss of coolant accident (LOCA).

Steam generator pressure measurement is provided by two pressure transmitters per SG. The channels provide readout on two indicators (one per SG) and two dual pen recorders (one per SG) and on the SPDS. The channels provide pressure indication over a range from 0 to 1200 psig. The pressure instrumentation for each steam generator is considered a separate Function of PAM Instrumentation.

14. <u>Condensate Storage Tank (QCST) Level</u>

QCST Level instrumentation is a Type A, Category I variable and is provided to ensure a readily available, condensate quality water supply for EFW. Inventory is monitored by a 0 to 30 feet level indication. QCST Level is displayed on one control room indicator and one recorder, and on the SPDS.

QCST Level is the primary indication used by the operator to identify loss of QCST volume and replenish the QCST or align suction of the EFW pumps to the safety related source, i.e., service water.

LCO (continued)

15. Borated Water Storage Tank Level

Borated Water Storage Tank (BWST) Level instrumentation is a Type A, Category I variable provided to support action for long term cooling requirements, i.e., to determine when to initiate the switch-over of the core cooling pump suction from the BWST to sump recirculation. BWST Level measurement is provided by two channels with readout on two indicators and one recorder and on the SPDS. The level transmitters are calibrated over a range of 0 to 45 feet. The "0" reference level is the level instrument tap, which is approximately 5 inches above the bottom of the tank.

16. Core Exit Temperature

Core Exit Temperature is a Type C, Category I variable and is provided for verification and long term surveillance of core cooling. Twenty-four (24) qualified core exit thermocouples (CETs) are provided with six (6) located in each core quadrant. Two CETs are required in each core quadrant and readout is provided on two indicators and on the SPDS. The channels provide core exit temperature indication over a range of 50 to 2300°F. This Function is specified on a "CETs per quadrant" basis. Therefore, each quadrant of required CETs is considered a separate Function for Condition entry.

17. Emergency Feedwater Flow

EFW Flow instrumentation is a Type D, Category I variable and is provided to monitor operation of RCS heat removal via the SGs. One channel is provided for each flow path of an EFW pump to each SG, i.e., each pump feeds both SGs so there are four flow paths. The channels provide indication of EFW Flow to each SG over a range of 0 to 900 gpm. Each transmitter provides an input to a control room indicator (four indicators total) and to the SPDS. Flow measurement to each steam generator is considered a separate Function of PAM Instrumentation.

EFW Flow is the primary indication used by the operator to verify that the EFW System is delivering flow to the correct SG.

18. High Pressure Injection Flow

See the discussion for Function 19 below.

LCO (continued)

19. Low Pressure Injection Flow

High and Low Pressure Injection Flow instrumentation is a Type A, Category I variable provided to support action for long term cooling requirements.

HPI flow may be throttled based on RCS pressure, subcooled margin and pressurizer level, and to balance flow rates between the injection lines. LPI flow information is used to determine when it is acceptable to terminate HPI. High and Low Pressure Injection Flow measurement is provided by two channels each with readout on two indicating recorders for high pressure injection (HPI), and with readout on two indicators and one recorder for low pressure injection (LPI) and on the SPDS. Each HPI channel includes four instruments (one per flow path) which provide flow indication over a range from 0 to 200 gpm, and the LPI channels provide flow indication over a range from 0 to 4500 gpm.

20. Reactor Building Spray Flow

Reactor Building Spray Flow instrumentation is a Type A, Category I variable provided to support action for long term reactor building cooling requirements (e.g., maintain NPSH) and iodine removal. Reactor Building Spray Flow measurement is provided by two channels with readout on two indicators and one recorder and on the SPDS. The channels provide flow indication over a range from 0 to 2000 gpm.

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables are related to the diagnosis and preplanned actions required for safe shutdown and to determine that safety systems are performing their intended function when required. In MODES 4, 5, and 6, unit conditions are such that the likelihood of an event occurring that would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 is added to the ACTIONS to exclude the MODE change restriction of LCO 3.0.4. This exception allows entry into an applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require a unit shutdown. This exception is acceptable due to the passive function of the instruments, the operator's ability to respond to an accident utilizing alternate instruments and methods, and the low probability of an event requiring these instruments.

ACTIONS (continued)

Note 2 is 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 3.3.15-1. The Completion Time(s) of the inoperable channels of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function. This Note is also applicable for Table 3.3.15-1 items 12a, 12b, 12c, 12d, 13a, 13b, 17a and 17b, each of which is considered a separate Function.

<u>A.1</u>

When one or more Functions have one required channel inoperable, the inoperable channel must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience. This 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.

<u>B.1</u>

Required Action B.1 specifies initiation of actions to prepare and submit a Special Report to the NRC. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative actions. The Special Report is to be submitted in accordance with 10 CFR 50.4 within 30 days of entering Condition B. 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 unit conditions that would require information provided by this instrumentation. The Completion Time of "Immediately" for Required Action B.1 identifies the start of the "clock" for submittal of the Special Report. Condition B is modified by a Note requiring Required Action B.1 to be completed whenever the Condition is entered. The Note ensures the requirement to prepare and submit the report is completed. Restoration alone per Required Action A.1 after the initial Completion Time of 30 days does not alleviate the need to report the extended inoperability to the NRC.

<u>C.1</u>

When one or more Functions have two required channels inoperable (i.e., two channels inoperable in the same Function), one channel in the Function should be restored to OPERABLE status within 7 days. This Condition does not apply to the hydrogen monitor channels. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrumentation action operation and the availability of alternative 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 of qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the probability that the PAM Function will be unavailable should an accident occur.

ACTIONS (continued)

D.1

When two required hydrogen monitor channels are inoperable, Required Action D.1 requires one channel to be restored to OPERABLE status. This action restores the monitoring capability of the hydrogen monitor. The 72 hour Completion Time is based on the relatively low probability of an event requiring hydrogen monitoring. Continuous operation with two required channels inoperable is not acceptable because alternate indications are not available.

E.1

Required Action E.1 directs entry into the appropriate Condition referenced in Table 3.3.15-1. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met the Required Action and associated Completion Time of Condition C or D, as applicable, Condition E is entered for that channel and provides for transfer to the appropriate subsequent Condition.

F.1

If the Required Action and associated Completion Time of Conditions C or D are not met and Table 3.3.15-1 directs entry into Condition F, the unit must be brought to a MODE in which the requirements of this LCO do not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

G.1

If the Required Action and associated Completion Time of Conditions C or D are not met and Table 3.3.15-1 directs entry into Condition E, alternate means of monitoring the parameter should be applied and the Required Action is not to shut down the unit but rather to initiate actions to prepare and submit a Special Report to the NRC. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels. The Special Report is to be submitted in accordance with 10 CFR 50.4 within 30 days of entering Condition F.

Both the RCS Hot Leg Level and the Reactor Vessel Level are methods of monitoring for inadequate core cooling.

ACTIONS (continued)

G.1 (continued)

The alternate means of monitoring the Reactor Building Area Radiation (High Range) consist of a combination of installed area radiation monitors and portable instrumentation.

The Completion Time of "Immediately" for Required Action G.1 identifies the start of the "clock" for submittal of the Special Report. Condition G is modified by a Note requiring Required Action G.1 to be completed whenever the Condition is entered. The Note ensures the requirement to prepare and submit the report is completed. Restoration alone per Required Action C.1 or Required Action D.1 after the initial Completion Time of 7 days, or 72 hours, respectively, does not alleviate the need to report the extended inoperability to the NRC.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs apply to each PAM instrumentation Function in Table 3.3.15-1.

SR 3.3.15.1

Performance of the CHANNEL CHECK once every 31 days for each required instrumentation channel that is normally energized provides reasonable assurance for prompt identification of a gross failure of instrumentation. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel with a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared with similar unit instruments located throughout the unit. For the reactor building hi-range radiation monitor, the CHANNEL CHECK should also note the detector's response to the keep alive source.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Offscale low current loop channels are, where practical, verified to be reading at the bottom of the range and not failed downscale.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.15.1 (continued)

The Frequency is based on unit 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 this LCO's required channels.

SR 3.3.15.2

A CHANNEL CALIBRATION is performed every 18 months. CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. This test verifies the channel responds to measured parameters within the necessary range and accuracy.

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult, and there is no adjustment that can be made to the detectors. Furthermore, adjustment of the detectors is unnecessary because they are passive devices, with minimal drift. Finally, the detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output.

For the Reactor Building Area Radiation instrumentation, a CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr, and a one point calibration check of the detector below 10 R/hr with a gamma source.

For the Reactor Building Hydrogen Concentration instrumentation, the calibration includes proper consideration of moisture effect.

Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature detector (RTD) sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the Core Exit thermocouple sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

The Frequency is based on operating experience and consistency with the typical industry refueling cycle and is justified by the assumption of at least an 18 month calibration interval in the determination of the magnitude of equipment drift.

REFERENCES

1. SAR, Table 7-11A.
2. Regulatory Guide 1.97.
3. NUREG-0737, 1979.
4. SAR, Section 7.3.4.
5. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.16 Control Room Isolation - High Radiation

BASES

BACKGROUND

The principal function of the Control Room Isolation - High Radiation is to provide an enclosed environment from which the unit can be operated following an uncontrolled release of radioactivity. The high radiation isolation function provides assurance that under the required conditions, an isolation signal will be initiated to provide isolation and shutdown the unit's normal control room ventilation supply fan.

The control room isolation signal is provided by two independent radiation monitoring systems; one associated with each unit. The Unit 1 radiation monitor is in the Unit 1 control room normal supply duct. The Unit 2 radiation monitor is in the Unit 2 control room normal supply duct. If a radioactivity concentration significantly above normal background level is detected, the unit monitor will initiate a shutdown of the unit's normal duty supply fans, place both unit's ventilation dampers in their recirculation mode, and start the unit's Control Room Emergency Ventilation System (CREVS) supply fan.

The trip setpoints are chosen sufficiently below hazardous radiation levels to minimize operator exposure during an accident and sufficiently above normally experienced background levels to minimize spurious actuation. The habitability systems functional design bases are provided in the ANO Unit 2 SAR, Section 6.4 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The control room must be maintained habitable during post accident operations and recovery. The CREVS is a shared system which provides a filtered makeup air source for the common control room habitability envelope from which the unit can be operated following an uncontrolled release of radioactivity. Upon receipt of a high radiation signal, the associated normal ventilation supply fans are shutdown, the control room isolation dampers are closed to isolate both normal outside air intakes, and the associated CREVS train emergency filtration function is initiated. Operator action is necessary to shut down one train of CREVS (if both actuate) in order to prevent operator doses greater than identified by the habitability analysis. Operator action is also necessary to verify that at least one door between the Unit 1 and Unit 2 control rooms is open to provide appropriate pressurization and recirculation.

APPLICABLE SAFETY ANALYSES (continued)

In MODES 1, 2, 3, and 4, the radiation monitor isolation of the control room habitability envelope and actuation of the CREVS provides a habitable environment for the operators following a design basis accident or any event with a significant release of radioactivity.

During movement of irradiated fuel assemblies, the radiation monitor isolation of the control room habitability envelope and actuation of the CREVS provides a habitable environment for the operators following a fuel handling accident.

The Control Room Isolation-High Radiation satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

The LCO requires that instrumentation necessary to initiate the CREVS is OPERABLE. Two channels of Control Room Isolation-High Radiation are required to be OPERABLE to provide actuation capability from high radiation either entering the control room habitability envelope via the Unit 1 normal supply duct (2RITS-8001) or entering the control room habitability envelope via the Unit 2 normal supply duct (2RITS-8750-1).

Trip setpoints are specified in the unit specific procedures. The setpoints are selected to ensure the as-found setpoint measured by the CHANNEL FUNCTIONAL TEST does not exceed the Allowable Value if the bistable is performing as required. The trip setpoint for this parameter does not include additional allowances for instrument uncertainties. Therefore, the trip setpoint and Allowable Value are the same.

APPLICABILITY

The control room isolation capability on high radiation shall be OPERABLE in MODES 1, 2, 3, 4, and during movement of irradiated fuel assemblies in any MODE. If a radioactive release were to occur during any of these conditions, the control room would have to remain habitable to ensure continued reactor control capability from the control room.

ACTIONS

A.1

Condition A applies to inoperability of one channel of the Control Room Isolation - High Radiation function in MODE 1, 2, 3, or 4.

With one channel of Control Room Isolation-High Radiation function inoperable, one channel remains OPERABLE to provide an automatic actuation function. Since the probability of an event which would be detected by only one of the radiation monitors is low, operation of the unit may continue for up to 7 days. If the CREVS actuation instrumentation is not returned to OPERABLE status, the unit ventilation system must be placed, within the 7 days, in a state equivalent to that which occurs after the high radiation actuation has occurred with one OPERABLE train of the CREVS in the emergency recirculation mode of operation. Reactor operation may then continue indefinitely in this state. The 7 day Completion Time is sufficient to restore most causes of inoperable actuation instrumentation.

B.1

Condition B applies to inoperability of both channels of the Control Room Isolation-High Radiation function in MODE 1, 2, 3, or 4.

With both channels of Control Room Isolation - High Radiation inoperable, the ventilation system must be placed in a condition that does not require the isolation to occur, i.e., in a state equivalent to that which occurs after the high radiation isolation has occurred with one OPERABLE train of the CREVS in operation. Reactor operation can continue indefinitely in this state. The 1 hour Completion Time is a sufficient amount of time in which to take the Required Action.

C.1 and C.2

If the CREVS cannot be placed into the emergency recirculation mode while in MODE 1, 2, 3, or 4, actions must be taken to minimize the chances of an accident that could lead to radiation releases. The unit must be placed in at least MODE 3 within 6 hours, with a subsequent cooldown to MODE 5 within 36 hours. This places the reactor in a low energy state that allows greater time for operator action if habitation of the control room is precluded. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

ACTIONS (continued)

D.1 and D.2

Required Action D.1 is the same as discussed earlier for Condition A, except for Completion Time. If the CREVS cannot be placed into recirculation mode while moving irradiated fuel assemblies, then Required Action D.2 suspends actions that could lead to an accident that could release radioactivity resulting from a fuel handling accident.

Required Action D.2 places the irradiated fuel in a safe and stable configuration in which it is less likely to experience an accident that could result in a release of radioactivity. The irradiated fuel must be maintained in these conditions until the automatic isolation capability is returned to operation or when manual action places one train of the CREVS into the emergency recirculation mode. The Completion Time of "Immediately" is consistent with the urgency of the situation and accounts for the high radiation function, which provides the only automatic Control Room Isolation function capable of responding to radiation release due to a fuel handling accident. The Completion Time does not preclude placing any fuel assembly into a safe position before ceasing any such movement.

Note that in certain circumstances, such as fuel handling in the fuel handling area during power operation, both Condition A or B and Condition D may apply in the event of channel failure(s).

SURVEILLANCE REQUIREMENTS

SR 3.3.16.1

Performance of a CHANNEL CHECK for the Control Room Isolation - High Radiation actuation instrumentation once every 12 hours provides reasonable assurance for prompt identification of a gross failure of instrumentation. Performance of the CHANNEL CHECK helps ensure that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Acceptance criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. The Frequency is based on operating experience that demonstrates channel failure is rare.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.16.2

A Note allows a channel to be inoperable for up to 3 hours for surveillance testing without entering the associated Conditions and Required Actions, although during this time period it cannot actuate a control room isolation. This is based on the average time required to perform channel surveillance. It is not acceptable to remove channels from service for more than 3 hours to perform required surveillance testing without declaring the channel inoperable.

SR 3.3.16.2 is the performance of a CHANNEL FUNCTIONAL TEST once every 31 days to ensure that the channels can perform their intended functions. This test verifies the capability of the instrumentation to provide the automatic Control Room Isolation. Any setpoint adjustment shall be consistent with the setpoint requirements.

The 31 day Frequency is based on operating experience which indicates that the instrumentation usually passes the CHANNEL FUNCTIONAL TEST when performed on a monthly basis.

SR 3.3.16.3

This SR requires the performance of a CHANNEL CALIBRATION to ensure that the instrument channel remains operational with the correct setpoint.

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATION must be performed consistent with the setpoint requirements.

The Frequency is based on the assumption of at least an 18 month calibration interval in the determination of the magnitude of equipment drift and is consistent with the typical refueling cycle.

REFERENCES

1. ANO-2 SAR, Section 6.4.

2. 10 CFR 50.36.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and abnormalities assume initial conditions within the normal steady state envelope. The limits placed on DNB related parameters ensure that these parameters will not be less conservative than were assumed in the analyses and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed.

The LCO for minimum RCS pressure is consistent with that used as the initial pressure in the analyses. Considering only pressure, a pressure greater than the minimum specified will produce a higher DNBR; and a pressure lower than the minimum specified will produce a lower DNBR.

The LCO for maximum RCS coolant hot leg temperature is consistent with the initial hot leg temperature in the analyses. Considering only temperature, a hot leg temperature lower than that specified will produce a higher DNBR; and a temperature higher than that specified will produce a lower DNBR.

The RCS flow rate is not expected to vary during operation with all pumps running. The LCO for the minimum RCS flow rate corresponds to that assumed for the DNBR analyses. Considering only flow rate, a higher RCS flow rate than that specified will produce a higher DNBR; and a lower RCS flow rate will produce a lower DNBR.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Refs. 1 and 2). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR criteria of ≥ 1.30 or ≥ 1.18, for the BAW-2 or the BWC critical heat flux correlation, respectively. For the locked rotor accident, the minimum DNB ratio is not less than applicable critical heat flux correlation limit, or fuel cladding is shown to experience no significant temperature excursions. These are the acceptance criteria for the RCS DNBR parameters. Changes to the facility that could impact these parameters must be assessed for their impact on the DNBR criterion. The transients analyzed include loss of coolant flow events and dropped or stuck control

APPLICABLE SAFETY ANALYSES (continued)

rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," LCO 3.2.4, "QUADRANT POWER TILT (QPT)," LCO 3.1.4, "CONTROL ROD Group Alignment Limits," and LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits."

The safety analyses to establish reload operating limits are performed using nominal values for RCS coolant average temperature, core outlet pressure, and RCS flow rate and core power level with appropriate application of associated uncertainty. Consistent with Statistical Core Design (SCD) methodology, applicable random parametric uncertainties are combined statistically. As necessary, bias parameters are included deterministically. The RCS temperature and pressure are measured in the hot leg. The surveillance criteria specified in the COLR include adjustment for measurement location. The COLR specified hot leg temperature is the maximum allowed so that the analysis value is not exceeded. The COLR specified hot leg pressure and flow are the minimum allowed so that the analysis values are not exceeded.

Analyses have been performed to establish the pressure, temperature, and flow rate requirements for two pump, three pump and four pump operation. The flow limits for two pump and three pump operation are substantially lower than for four pump operation. To meet the DNBR criterion, a corresponding maximum power limit is required (see Bases for LCO 3.4.4, "RCS Loops-MODES 1 and 2").

The steady state limits on DNBR related parameters are provided in Safety Limit (SL) 2.1.1, "Reactor Core SLs." Those limits are less restrictive on plant operations than the limits of LCO 3.4.1, but violation of an SL merits a stricter, more severe Required Action. Should a violation of LCO 3.4.1 occur, a check must be performed to determine whether an SL may have been exceeded.

The RCS DNB limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

This LCO specifies limits on the monitored process variables: RCS loop (hot leg) pressure, RCS hot leg temperature, and RCS total flow rate to ensure that the core operates within the limits assumed for the plant safety analyses. Operating within these limits will result in meeting DNBR criteria in the event of a DNB limited transient.

The pressure and temperature limits are to be applied to the loop with two reactor coolant pumps (RCPs) running for the three RCPs operating condition.

The surveillance criteria for pressure, temperature, and flow rate as specified in the COLR have been appropriately adjusted for the measurement location and for instrument error consistent with supporting analysis.

APPLICABILITY

In MODE 1, the limits on RCS pressure, RCS hot leg temperature, and RCS flow rate must be maintained during steady state operation in order to ensure that DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES the power level is low enough so that DNB is not a significant concern.

The Note indicates the limit on RCS pressure may be exceeded during short term operational pressure transients resulting from a THERMAL POWER change > 5% RTP per minute. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, for transients initiated from power levels less than the Allowable Thermal Power, increased DNBR margin exists to offset the temporary pressure variations.

ACTIONS

A.1

Loop pressure and hot leg coolant temperature are controllable and measurable parameters. With one or both of these parameters not within the LCO limits, action must be taken to restore the parameters. RCS flow rate is not a controllable parameter and is not expected to vary during steady state operation. However, if the flow rate is below the LCO limit, the parameter must be restored to within limits or power must be reduced as required in Required Action B.1, to eliminate the potential for violation of the minimum DNBR limit.

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, determine the cause for the off normal condition, and restore the readings within limits. The Completion Time is based on plant operating experience.

B.1

If the Required Action and associated Completion Time 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 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis assumptions.

The 6 hour Completion Time is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

The RCS pressure value specified is dependent on the number of pumps in operation and has been adjusted to account for the pressure difference between the core exit and the measurement location. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation is within safety analysis assumptions.

A Note has been added to indicate the pressure limits are to be applied to the loop with two pumps in operation for the three pump operating condition.

SR 3.4.1.2

The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify that operation is within safety analysis assumptions.

A Note has been added to indicate the temperature limits are to be applied to the loop with two pumps in operation for the three pump operating condition.

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the available flow indications. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify that operation is within safety analysis assumptions.

SR 3.4.1.4

Measurement of RCS total flow rate once every 18 months allows the installed RCS flow instrumentation to be calibrated and verifies that the actual RCS flow is greater than or equal to the minimum required RCS flow rate specified in the COLR.

The Frequency of 18 months reflects the importance of verifying flow after a refueling outage when the core has been altered or RCS flow characteristics may have been modified, which may have caused change of flow.

The Surveillance is modified by a Note that indicates the SR does not need to be performed until stable thermal conditions are established at higher power levels (i.e., ≥ 90% RTP). The Note provides for measurement of the flow rate at normal operating conditions at power in MODE 1. The Surveillance may be performed at low power or in MODE 2 or below. However, at low or zero power conditions, the indications are less accurate and significant penalties for uncertainties may be necessary. Performance of the calorimetric heat balance at a high power level and normal operating conditions provides for the most accurate flow verification.

REFERENCES

1. SAR, Chapter 14.

2. SAR, Section 3A.6.

3. BAW-10179P-A, 2/96.

4. 10 CFR 50.36.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

Establishing the value for the minimum temperature for reactor criticality is based upon considerations for:

a. Operation within the existing instrumentation ranges;

b. Operation with reactor vessel above its minimum nil ductility reference temperature when the reactor is critical.

The reactor coolant moderator temperature coefficient used in core operating and accident analysis is typically defined for the normal (average) operating temperature range (532°F to 579°F). The Reactor Protection System (RPS) receives inputs from the narrow range hot leg temperature detectors, which have a range of 520°F to 620°F. The integrated control system controls average temperature (Tavg) using inputs of the same range. Nominal Tavg for making the reactor critical is 532°F. Safety and operating analyses for lower temperatures have not been performed for all possible scenarios.

APPLICABLE SAFETY ANALYSES

There are no accident analyses that dictate the minimum temperature for criticality, but all low power safety analyses assume initial temperatures near the 525°F limit (Ref. 1).

The RCS minimum temperature for criticality satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2).

LCO

Compliance with the LCO ensures that the reactor will not be made or maintained critical at a temperature significantly less than the hot zero power (HZP) temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

LCO (continued)

The LCO limit of 525°F has been selected to be within the instrument indicating range (520°F to 620°F). This parameter value is considered to be a nominal value. No additional allowances for instrument uncertainty are required in the implementing procedures.

APPLICABILITY

The reactor has been designed and analyzed to be critical in MODES 1 and 2 only with Tavg ≥ 525°F. Criticality is not permitted in any other MODE. Therefore, this LCO is applicable in MODE 1 and MODE 2.

ACTIONS

A.1

With Tavg below 525°F, 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 MODE 3 in 30 minutes. Rapid reactor shutdown can be readily and practically achieved in a 30 minute period. The Completion Time reflects the ability to perform this Action and maintain the plant within the analyzed range. If Tavg can be restored within the 30 minute time period, shutdown is not required.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

RCS average temperature is required to be verified at or above 525°F every 12 hours. The SR to verify RCS average temperature every 12 hours takes into account indications that are continuously available to the operator in the control room and is consistent with other routine surveillances which are typically performed once per shift. In addition, Operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.

RCS T_{avg} is normally calculated as the average of the unit T_{hot} (hot temperature average of loops A and B) and the unit T_{cold} (cold temperature average of loops A and B). During operation with 3 RCPs in operation, T_{avg} is calculated as the average of the loop T_{hot} and loop T_{cold} in the loop that has 2 RCPs running.

REFERENCES

1. SAR, Chapter 14.

2. 10 CFR 50.36.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, and unit transients. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

Figures 3.4.3-1, 3.4.3-2, and 3.4.3-3 contain P/T limit curves for heatup, cooldown, inservice hydrostatic testing, and physics testing at RCS temperatures ≤ 525°F, and the maximum rate of change of reactor coolant temperature. The methods and criteria employed to establish operating pressure and temperature limits are described in BAW-10046A (Ref. 1). These limit curves are applicable through thirty-one effective full power years (EFPY) of operation. The pressure limit is adjusted for the pressure differential between the point of system pressure measurement and the limiting component for the various operating reactor coolant pump combinations.

Each P/T curve defines an acceptable region for normal operation below and to the right of the limit curve. The curves are used to develop operational guidance for use during heatup or cooldown maneuvering.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel. The vessel is the component most subject to brittle failure due to the fast neutron embrittlement it experiences during power operation, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for material fracture toughness requirements of the reactor coolant pressure boundary (RCPB) materials. Reference 2 requires an adequate margin to brittle failure during normal operation, abnormalities, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Appendix G (Ref. 3).

Linear elastic fracture mechanics (LEFM) methodology is used to determine the stresses and material toughness at locations within the RCPB. The LEFM methodology follows the guidance given by 10 CFR 50, Appendix G; ASME Code, Section III, Appendix G; and Regulatory Guide 1.99 (Ref. 4).

BACKGROUND (continued)

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10CFR50. Results of this analysis, including the actual pressure-temperature limitations of the reactor coolant pressure boundary, are given in FTI Document 77-1258569-01 (Ref. 5). The service period was reduced by one effective full power year from that assumed in Reference 5 to be conservative with respect to independent calculations performed by the NRC staff. The limiting weld material is being irradiated as part of the B&W Owners Group Integrated Reactor Vessel Material Surveillance Program and the identification and locations of the capsules containing the limiting weld material is discussed in the latest revision to B&W report, BAW-1543 (Rev. 6). The chemical composition of the limiting weld material is reported in the B&W report, BAW-2121P (Rev. 7). The effect of neutron irradiation on the nil ductility reference temperature (RT_{NDT}) of the limiting weld material is reported in FTI Calculations 32-1245917-00 and 32-1257716-00 (Rev. 8).

The actual shift in the RT_{NDT} of the vessel beltline region material will be established periodically by removing and evaluating the irradiated reactor vessel material surveillance specimens, in accordance with Appendix H of 10 CFR 50 (Ref. 9). These specimens are installed near the inside wall of this or a similar reactor vessel in the core region. The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 3.

Prior to reaching thirty-one effective full power years of operation, Figures 3.4.3-1, 3.4.3-2, and 3.4.3-3 must be updated for the next service period in accordance with 10 CFR 50, Appendix G. The service period must be of sufficient duration to permit the scheduled evaluation of a portion of the surveillance data scheduled in accordance with the latest revision of Topical Report BAW-1543 (Ref. 6). The highest predicted adjusted reference temperature of all the beltline region materials is used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction is submitted for NRC staff review at least 90 days prior to the end of the service period.

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The calculation to generate the inservice hydrostatic testing curve uses different safety factors (per Ref. 3) than the heatup and cooldown curves. The testing curve also extends to the RCS design pressure of 2500 psia.

BACKGROUND (continued)

The P/T limit curves and associated temperature rate of change limits are developed in conjunction with stress analyses for large numbers of operating cycles and provide conservative margins to nonductile failure. Although created to provide limits for these specific normal operations, the curves also can be used to determine if an evaluation is necessary for an abnormal transient.

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 10) provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 11).

LCO

The three elements of this LCO are:

a. The limit curves for heatup, cooldown, normal operation, PHYSICS TESTING and inservice hydrostatic testing;

b. Limits on the rate of change of temperature; and

c. Limits on RCP combinations.

The LCO limits apply to all components of the RCS, except the pressurizer (as indicated by the Note). These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

LCO (continued)

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

The limit curves include the limiting pressure differential between the point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. However, the limit curves are not adjusted for possible instrument error and should not be used for operation.

The heatup and cooldown rates stated are intended as the maximum changes in temperature in one direction in the stated time periods. The actual temperature linear ramp rate may exceed the stated limits for a shorter time period provided that the maximum total temperature difference does not exceed the limit and that a temperature hold is observed to prevent the total temperature difference from exceeding the limit for the stated time period.

The acceptable P/T combinations are below and to the right of the limit curves which are applicable for the first 31 EFPY. The limit curves include the limiting pressure differential between the point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. However, the limit curves are not adjusted for possible instrument error and should not be used for operation.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follows:

a. The magnitude of the departure from the allowable operating P/T regime or the magnitude of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits Specification provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or inservice hydrostatic testing, their applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

ACTIONS

A.1 and A.2

With RCS pressure and temperature not within criticality limit of Figure 3.4.3-1 during PHYSICS TESTS with RCS temperature ≤ 525°F, 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 MODE 3 in 30 minutes. Rapid reactor shutdown can be readily and practically achieved in a 30 minute period. The Completion Time reflects the ability to perform this Action and maintain the plant within the analyzed range. If RCS pressure and temperature can be restored within the 30 minute time period, shutdown is not required.

B.1 and B.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation beyond the 72 hour Completion Time of Required Action B.2. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components. The evaluation must be completed, documented, and approved in accordance with established unit procedures and administrative controls.

ASME Code, Section XI, Appendix E (Ref. 10) may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline. The evaluation must extend to all components of the RCPB.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate beyond the 72 hour Completion Time.

Condition B is modified by a Note requiring Required Action B.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action B.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

ACTIONS (continued)

C.1 and C.2

If a Required Action and associated Completion Time of Condition B are not met, the unit must be brought to a lower MODE because: (a) the RCS remained in an unacceptable pressure and temperature region for an extended period of increased stress, or (b) a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event. Performing this examination in the required MODES reduces the RCS at reduced pressure and temperature, which decreases the possibility of propagation of undetected flaws.

If the required restoration activity cannot be accomplished within 30 minutes, Required Action C.1 and Required Action C.2 must be initiated to reduce pressure and temperature.

If the required evaluation for continued operation cannot be accomplished within 72 hours, or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Actions C.1 and C.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions. However, if the favorable evaluation is accomplished while reducing pressure and temperature conditions, a return to power operation may be initiated without completing these Required Actions.

Pressure and temperature are reduced by bringing the unit to 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 MODE from full power conditions in an orderly manner and without challenging unit systems.

D.1 and D.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified acceptable by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished promptly in a controlled manner.

In addition to restoring operation to within limits, an evaluation is required to verify that the RCPB integrity remains acceptable. The evaluation must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analysis, or inspection of the components.

ACTIONS (continued)

D.1 and D.2 (continued)

ASME Code, Section XI, Appendix E (Ref. 10), may also be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

Condition D is modified by a Note requiring Required Action D.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone, per Required Action D.1, is insufficient because higher than analyzed stresses may have occurred and may have affected RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1, SR 3.4.3.2, SR 3.4.3.3, and SR 3.4.3.4

Verification that operation is within the limits of the appropriate figure is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes.

This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or inservice hydrostatic testing may be discontinued when the definition given in the relevant unit procedure for ending the activity is satisfied.

The acceptable P/T combinations are below and to the right of the limit curves which are applicable for the first 31 EFPYs. The limit curves include the limiting pressure differential between the point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. However, the limit curves are not adjusted for possible instrument error and should not be used for operation (as identified in Note 1 on each applicable Figure).

SR 3.4.3.1 is modified by a Note that requires this SR to be performed only during system heatup operations with fuel in the reactor vessel. This SR refers to Figure 3.4.3-1 which provides applicable heatup limitations, including reactor coolant pump (RCP) operating restrictions and allowable heatup rates.
Figure 3.4.3-1 Note 2 identifies that when the decay heat removal system is operating with no RCPs operating, the indicated DHR system return temperature to the reactor vessel is the appropriate temperature indicator.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.3.1, SR 3.4.3.2, SR 3.4.3.3, and SR 3.4.3.4 (continued)

SR 3.4.3.2 is modified by a Note that requires this SR to be performed only during system cooldown operations with fuel in the reactor vessel. This SR refers to Figure 3.4.3-2 which provides applicable cooldown limitations, including reactor coolant pump (RCP) operating restrictions and allowable cooldown rates. During system cooldown operations with fuel in the reactor vessel, the RCPs are eventually removed from service. Figure 3.4.3-2 Note 2 identifies that when the decay heat removal system is operating with no RCPs operating, the indicated decay heat removal system return temperature to the reactor vessel is the appropriate temperature indicator. Figure 3.4.3-2 Note 2 also indicates that a maximum step temperature change of 25°F is allowable when removing all RCPs from operation with the decay heat removal system operating. The step temperature change is defined as the reactor coolant temperature (prior to stopping all RCPs) minus the decay heat removal system return temperature to the reactor vessel (after stopping all RCPs). The step change of 25°F is applicable only during transition from RCP operation to DHR. This step change must be included when determining the cooldown rate.

SR 3.4.3.3 is modified by a Note that requires this SR to be performed only during system heatup and cooldown operations with no fuel in the reactor vessel. This SR refers to Figure 3.4.3-2 which provides applicable heatup and cooldown limitations, including reactor coolant pump (RCP) operating restrictions and allowable heatup and cooldown rates. These curves are used during inservice hydrostatic testing that is performed in a defueled condition. The Notes on Figure 3.4.3-1 and Figure 3.4.3-2 are applicable to heatups and cooldowns performed within these limits.

SR 3.4.3.4 is modified by a Note that requires this SR to be performed only during PHYSICS TESTS with the average RCS temperature ≤ 525°F. This SR refers to Figure 3.4.3-1 which provides applicable limitations under which the unit may be critical, including Reactor Coolant Pump (RCP) operating restrictions and allowable heatup rates. This curve is used during PHYSICS TESTING. This is because LCO 3.4.2, "RCS Minimum Temperature for Criticality," normally limits the temperature for criticality to well above this curve. However, an exception to LCO 3.4.2 is provided by LCO 3.1.9, "PHYSICS TEST Exceptions - MODE 2," during PHYSICS TESTS initiated in MODE 2.

When the decay heat removal (DHR) system is operating with no RCPs operating, the indicated DHR system return temperature to the reactor vessel is the appropriate temperature indicator.

REFERENCES

1. BAW-10046A, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10CFR50, Appendix G", Rev. 2, June 1986.
2. 10 CFR 50, Appendix G.
3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
4. Regulatory Guide 1.99, Revision 2, May 1988.
5. FTI Document 77-1258569-01.
6. BAW-1543, Integrated Reactor Vessel Material Surveillance Program (latest revision).
7. BAW-2121P, Irradiation Induced Reduction in Charpy Upper Shelf Energy of Reactor Vessel Welds.
8. FTI Calculations 32-1245917-00 and 32-1257716-00.
9. 10 CFR 50, Appendix H.
10. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
11. 10 CFR 50.36.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops - MODES 1 and 2

BASES

BACKGROUND

The primary function of the reactor coolant is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the reactor coolant include:

a. Moderating the neutron energy level to the thermal state, to increase the probability of fission;

b. Improving the neutron economy by acting as a reflector;

c. Carrying the soluble neutron poison, boric acid;

d. Providing a second barrier against fission product release to the environment; and

e. Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

The RCS configuration for heat transport uses two RCS loops. Each RCS loop contains an SG and two reactor coolant pumps (RCPs). An RCP is located in each of the two SG cold legs. The pump flow rate has been sized to provide core heat removal with appropriate margin to departure from nucleate boiling (DNB) during power operation and for anticipated transients originating from power operation. This Specification requires two RCS loops with either three or four pumps to be in operation. With only two or three pumps in operation the reactor power level is restricted to a nominal 49% RTP or 75% RTP, respectively, to preserve the core power to flow relationship, thus maintaining the margin to DNB. The intent of the Specification is to require core heat removal with forced flow during power operation. Specifying the minimum number of pumps is an effective technique for designating the proper forced flow rate for heat transport, and specifying two loops provides for the needed amount of heat removal capability for the allowed power levels. Specifying two RCS loops also provides the minimum necessary paths (two SGs) for heat removal.

The Reactor Protection System (RPS) Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip setpoint is automatically reduced when a pump is taken out of service. Manual resetting is not necessary.

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 1) contain various assumptions for the Design Bases Accident (DBA) initial conditions including: RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of pumps in service.

Both transient and steady state analyses have been performed to establish the effect of RCS flow on DNB. The initial condition DNB protection for the limiting loss of coolant flow event for four, three, and two pump operation is provided by the RCS flow surveillance criteria specified in the COLR for SR 3.4.1.3 and SR 3.4.1.4. The loss of coolant flow event which has been found to produce the limiting DNB is the four-to-two pump coastdown. In addition to the coastdown events, the single pump locked rotor event has been analyzed and shows that either the minimum DNB ratio is not less than the applicable critical heat flux correlation limit, or fuel cladding was shown to experience no significant temperature excursions.

Steady state DNB analysis has been performed for four, three, and two pump combinations. For four pump operation, the steady state DNB analysis, which generates the pressure and temperature SL (i.e., the departure from nucleate boiling ratio (DNBR) limit), assumes a maximum power level of 112% RTP. This is the design overpower condition for four pump operation. The 112% value is the accident analysis limit of the nuclear overpower (high flux) trip and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR that protects the critical heat flux correlation limit.

The three pump pressure temperature limit is tied to the steady state DNB analysis, which is evaluated each cycle. The flow used is the minimum allowed for three pump operation. The actual RCS flow rate will exceed the assumed flow rate. With three pumps operating, overpower protection is automatically provided by the RPS nuclear overpower RCS flow and measured AXIAL POWER IMBALANCE Function. The maximum power level for three pump operation is identified in the COLR and is based on the three pump flow as a fraction of the four pump flow at full power.

Although the Specification limits operation to a minimum of three pumps total, design evaluation (including analyses at steady state, ECCS initial conditions, and DNB conditions) also shows that operation with one pump in each loop (two pumps total) is acceptable when core THERMAL POWER is restricted to be proportionate to the flow. However, continued power operation with two RCPs removed from service is restricted to 24 hours (Ref. 2) since not all transient and accident conditions have been analyzed.

RCS Loops-MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

The purpose of this LCO is to require adequate forced flow for core heat removal via two RCS loops. An operating loop consists of at least one operating RCP and a SG capable of heat removal. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power; if fewer pumps are available, power must be reduced as specified in the COLR.

APPLICABILITY

In MODES 1 and 2, the reactor may be critical and has the potential to produce maximum THERMAL POWER. To ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

Operation in other MODES is covered by:

- LCO 3.4.5, "RCS Loops-MODE 3";
- LCO 3.4.6, "RCS Loops-MODE 4";
- LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled";
- LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled";
- LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation-High Water Level" (MODE 6); and
- LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation-Low Water Level" (MODE 6).

ACTIONS

A.1

With one RCP not in operation in each loop, the assumptions of the safety analyses are not met, but design evaluation provided in Reference 2 concludes that events initiated during two pump operation would be expected to respond within the acceptance criteria for the ECCS. However, since no analysis was performed, Technical Specifications for two pump operation will only allow operation in MODES 1 or 2 for a period not to exceed 24 hours. The Completion Time of 18 hours provides sufficient time to restore operation of an additional RCP, while allowing time to place the unit in MODE 3 within the 24 hour limitation if restoration of a third RCP is not accomplished.

ACTIONS (continued)

B.1

If the Required Action and associated Completion Time of Condition A are not met, or if the LCO is not met for any reason other than provided in Condition A, the unit must be placed in a MODE in which the requirements are not applicable. This is accomplished by placing the unit in MODE 3. This reduces the core heat removal needs and minimizes the possibility of violating DNB limits. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification every 12 hours of the required number of loops in operation. Verification includes flow rate, temperature, or pump status monitoring, which helps ensure that forced flow is providing heat removal while maintaining the margin to DNB. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.

REFERENCES

1. SAR, Chapters 14 and 3A.
2. BAW-10103A, Revision 3, July 1977.
3. 10 CFR 50.36.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops - MODE 3

BASES

BACKGROUND

The primary function of the reactor coolant in MODE 3 is removal of decay heat and transfer of this heat, via the steam generators (SGs), to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 3, reactor coolant pumps (RCPs) are used to provide forced circulation for heat removal during heatup and cooldown. The number of RCPs in operation will vary depending on operational needs, and the intent of this LCO is to provide forced flow from at least one RCP for core heat removal and transport. The flow provided by one RCP is adequate for heat removal and for boron mixing. However, two RCS loops are required to be OPERABLE to provide redundant paths for heat removal.

Reactor coolant natural circulation is not normally used. If entry into natural circulation is required, the reactor coolant at the highest elevation of the hot leg must be maintained subcooled for single phase circulation. When in natural circulation, it is preferable to remove heat using both SGs to avoid idle loop stagnation that might occur if only one SG were in service. One generator will provide adequate heat removal. Boron reduction in natural circulation is prohibited because mixing to obtain a homogeneous concentration in all portions of the RCS cannot be ensured.

APPLICABLE SAFETY ANALYSES

No safety analyses are performed with initial conditions in MODE 3.

Failure to provide heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

RCS Loops-MODE 3 satisfy Criterion 4 of 10 CFR 50.36 (Ref. 1).

LCO

The purpose of this LCO is to require two loops to be available for heat removal thus providing redundancy. The LCO requires the two loops to be OPERABLE with the intent of requiring both SGs to be capable of transferring heat from the reactor coolant at a controlled rate. Forced reactor coolant flow is the preferred way to transport heat, although natural circulation flow is also acceptable under certain conditions. A minimum of one running RCP meets the LCO requirement for one loop in operation.

The Note permits a limited period of operation without RCPs. All RCPs may be removed from operation of ≤ 8 hours per 24 hour period for the transition to or from the Decay Heat Removal (DHR) System, and otherwise may be removed from operation for ≤ 1 hour per 8 hour period. During this condition, boron reduction with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, is prohibited because an even concentration distribution throughout the RCS cannot be ensured. Core outlet temperature is to be maintained at least 10°F below the saturation temperature so that: a) no vapor bubble may form and possibly cause a natural circulation flow obstruction; and b) pump restart criteria (which vary with pressure) are met.

In MODES 3, 4, and 5, it is sometimes necessary to stop all RCP or DHR pump forced circulation (e.g., change operation from one DHR train to the other, to perform surveillance or startup testing, to perform the transition to and from DHR System cooling, or to avoid operation below the RCP minimum net positive suction head limit). This is acceptable because the reactor coolant temperature can be maintained subcooled and boron stratification affecting reactivity control is not expected.

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE. To be considered OPERABLE, an RCP must be capable of being powered and able to provide forced flow if required. Similarly, an SG must be capable of transferring heat from the reactor coolant at a controlled rate and be in compliance with the Steam Generator Tube Surveillance Program.

APPLICABILITY

In MODE 3, the heat load is lower than at power; therefore, one RCS loop in operation is adequate for transport and heat removal. A second RCS loop is required to be OPERABLE but not in operation for redundant heat removal capability.

- Operation in other MODES is covered by:
- LCO 3.4.4, "RCS Loops-MODES 1 and 2";
- LCO 3.4.6, "RCS Loops-MODE 4";

APPLICABILITY (continued)

- LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled";
- LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled";
- LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation-High Water Level" (MODE 6); and
- LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation-Low Water Level" (MODE 6).

ACTIONS

A.1

If one RCS loop is inoperable, redundancy for forced flow heat removal is lost. The Required Action is restoration of the RCS loop to OPERABLE status within a Completion Time of 72 hours. This time allowance is a justified period to be without the redundant nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core.

B.1

If the Required Action and associated Completion Time of Condition A are not met, the unit must be brought to MODE 4. In MODE 4, the unit may be placed on the DHR System. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to achieve cooldown and depressurization from the existing unit conditions and without challenging unit systems.

C.1 and C.2

If no RCS loop is OPERABLE or a required RCS loop is not in operation, (no RCS loop is required to be in operation provided the conditions of the Note in the LCO section are met), all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be immediately suspended. Action to restore one RCS loop to operation shall be immediately initiated and continued until one RCS loop is restored to operation and to OPERABLE status. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification every 12 hours that the required loop (and pump) is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess RCS loop status. In addition, control room indication and alarms will normally indicate loop status.

SR 3.4.5.2

Verification that each required RCP is OPERABLE ensures that an RCS loop can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to each required pump that is not in operation. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES

1. 10 CFR 50.36.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the steam generators (SGs) or decay heat removal (DHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 4, either reactor coolant pumps (RCPs) or DHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RCP or one DHR pump for decay heat removal and transport. The flow provided by one RCP or one DHR pump is adequate for heat removal. The other intent of this LCO is to require that two paths (loops) be available to provide redundancy for heat removal.

APPLICABLE SAFETY ANALYSES

No safety analyses are performed with initial condition in MODE 4.

RCS Loops-MODE 4 satisfies Criterion 4 of 10 CFR 50.36 (Ref. 1).

LCO

The purpose of this LCO is to require that two loops, RCS or DHR, be OPERABLE in MODE 4 and one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS or DHR System loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. The second loop that is required to be OPERABLE provides redundant paths for heat removal.

The Note permits a limited period of operation with the normally required RCP or DHR pump removed from operation. The Note prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained when forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained below saturation temperature by ≥ 10°F so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

LCO (continued)

When the DHR pumps are stopped, no alternate heat removal path exists, unless the RCS and SGs have been placed in service in forced or natural circulation. The response of the RCS without heat removal through the DHR System or the SGs depends on the core decay heat load and the length of time that the DHR pumps are stopped. As decay heat diminishes, the effects on RCS temperature and pressure diminish. Without cooling by DHR, if the SGs are not capable of removing heat, higher heat loads will cause the reactor coolant temperature and pressure to increase at a rate proportional to the decay heat load. Because pressure can increase, the applicable system pressure limits (pressure and temperature (P/T) or low temperature overpressure protection (LTOP) limits) must be observed and forced DHR flow or heat removal via the SGs must be re-established prior to reaching the pressure limit.

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an OPERABLE SG. To be considered OPERABLE, an SG must be capable of transferring heat from the reactor coolant at a controlled rate and be in compliance with the Steam Generator Tube Surveillance Program.

Similarly for the DHR System, an OPERABLE DHR loop is comprised of the OPERABLE DHR pump(s) capable of circulating RCS fluid through the DHR heat exchanger(s) and back to the RCS. To be considered OPERABLE, a DHR pump must be capable of being powered and able to provide flow if required, and a DHR heat exchanger must be capable of transferring heat from the reactor coolant at a controlled rate.

A DHR loop may be considered OPERABLE during alignment and when aligned for low pressure injection if it is capable of being manually (locally or remotely) realigned to the DHR mode of operation and is not otherwise inoperable. This provision arises because of the dual requirements of the components that comprise the low pressure injection/decay heat removal system.

APPLICABILITY

In MODE 4, this LCO applies because it is possible to remove core decay heat and to provide proper boron mixing with either the RCS loops and SGs or the DHR System.

Operation in other MODES is covered by:

- LCO 3.4.4, "RCS Loops-MODES 1 and 2";
- LCO 3.4.5, "RCS Loops-MODE 3";
- LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled";

APPLICABILITY (continued)

- LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled";
- LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation-High Water Level" (MODE 6); and
- LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation-Low Water Level" (MODE 6).

ACTIONS

A.1

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

A.2

If restoration is not accomplished and a DHR loop is OPERABLE, the unit must be brought to MODE 5 within the following 24 hours. Bringing the unit to MODE 5 is a conservative action with regard to decay heat removal. With only one DHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining DHR loop, it would be safer to initiate that loss from MODE 5 rather than MODE 4. The Completion Time of 24 hours is reasonable, based on operating experience, to reach MODE 5 in an orderly manner and without challenging unit systems.

This Required Action is modified by a Note which indicates that the unit must be placed in MODE 5 only if a DHR loop is OPERABLE. With no DHR loop OPERABLE, the unit is in a condition with only limited cooldown capabilities. Therefore, the actions are to be concentrated on restoration of a DHR loop, rather than a cooldown of extended duration.

B.1 and B.2

If no RCS or DHR loops are OPERABLE or a required loop is not in operation (no loop is required to be in operation provided the conditions of the Note in the LCO section are met), all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RCS or DHR loop to OPERABLE status and operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into

ACTIONS (continued)

B.1 and B.2 (continued)

the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must continue until one loop is restored to operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This Surveillance requires verification every 12 hours of the required DHR or RCS loop in operation to ensure forced flow is providing decay heat removal. Verification includes flow rate, temperature, or pump status monitoring. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess RCS loop status. In addition, control room indication and alarms will normally indicate loop status.

SR 3.4.6.2

Verification that each required pump is OPERABLE ensures that an RCS or DHR loop can be placed in operation if needed to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other administrative controls and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES

1. 10 CFR 50.36.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat either to the steam generator (SG) secondary side coolant or the service water via the decay heat removal (DHR) heat exchangers. While the principal means for decay heat removal is via the DHR System, the SGs are specified as a backup means for redundancy. Although the SGs do not typically remove heat unless steaming occurs, they are available as a temporary heat sink and can be used by allowing the RCS to heat up into the temperature region of MODE 4 where steaming can be effective for heat removal. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with RCS loops filled, DHR loops are the principal means for heat removal. The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one DHR loop for decay heat removal and transport. The flow provided by one DHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide a backup method for heat removal.

APPLICABLE SAFETY ANALYSES

No safety analyses are performed with initial conditions in MODE 5.

RCS Loops-MODE 5 (Loops Filled) satisfies Criterion 4 of 10 CFR 50.36 (Ref. 1).

LCO

The purpose of this LCO is to require that at least one of the DHR loops be OPERABLE and in operation with an additional DHR loop OPERABLE or both SGs with secondary side water level ≥ 20 inches. One DHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. The second DHR loop is normally maintained as a backup to the operating DHR loop to provide redundancy for decay heat removal. However, if the standby DHR loop is not OPERABLE, a sufficient alternate method of providing redundant heat removal paths is to provide both SGs with their secondary side water levels ≥ 20 inches. Should the operating DHR loop fail, the SGs could be used to remove the decay heat.

LCO (continued)

The LCO provides for either SG heat removal or DHR System heat removal. In this MODE, reactor coolant pump (RCP) operation may be restricted because of net positive suction head (NPSH) limitations, and the SG will not be able to provide steam for the turbine driven feed pumps. However, to ensure that the SG(s) can be used as a heat sink, a motor driven feedwater pump is needed, because it is independent of steam. Condensate pumps, the auxiliary feedwater pump, or the motor driven emergency feedwater pump can be used. If RCPs are available, the steam generator level need not be adjusted. If RCPs are not available, the water level must be adjusted for natural circulation. The high entry point in the generator should be accessible from the feedwater pumps so that natural circulation can be stimulated. The SGs are primarily a backup to the DHR pumps, which are used for forced flow. By requiring the SGs to be a backup heat removal path, the option to increase RCS pressure and temperature for heat removal in MODE 4 is provided.

Note 1 permits the DHR pumps to be stopped for up to 1 hour. The circumstances for stopping both DHR trains are to be limited to situations where: (a) Pressure and temperature increases can be maintained well within the allowable pressure (P/T and low temperature overpressure protection) and 10°F subcooling limits; and (b) no operations are in process that would cause reduction of the RCS boron concentration.

The Note prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained when DHR forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained below saturation temperature by ≥ 10°F so that no vapor bubble would form and possibly cause a natural circulation flow obstruction. In this MODE, the steam generators are used as a backup for decay heat removal and, to ensure their availability, the RCS loop flow path is to be maintained with subcooled liquid.

In MODE 5, it is sometimes necessary to stop all RCP or DHR pump forced circulation. For example, this may be necessary to change operation from one DHR train to the other, perform surveillance or startup testing, perform the transition to and from the DHR System, or to avoid operation below the RCP minimum NPSH limit. The time period is acceptable because the reactor coolant temperature can be maintained subcooled, and boron stratification affecting reactivity control is not expected.

Note 2 allows one required DHR loop to be inoperable for a period of ≤ 2 hours provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 3 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of DHR loops from operation when at least one RCP is in operation. This Note provides for the transition to MODE 4 where an RCP is permitted to be in operation and replaces the RCS circulation function provided by the DHR loops.

LCO (continued)

A DHR loop may be considered OPERABLE during alignment and when aligned for low pressure injection if it is capable of being manually (locally or remotely) realigned to the DHR mode of operation and is not otherwise inoperable. This provision arises because of the dual requirements of the components that comprise the low pressure injection/decay heat removal system.

An OPERABLE DHR loop is composed of an OPERABLE DHR pump and an OPERABLE DHR heat exchanger.

To be considered OPERABLE, DHR pumps must be capable of being powered and are able to provide flow if required. During performance of SR 3.8.1.7 or SR 3.8.1.8, the affected DHR pump may be considered OPERABLE even with the breaker "racked down" since placing this second pump in operation is a manual action. Similarly, an OPERABLE SG can perform as a heat sink when it has an adequate water level and is in compliance with the Steam Generator Tube Surveillance Program.

APPLICABILITY

In MODE 5 with loops filled, forced circulation is provided by this LCO to remove decay heat from the core and to provide proper boron mixing. One loop of DHR provides sufficient circulation for these purposes.

Operation in other MODES is covered by:

- LCO 3.4.4, "RCS Loops-MODES 1 and 2";
- LCO 3.4.5, "RCS Loops-MODE 3";
- LCO 3.4.6, "RCS Loops-MODE 4";
- LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled";
- LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation-High Water Level" (MODE 6); and
- LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation-Low Water Level" (MODE 6).

ACTIONS

A.1, A.2, B.1, and B.2

If one required DHR loop is inoperable and any required SG has secondary side water level < 20 inches, redundancy for heat removal is lost. Action must be initiated to restore a second DHR loop to OPERABLE status or initiate action to restore the secondary side water level in the SG(s), and action must be taken immediately. Either Required Action will restore redundant decay heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

C.1 and C.2

If no required DHR loop is in operation, except as provided in Note 1, or no required DHR loop is OPERABLE, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore a DHR loop to OPERABLE status and operation must be initiated. The margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that the required DHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation. In addition, control room indication and alarms will normally indicate loop status.

SR 3.4.7.2

Verifying the SGs are OPERABLE by ensuring their secondary side water levels are ≥ 20 inches ensures that redundant heat removal paths are available if the second DHR loop is not OPERABLE. If both DHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess RCS loop status.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.7.3

Verification that each required DHR pump is OPERABLE ensures that a DHR loop can be placed in operation if needed to maintain decay heat removal and reactor coolant circulation. If the secondary side water level is ≥ 20 inches in both SGs, this Surveillance is not needed. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES

1. 10 CFR 50.36.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops - MODE 5, Loops Not Filled

BASES

BACKGROUND

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

Loops are considered not filled when the RCS draining is initiated (as might be the case for refueling or maintenance). Additionally, reductions of RCS inventory below el. 375 ft. are termed reduced inventory operations. GL 88-17 (Ref. 1) expresses concerns for loss of decay heat removal for this operating condition. With water at this low level, the margin above the decay heat suction piping connection to the hot leg is small. The possibility of loss of level or inlet vortexing exists and if it were to occur, the operating DHR pump could become air bound and fail resulting in a loss of forced flow for heat removal. As a consequence the water in the core will heat up and could boil with the possibility of core uncovering due to boil off.

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

APPLICABLE SAFETY ANALYSES

No safety analyses are performed with initial conditions in MODE 5 with loops not filled.

RCS Loops-MODE 5 (Loops Not Filled) satisfies Criterion 4 of 10 CFR 50.36 (Ref. 2).

LCO

The purpose of this LCO is to require that a minimum of two DHR loops be OPERABLE and that one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the DHR system unless forced flow is used. A minimum of one running decay heat removal pump meets the LCO requirement for one loop in operation. An additional DHR loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits the DHR pumps to be de-energized for ≤ 1 hour. The Note prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained or draining operations when DHR forced flow is stopped.

Note 2 allows one DHR loop to be inoperable for a period of ≤ 2 hours provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during MODE 5 when these tests are safe and possible.

A DHR loop may be considered OPERABLE during alignment and when aligned for low pressure injection if it is capable of being manually (locally or remotely) realigned to the DHR mode of operation and is not otherwise inoperable. This provision arises because of the dual requirements of the components that comprise the low pressure injection/decay heat removal system.

An OPERABLE DHR loop is composed of an OPERABLE DHR pump capable of circulating RCS fluid through an OPERABLE DHR heat exchanger and back to the RCS. To be considered OPERABLE, the DHR pumps must be capable of being powered and able to provide flow if required. During performance of SR 3.8.1.7 or SR 3.8.1.8, the affected DHR pump may be considered OPERABLE even with the breaker "racked down" since placing this second pump in operation is a manual action.

APPLICABILITY

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

Operation in other MODES is covered by:

- LCO 3.4.4, "RCS Loops-MODES 1 and 2";
- LCO 3.4.5, "RCS Loops-MODE 3";
- LCO 3.4.6, "RCS Loops-MODE 4";

APPLICABILITY (continued)

- LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled";
- LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation-High Water Level" (MODE 6); and
- LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation-Low Water Level" (MODE 6).

ACTIONS

A.1

If one required DHR loop is inoperable, redundancy for heat removal is lost. Required Action A.1 is to immediately initiate activities to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1, B.2, and B.3

If no required loop is OPERABLE or the required loop is not in operation, except as provided by Note 1 in the LCO, the Required Action requires immediate suspension of all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 or reduction of RCS water inventory and requires initiation of action to immediately restore one DHR loop to OPERABLE status and operation. The Required Action for restoration does not apply to the condition of both loops not in operation when the exception Note in the LCO is in force. Suspending the introduction of coolant into the RCS of coolant with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operations for decay heat removal. The action to restore must continue until one loop is restored.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This Surveillance requires verification every 12 hours that at least one loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess RCS loop status.

SR 3.4.8.2

Verification that each required pump is OPERABLE ensures that redundancy for heat removal is provided. The requirement also ensures that a DHR loop can be placed in operation if needed to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES

1. Generic Letter 88-17, October 17, 1988.

1. 10 CFR 50.36.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters, and their controls. Pressurizer safety valves are addressed by LCO 3.4.10, "Pressurizer Safety Valves."

The maximum water level limit has been established to ensure that a liquid to vapor interface exists to permit RCS pressure control during normal operation and proper pressure response for abnormalities. The water level limit thus serves two purposes:

a. Provides pressure control during normal operation; and

b. Prevents the peak RCS pressure from exceeding the safety limit of 2750 psig during an abnormality.

The maximum water level limit thus permits pressure control equipment to function as designed. The limit preserves the steam space during normal operation, so that both sprays and heaters can operate to maintain the design operating pressure. The level limit also prevents filling the pressurizer (water solid) during abnormalities, thus ensuring that pressure relief devices (electromatic relief valve (ERV) or code safety valves) can control pressure by steam relief rather than water relief. If the level limits were exceeded prior to an abnormality that creates a large pressurizer insurge volume leading to water relief, the maximum RCS pressure might exceed the design Safety Limit (SL) of 2750 psig or damage may occur to the ERV or pressurizer code safety valves.

The minimum water level limit has been established to ensure that water level is above the minimum detectable level.

The pressurizer heaters are used to maintain a pressure in the RCS so reactor coolant in the loops is subcooled and thus in the preferred state for heat transport to the steam generators (SGs). This function must be maintained with a loss of offsite power. Consequently, the emphasis of this LCO is to ensure that the Engineered Safeguards (ES) bus powered heaters are adequate to maintain pressure for RCS loop subcooling with an extended loss of offsite power.

BACKGROUND (continued)

A minimum required available capacity of 126 kW ensures that the RCS pressure can be maintained. Unless adequate heater capacity is available, reactor coolant subcooling may not be maintained (although the pressure control provided by the high head high pressure injection pumps is an alternate method of maintaining subcooling). Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to loss of single phase natural circulation and decreased capability to remove core decay heat.

APPLICABLE SAFETY ANALYSES

In MODES 1 and 2, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. No safety analyses are performed in lower MODES. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present.

Safety analyses presented in the SAR do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

The maximum level limit is provided to prevent the peak RCS pressure from exceeding the safety limit of 2750 psig in the event of a rod withdrawal accident or a startup accident. Assuming proper response by reactor protection systems, the level limit prevents water relief through the pressurizer safety valves. If the level limits were exceeded prior to an abnormality that creates a large pressurizer insurge volume leading to water relief, the maximum RCS pressure might exceed the design SL of 2750 psig or damage may occur to the ERV or pressurizer code safety valves. The value for pressurizer level is the safety analysis value. Therefore, the implementing procedures must contain allowances for instrument error.

The requirement for emergency power supplies is based on NUREG-0578 (Ref. 1), item 2.1.1. The intent is to maintain the reactor coolant in a subcooled condition with natural circulation at hot, high pressure conditions for an extended time period after a loss of offsite power. While loss of offsite power is an initial condition or coincident event assumed in many accident analyses, maintaining hot, high pressure conditions over an extended time period is not evaluated as part of SAR accident analyses.

In MODES 1 and 2, the maximum pressurizer water level limit satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2). In MODE 3 and MODE 4 above the LTOP enable temperature, the maximum pressurizer water level limit satisfies Criterion 4 of 10 CFR 50.36. Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0578 (Ref. 1), is the reason for providing an LCO. Therefore, the pressurizer heaters satisfy Criterion 4 of 10 CFR 50.36 (Ref. 2).

LCO

The LCO requirement for the pressurizer to be OPERABLE with a water level $\geq$ 45 inches and $\leq$ 320 inches ensures that a steam bubble exists prior to criticality and that the indication of the level is above the minimum detectable level. Limiting the maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

The LCO requires a minimum of 126 kW of pressurizer heaters OPERABLE. To be considered OPERABLE, the required heaters must be powered from an ES bus. This provides assurance that sufficient heater capacity is available to provide RCS pressure control during a loss of off-site power. The amount needed to maintain pressure is dependent on the insulation losses, which can vary due to tightness of fit and condition.

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus Applicability has been designated for MODES 1 and 2. The Applicability is also provided for MODE 3 and, for pressurizer water level, for MODE 4 with RCS temperature > 262°F. The purpose is to prevent water solid RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbations, such as reactor coolant pump startup. The temperature of 262°F has been designated as the cutoff for applicability because LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)," provides a requirement for pressurizer level at or below 262°F. The LCO does not apply to MODE 5 with loops filled because LCO 3.4.11 applies and provides adequate overpressure protection. This parameter value does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures. The LCO does not apply to MODES 5 and 6 with partial loop operation.

In MODES 1, 2, and 3, there is the need to maintain the availability of pressurizer heaters capable of being powered from an emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. The Applicability is modified by a Note stating that the OPERABILITY requirements on pressurizer heaters do not apply in MODE 4. For MODE 4, 5, or 6, the need to control pressure (by heaters) to ensure loop subcooling for heat transfer is significantly reduced when the Decay Heat Removal System is in service, and therefore the LCO is not applicable.

ACTIONS

A.1

With pressurizer water level outside the limits, action must be taken to restore pressurizer operation to within the bounds assumed in the analysis. This is done by restoring the pressurizer water level to within the limits. The 1 hour Completion Time is considered to be a reasonable time for adjusting pressurizer level.

B.1 and B.2

If the water level cannot be restored, reducing core power constrains heat input effects that drive pressurizer insurge that could result from an anticipated transient. By shutting down the reactor and reducing reactor coolant temperature to at least MODE 3, the potential thermal energy of the reactor coolant mass for mass and energy releases is reduced.

Six hours is a reasonable time based upon operating experience to reach MODE 3 from full power in an orderly manner and without challenging unit systems. Further pressure and temperature reduction to MODE 4 with RCS temperature ≤ 262°F places the unit into a MODE where the LCO is not applicable. The 24 hour Completion Time to reach the nonapplicable MODE is reasonable based upon operating experience.

C.1

If the required pressurizer heaters are inoperable, restoration is required in 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand caused by loss of offsite power will not occur in this period. Pressure control may be maintained during this time using non-ES bus powered heaters.

D.1 and D.2

If the Required Action and associated Completion Time are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to MODE 3 within 6 hours and to MODE 4 within the following 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 unit systems. Similarly, the Completion Time of 12 hours to reach MODE 4 is reasonable based on operating experience to achieve power reduction from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that pressurizer water level is maintained below the upper limit to provide a minimum space for a steam bubble. The values specified for pressurizer level do not contain an allowance for instrument error. Therefore, additional allowances for instrument uncertainties must be provided in the implementing procedures. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess the level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level.

SR 3.4.9.2

The SR requires sufficient pressurizer heaters which are connected to an ES bus verified to be capable of providing the required capacity. (This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance.) The Frequency of 18 months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

REFERENCES

1. NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," July 1979.
2. 10 CFR 50.36.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The purpose of the two spring loaded pressurizer safety valves is to provide RCS overpressure protection (Ref. 1). Operating in conjunction with the Reactor Protection System (RPS), two valves are used to ensure that the Safety Limit (SL) of 2750 psig is not exceeded for analyzed transients during operation in MODES 1 and 2. One safety valve is required for MODE 3 and portions of MODE 4. For the remainder of MODE 4, MODE 5, and MODE 6 with the reactor head on, overpressure protection is provided by operating procedures and LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)."

The self actuated pressurizer safety valves are designed in accordance with the requirements set forth in the ASME Boiler and Pressure Vessel Code, Section III (Ref. 2). The required lift pressure is 2500 psig + 1%, - 3%. The safety valves discharge steam from the pressurizer to a quench tank located in the reactor building. The discharge flow is indicated by acoustic flow monitoring devices, by an increase in temperature downstream of the safety valves, and by an increase in the quench tank temperature, pressure, and level.

The upper and lower as-left pressure limits are based on the ± 1% tolerance requirement for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the ASME pressure limit could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE SAFETY ANALYSES

The overpressure protection analysis (Ref. 3) is based on operation of both safety valves and assumes that the valves open at the high range of the setting (2500 psig system design pressure plus 1%). One pressurizer code safety valve is capable of preventing overpressurization in MODE 3 and in MODE 4 with RCS temperature > 262°F since its relieving capacity is greater than that required by the sum of the available heat sources, i.e., pump energy, pressurizer heaters, and reactor decay heat (Ref. 1 and 4). These valves must accommodate pressurizer insurges that

APPLICABLE SAFETY ANALYSES (continued)

could occur during a startup, rod withdrawal, or ejected rod event. The startup accident establishes the minimum safety valve capacity. The startup accident is assumed to occur at low power. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this Specification is required to ensure that the accident analysis and design basis calculations remain valid.

In MODES 1 and 2, pressurizer safety valves satisfy Criterion 3 of the 10 CFR 50.36 (Ref. 5). In MODE 3 and MODE 4 above the LTOP enable temperature, the pressurizer safety valves satisfy Criterion 4 of 10 CFR 50.36.

LCO

The two pressurizer safety valves are set to open at the RCS design pressure (2500 psig) and within the specified tolerance to avoid exceeding the maximum RCS design pressure SL, to maintain accident analysis assumptions and to comply with ASME Code requirements. The upper and lower as-left pressure tolerance limits are based on the ± 1% tolerance requirements (Ref. 2) for lifting pressures above 1000 psig. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or both valves could result in exceeding the SL if a transient were to occur.

The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

The LCO is modified by four Notes. Note 1 states that in MODE 3 and MODE 4 with RCS temperature above 262°F, only one pressurizer safety valve is required to be OPERABLE. In this condition, one pressurizer safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than the sum of the available heat sources.

Note 2 allows entry into MODE 3, and into MODE 4 with RCS temperature > 262°F, with the lift settings potentially outside the limits. This permits testing of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 36 hour exception is based on an 18 hour outage time for each of the two valves. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe.

LCO (continued)

Note 3 states that the LCO is not applicable in MODE 3, and in MODE 4 with RCS temperature > 262°F during hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III. During hydrostatic tests, the code safeties must be gagged to prevent them from relieving at the target test pressure. RCS pressure is carefully observed and compensatory measures are in place to provide assurance that the pressure is appropriately controlled during the performance of hydrostatic tests.

Note 4 states that the provisions of LCO 3.0.3 are not applicable in MODE 3, and in MODE 4 with RCS temperature > 262°F. In the event no code safety valve is OPERABLE in this MODE, the Required Actions ensure that the RCS is placed in a condition in which the ERV is capable of relieving any potential LTOP pressure transient.

The parameter value (262°F) does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP enable temperature, OPERABILITY of pressurizer safety valve(s) is required to ensure adequate relieving capacity is available to keep reactor coolant pressure below 110% of its design value during certain accidents.

The LCO is not applicable in MODE 4 with RCS temperature ≤ 262°F, in MODE 5, nor in MODE 6 when the reactor vessel head is on because LTOP protection is provided. Overpressure protection is not required in MODE 6 with the reactor vessel head removed.

The parameter value (262°F) does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

ACTIONS

A.1

With one pressurizer safety valve inoperable in MODES 1 and 2, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS overpressure protection system. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the RCPB.

ACTIONS (continued)

B.1

If the Required Action and associated Completion Time of Condition A are not met, or if both pressurizer safety valves are inoperable in MODES 1 and 2, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours. The 6 hours allowed is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by two pressurizer safety valves.

C.1

With the required pressurizer code safety valve inoperable, the RCS overpressure protection capability is significantly reduced and an overpressure event could challenge the integrity of the RCPB. Therefore, the unit must be placed in a condition in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 4 with RCS temperature at or below the LTOP enable temperature within 18 hours. The 18 hours allowed is reasonable, based on operating experience, to reach a low temperature within MODE 4 without challenging unit systems. With RCS temperature at or below 262°F, overpressure protection is provided by LTOP.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 6), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is + 1%, - 3% for OPERABILITY (Ref. 7); however, the valves are reset to ± 1% during the Surveillance to allow for drift.

REFERENCES

1. SAR, Section 4.2.4.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1968.

3. SAR, Section 4.3.8.

4. SAR, Section 4.3.11.4.

5. 10 CFR 50.36.

6. ASME, Boiler and Pressure Vessel Code, Section XI.

7. ASME/ANSI, Operations and Maintenance Codes (OM), Part 10, 1987, Part 10 Addenda, 1988, and Part 1, 1987.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP controls prevent RCS overpressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) requirements of 10 CFR 50, Appendix G (Ref. 1) as modified by approved exemptions. The reactor vessel is the limiting RCPB component requiring such protection. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," provides the allowable combinations for pressure and temperature during cooldown, shutdown, and heatup to keep from violating the Reference 1 limits.

The reactor vessel material is less tough at reduced temperatures than at normal operating temperature. Also, as vessel neutron irradiation accumulates, the material becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure must be maintained low when temperature is low and may be increased only as temperature is increased.

Operational maneuvering during cooldown, heatup, or any anticipated operational occurrence must be controlled to not violate LCO 3.4.3. Exceeding these limits could lead to brittle fracture of the reactor vessel. LCO 3.4.3 presents requirements for administrative control of RCS pressure and temperature to prevent exceeding the P/T limits.

This LCO provides RCS overpressure protection in the applicable MODES by ensuring an adequate pressure relief capacity and a minimum coolant addition capability. The pressure relief capacity requires the (power operated) electromatic relief valve (ERV) to be OPERABLE with the lift setpoint reduced and pressurizer coolant level at or below a maximum limit for the RCS pressure, or the RCS depressurized and with an RCS vent of sufficient size to handle the limiting LTOP transient.

The LTOP approach to protecting the vessel by limiting coolant addition capability requires deactivating HPI, and isolating the core flood tanks (CFTs).
Should an HPI pump inject on an HPI actuation, the pressurizer level and ERV or another RCS vent may not prevent overpressurizing the RCS. As indicated in Reference 3, the deactivation of HPI injection capability, along with the LTOP alarms, provides sufficient basis for excluding the inadvertent actuation of HPI as a design basis event. Additionally, the CFT controls preclude the inadvertent mass input from the CFT. Finally, maintaining the pressurizer level to prevent operation in a water solid condition with the RCS pressure boundary intact provides a compressible vapor space or cushion (either steam or nitrogen) that can

BACKGROUND (continued)

accommodate a coolant insurge and prevent a rapid pressure increase, allowing the operator time to stop the increase. The ERV, with reduced lift setting, or the RCS vent is the overpressure protection device that acts as backup to the operator in terminating an increasing pressure event.

With HPI deactivated, the ability to provide RCS coolant addition is restricted. To allow for coolant addition, the LCO does not require the makeup function to be deactivated. Due to the lower pressures associated with the LTOP MODES and the expected decay heat levels, the makeup function can provide flow through the makeup control valve.

ERV Requirements

As designed for the LTOP, the ERV is signaled to open if the RCS pressure reaches a limit set in the LTOP actuation circuit. The LTOP actuation circuit monitors RCS pressure and determines when an overpressure condition is approached. When the monitored pressure meets or exceeds the setting, the ERV is signaled to open. Maintaining the lowered setpoint ensures the Reference 1 limits will be met in any event analyzed for LTOP.

RCS Vent Requirements

Once the RCS is depressurized, adequate pressure relief capability may be provided by a vent path to the reactor building atmosphere which is capable of relieving the flow of the limiting LTOP transient and maintaining pressure below P/T limits. The required vent capacity may be provided by one or more vent paths. Acceptable RCS vent paths include any of the following: removing a pressurizer safety valve, locking the ERV in the open position and disabling its block valve in the open position, or similarly establishing a vent by removing a steam generator (SG) primary manway, removing a SG primary hand hole cover, removing all control rod drive top closure assemblies (excluding reactor vessel level probe), or removing a pressurizer manway. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

APPLICABLE SAFETY ANALYSES

Safety analyses (Refs. 4, 5, 6, and 7) demonstrate that the reactor vessel can be adequately protected against overpressurization transients during shutdown. The pressure and temperature limits are derived from fracture mechanics analyses. Transients are then evaluated to determine a required ERV setpoint and other unit conditions that will ensure that the P/T limits are not exceeded.

Fracture mechanics analyses (using the safety margins of Reference 8) established the temperature of LTOP Applicability at 262°F. Above this temperature, the pressurizer safety valves provide the reactor vessel overpressure protection. The actual temperature at which the allowable pressure falls below the pressurizer

APPLICABLE SAFETY ANALYSES (continued)

safety valve setpoint increases as vessel material ductility decreases due to neutron embrittlement. P/T limits are periodically determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3 discuss these examinations. For the current limits, vessel materials are assumed to have a neutron irradiation accumulation equivalent to 31 effective full power years (EFPYs) of operation. Each time the P/T limit curves are revised, the LTOP is re-evaluated to ensure that its functional requirements can still be met. The ERV setpoint is revised if necessary.

Transients that are capable of overpressurizing the RCS at low temperature result in either excessive mass input or excessive heat input. Such transients include: HPI actuation, CFT discharge, energization of the pressurizer heaters, failing the makeup control valve open, loss of decay heat removal, starting a reactor coolant pump (RCP) with a large temperature mismatch between the primary and secondary coolant systems, and addition of nitrogen to the pressurizer. Without controls, HPI actuation and CFT discharge would be transients that result in exceeding P/T limits within the 10 minute period in which time no operator action can be assumed to take place. For the remaining events, operator action after that time precludes overpressurization.

This specification prevents exceeding the P/T limits by: 1) limiting the capability for rapid mass input to the RCS; and 2) ensuring that adequate vent capability exists to accommodate inadvertent mass or energy addition to the RCS. Pressurizer level is also limited to ensure that increasing pressure during a transient will be slow enough to preclude exceeding pressure limits within the 10 minutes assumed to be required for operator action to mitigate the transient. Mass input into the system is limited by disabling HPI (with specific exceptions) and by deactivating pressurized CFT discharge isolation valves in the closed position with their power breakers open (with specific exceptions). The analyses demonstrate that HPI transients involving one HPI pump can be accommodated by the ERV without exceeding the maximum allowable pressure.

The ERV setpoint is determined by modeling LTOP performance assuming the most limiting LTOP transient of a makeup control valve failing open. Pressure overshoot beyond the setpoint resulting from signal processing and valve stroke times is considered. The resulting ERV setpoint ensures the reference 1 limits will not be exceeded.

Vent capability is required to ensure that the maximum allowable pressure is not exceeded in the event of full opening of the makeup control valve while one makeup pump is running. Acceptable vent paths have adequate capacity at a system pressure of 100 psig which is less than the maximum RCS pressure on the P/T limit curve in LCO 3.4.3.

APPLICABLE SAFETY ANALYSES (continued)

The ERV is an active component. Therefore, its failure represents the worst case single active failure of LTOP features. The other vent paths are passive and not subject to active failure.

The LTOP satisfies Criterion 2 of 10 CFR 50.36 (Ref. 9).

LCO

The LCO requires an LTOP system OPERABLE with a limited coolant input capability and a pressure relief capability. To limit coolant input, the LCO requires the HPI deactivated, and the CFT discharge isolation valves closed and deactivated. For pressure relief, the LCO requires the pressurizer coolant level to be below a level which represents a water solid condition, and the ERV OPERABLE with a lowered lift setting or the RCS depressurized and a vent established.

HPI deactivation requires that the motor operated valves be closed and the opening control circuits for the motor operators disabled. CFT isolation requires the CFT discharge valves to be closed and the circuit breakers for the motor operators to be opened.

The HPI deactivation and CFT isolation requirements are modified by five Notes. Note 1 indicates that the requirements are not applicable during ASME Section XI testing. This exception provides for required testing during these shutdown conditions rather than at power when the HPI and CFTs are required to be OPERABLE for the ECCS function. Note 2 indicates that the requirements are not applicable for the HPI deactivation during fill and vent of the RCS. The HPI pumps are used for this normal makeup function and must be available. Specific procedural controls are provided to prevent overpressurization during this activity. Note 3 indicates that the requirements are not applicable for the HPI deactivation during emergency RCS makeup. This exception is necessary to enhance the response capability to a loss of decay heat removal event without violating the TS (Ref. 10). Note 4 indicates that the requirements are not applicable for the HPI deactivation during valve maintenance. This exception allows maintenance to be performed during these shutdown conditions rather than at power when the HPI is required to be OPERABLE for the ECCS function. Note 5 states that CFT isolation is only required when CFT pressure is more than or equal to the maximum RCS pressure for the existing RCS temperature, as allowed in LCO 3.4.3. This is acceptable since the CFT can not be the source of an overpressurization event when its pressure is less than the allowable RCS pressure.

The pressurizer is considered to represent a water solid condition when coolant level is > 105 inches, when RCS pressure is > 100 psig, or > 150 inches, when RCS pressure is ≤ 100 psig. Although a vapor space still exists with pressurizer level above these values, from an analytical point of view, the unit is considered to be water solid. These parameter values contain allowances for instrument error.

LCO (continued)

The pressurizer level requirements are modified by two Notes. Note 1 indicates that the requirements are not applicable during operation allowed by the Emergency Operating Procedures (EOPs). This exception provides for use of the "feed and bleed" process when necessary as determined by the EOPs. Note 2 indicates that the requirements are not applicable during RCS hydrotesting. Specific procedural controls are provided to prevent overpressurization during this activity.

OPERABLE pressure relief capability may be provided by an OPERABLE ERV, or by depressurizing the RCS and providing an alternate RCS vent path. For the ERV to be considered OPERABLE, its block valve must be open, its lift setpoint must be set at ≤ 460 psig, testing must have proven its ability to open at that setpoint, and motive power must be available to the ERV and its control circuits. With the RCS depressurized, acceptable alternate vent paths include removing a pressurizer safety valve, locking the ERV in the open position and disabling its block valve in the open position, removing a SG primary manway, removing a SG primary hand hole cover, removing all control rod drive top closure assemblies (excluding reactor vessel level probe), or removing a pressurizer manway.

APPLICABILITY

This LCO is applicable in MODE 4 with RCS temperature ≤ 262°F, in MODE 5, and in MODE 6 when the reactor vessel head is on. The Applicability temperature of 262°F is established by fracture mechanics analyses. The pressurizer safety valves provide overpressure protection to meet LCO 3.4.3 P/T limits above 262°F. With the vessel head off, overpressurization is not possible.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the pressurizer safety valves OPERABLE to provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above 262°F.

The parameter value (262°F) does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

ACTIONS

<u>A.1, B.1, and B.2</u>

With the pressurizer level not within its required limits, the time for operator action in a pressure increasing event is reduced. The postulated event most affected in the LTOP MODES is failure of the makeup control valve, which fills the pressurizer relatively rapidly. Restoration is required within 1 hour.

ACTIONS (continued)

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

If restoration within 1 hour in either case cannot be accomplished, Required Actions B.1 and B.2 must be performed within 12 hours to close the makeup control valve and its isolation valve. These Required Actions limit the makeup capability, which is not required with a high pressurizer level, and permit cooldown and depressurization to continue. Heatup must be stopped because heat addition decreases the reactor coolant density and increases the pressurizer level.

The Completion Times again are based on operating experience that these activities can be accomplished in these time periods and that a limiting LTOP transient is not likely in the allowed times.

C.1 and D.1

With the required ERV inoperable, overpressure relieving capability is lost, and restoration of the ERV within 1 hour is required. If that cannot be accomplished, the ability of the Makeup System to add water must be limited within the next 12 hours.

If restoration cannot be completed within 1 hour, Required Action D.1 must be performed to limit RCS water addition capability. Makeup is not deactivated to maintain the RCS coolant level. Required Action D.1 requires reducing the makeup tank level to ≤ 73 inches. This makes the available makeup water volume insufficient to exceed the LTOP limit by a makeup control valve full opening (Ref. 3). This parameter value does contain allowances for instrument error. No additional allowances for instrument error are required in the implementing procedures.

These Completion Times also consider these activities can be accomplished in these time periods. A limiting LTOP event is not likely in those times.

Some ERV testing or maintenance can only be performed at unit shutdown. Such activity is permitted if Required Action D.1 is taken to compensate for required ERV unavailability.

E.1

With the LTOP requirements not met for any reason other than cited in Condition A through D, action must be initiated to restore compliance immediately. The immediate Completion Time reflects the urgency of quickly proceeding with the Required Actions.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Verification of the pressurizer level at ≤ 105 inches when RCS pressure is > 100 psig or ≤ 150 inches when RCS pressure is ≤ 100 psig, by observing control room or other indications ensures that the unit is not in a water solid condition and that a cushion of sufficient size is available to reduce the rate of pressure increase from potential transients (Ref. 3).

The 30 minute Surveillance Frequency during heatup and cooldown must be performed for the LCO Applicability period when temperature changes can cause pressurizer level variations. This Frequency may be discontinued when these evolutions are complete, as defined in unit procedures. Thereafter, the Surveillance is required at 12 hour intervals.

These Frequencies are shown by operating practice sufficient to regularly assess indications of potential degradation and verify operation within the safety analysis.

SR 3.4.11.2 and SR 3.4.11.3

Verifications must be performed that the HPI is deactivated, and each pressurized CFT is isolated. These Surveillances ensure the minimum coolant input capability will not create an RCS overpressure condition to challenge the LTOP . The Surveillances are required at 12 hour intervals.

The 12 hour intervals are shown by operating practice to be sufficient to assess coolant input capability and verify operation within the safety analysis.

SR 3.4.11.4

OPERABLE pressure relief capability must be provided to prevent overpressurization due to inadvertent full makeup system operation. Such a vent keeps the pressure from full makeup flow within the LCO limit. OPERABLE pressure relief capability may be provided by an OPERABLE ERV, or by depressurizing the RCS and providing an alternate RCS vent path.

For the ERV to be considered OPERABLE, its block valve must be open, its lift setpoint must be set at ≤ 460 psig, testing must have proven its ability to open at that setpoint, and motive power must be available to the two valves and their control circuits. The parameter value of 460 psig does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.11.4 (continued)

With the RCS depressurized, acceptable alternate vent paths include: a) removing a pressurizer safety valve; b) locking the ERV in the open position and disabling its block valve in the open position; c) removing a SG primary manway; c) removing a SG primary hand hole cover; d) removing all control rod drive top closure assemblies (excluding reactor vessel level probe); and e) removing a pressurizer manway.

For a vent path not locked open, the Frequency is every 12 hours. For a locked open vent path, the required Frequency is every 31 days.

The Frequency intervals are considered adequate based on operating practice to determine adequacy of pressure relief capability and verify operation within the safety analysis.

SR 3.4.11.5

The performance of a CHANNEL CALIBRATION is required every 18 months. The CHANNEL CALIBRATION for the LTOP ERV opening logic, including the ERV setpoint, ensures that the ERV will be actuated at the appropriate RCS pressure by verifying the accuracy of the instrument string. The calibration can only be performed in shutdown.

The 18 month Frequency considers a typical refueling cycle and industry accepted practice.

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. ANO-1 LTOP Safety Evaluation Report (1CNA058302) dated May 5, 1983.
4. Response to NRC Request for Additional Information (1CAN117608) dated November 15, 1976.
5. Response to NRC Request for Additional Information (1CAN127602) dated December 3, 1976.
6. Response to NRC Request for Additional Information (1CAN037716) dated March 24, 1977.

REFERENCES (continued)

7. ANO-1 License Amendment Request (1CAN119608), dated November 26, 1988, and Operating License Amendment 188, (1CNA039703) dated March 14, 1997.

8. ANO-1 Request for Exemption (1CAN119608), dated November 26, 1996, and Exemption from Requirements of 10 CFR 50.60, (1CNA039702) dated March 12, 1997.

9. 10 CFR 50.36.

10. ANO-1 License Amendment Request (1CAN059008), dated May 22, 1990, and Operating License Amendment 138, (1CNA119002) dated November 1, 1990.