November 15, 2001

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-001

10 CFR 50.4

DOCKET 50-255 - LICENSE DPR-20 - PALISADES PLANT
REPORT OF CHANGES TO TECHNICAL SPECIFICATIONS BASES

This report is submitted in compliance with Palisades Technical Specification 5.5.12.d, which requires that changes to the Technical Specification Bases implemented without prior NRC approval be provided to the NRC on a frequency consistent with 10 CFR 50.71(e). Enclosure 1 provides a listing of all Bases changes since the issuance of Amendment No. 189 to Facility Operating License No. DPR-20 for the Palisades Plant, and identifies the affected sections and nature of the change. Enclosure 2 provides a copy of the current Technical Specifications Bases.

SUMMARY OF COMMITMENTS

This letter contains no new commitments and no revisions to existing commitments.

Laurie Lahti

Laurie A. Lahti,
Manager, Licensing

CC Regional Administrator, USNRC, Region III
Project Manager, USNRC, NRR
NRC Resident Inspector – Palisades

Enclosures

A001

ENCLOSURE 1

NUCLEAR MANAGEMENT COMPANY
PALISADES PLANT
DOCKET 50-255

November 15, 2001

TECHNICAL SPECIFICATIONS BASES
CHRONOLOGY

4 Pages

TECHNICAL SPECIFICATION BASES CHANGES
CHRONOLOGY

DATE	AFFECTED BASES SECTION(S)		CHANGE(S)
03/14/00	B3.7.5		Make Bases consistent with NRC approved Amendment 190 re: steam supply to turbine driven auxiliary feedwater (AFW) pump.
08/09/00	SR 3.02 B3.1.4 B3.1.6 B3.2.1 B3.2.4 B3.3.2 B3.3.9 B3.4.6 B3.4.8 B3.4.12 B3.4.14 B3.4.16 B3.6.2 B3.6.6 B3.7.2 B3.7.4 B3.7.6 B3.7.8 B3.8.1 B3.8.6 B3.9.1 B3.9.3 B3.9.5	B3.1.2 B3.1.5 B3.1.7 B3.2.3 B3.3.1 B3.3.3 B3.4.5 B3.4.7 B3.4.11 B3.4.13 B3.4.15 B3.5.2 B3.6.3 B3.6.7 B3.7.3 B3.7.5 B3.7.7 B3.7.13 B3.8.4 B3.8.9 B3.9.2 B3.9.4	Correction of typographical and grammar errors. Restore consistency of terminology. Removal of Standard Technical Specifications statements that were not correctly applied to Palisades. Clarify existing wording where questions had arisen.
10/02/00	B3.5.1	B3.5.2	Make Bases consistent with NRC approved Amendment 191 re: allowed outage time (AOT) for safety injection tanks (SITs) and single low pressure safety injection (LPSI) train.

TECHNICAL SPECIFICATION BASES CHANGES CHRONOLOGY

DATE	AFFECTED BASES SECTION(S)		CHANGE(S)
10/11/00	B3.6.6	B3.7.8	Changes resulting from recently completed main steam line break (MSLB) analysis. Maintain consistency between Technical Specification (TS) Bases and existing safety analyses.
10/24/00	B3.3.1		Clarify acceptance criteria for cross calibration of excore with incore detectors.
11/09/00	B3.6.7		Clarify hydrogen recombiner supporting air cooler fan requirements.
12/11/00	B3.7.7	B3.7.8	Clarify component cooling water (CCW) and service water system (SWS) operability with SWS isolated to one CCW heat exchanger. Define what equipment gives 100% required CCW and SWS post-accident cooling capability.
02/01/01	B3.3.1		Provide table of power and rod dependent axial offset/axial shape index (AO/ASI) deviation limits consistent with current analysis.
02/12/01	B3.3.1 B3.3.3 B3.4.6 B3.4.8 B3.6.2 B3.7.1 B3.7.9 B3.9.4	B3.3.2 B3.3.6 B3.4.7 B3.4.13 B3.6.6 B3.7.7 B3.9.2 B3.9.5	Correction of typographical and grammar errors. Maintain consistent terminology. Correct Table 3.3.1-1 to be consistent with referenced Limiting Conditions for Operation (LCOs) and other tables.
02/27/01	B3.9.6		Clarify iodine removal basis for requirements for minimum refueling cavity water level during movement of irradiated fuel.
03/30/01	B3.6.1 B3.6.3	B3.6.2	Make Bases consistent with NRC approved Amendment 194, re: provided for use of Option B of 10CFR50, Appendix J for containment penetration and airlock leakage testing.

TECHNICAL SPECIFICATION BASES CHANGES CHRONOLOGY

DATE	AFFECTED BASES SECTION(S)	CHANGE(S)
04/23/01	B3.2.1	Make Bases consistent with NRC approved PIDAL-3 methodology for the incore monitoring system.
04/27/01	B3.6.4 B3.6.5 B3.6.6	Update information based on revised calculations of containment response to MSLB and loss of coolant accident (LOCA). Correct discussion regarding what equipment gives 100% containment cooling capability.
05/04/01	B3.3.7	Make Bases consistent with NRC approved Amendment 193, re: eliminated post-accident sampling system (PASS) requirements from TS 5.5.3.
07/16/01	B3.7.9	Make Bases consistent with NRC approved Amendment 202, re: changed the maximum ultimate heat sink (UHS) temperature.
08/01/01	B3.5.2 B3.6.6 B3.7.5 B3.7.7 B3.7.8 B3.7.10	Make Bases consistent with NRC approved Amendment 197 re: control room envelope operability and AOT. Make Bases consistent with NRC approved Amendments 198, 199 and 200 re: reformatting 100% flow conditions for emergency core cooling system (ECCS)-Operating, CCW, SWS and AFW.
09/28/01	B2.1.1 B3.1.4 B3.1.6 B3.2.1 B3.2.2 B3.2.3	Make Bases consistent with NRC approved Amendment 205 re: deletion of assembly radial peaking factor.
11/06/01	B3.8.2 B3.8.5 B3.8.8 B3.8.9 B3.8.10	Clarify DC source, inverter and distribution requirements below MODE 4.

ENCLOSURE 2

NUCLEAR MANAGEMENT COMPANY
PALISADES PLANT
DOCKET 50-255

November 15, 2001

CURRENT
TECHNICAL SPECIFICATIONS BASES

LIST OF EFFECTIVE PAGES

COVERSHEET

	Title Page		205 - Revised 11/06/01

TABLE OF CONTENTS

	Page i		205
	Page ii		189

TECHNICAL SPECIFICATIONS BASES

Bases 2.0	Pages	B 2.1.1-1 - B 2.1.1-4	Revised 09/28/01
	Pages	B 2.1.2-1 - B 2.1.2-4	189
Bases 3.0	Pages	B 3.0-1 - B 3.0-14	189 - Revised 08/09/00
Bases 3.1	Pages	B 3.1.1-1 - B 3.1.1-5	189
	Pages	B 3.1.2-1 - B 3.1.2-6	189 - Revised 08/09/00
	Pages	B 3.1.3-1 - B 3.1.3-4	189
	Pages	B 3.1.4-1 - B 3.1.4-13	Revised 09/28/01
	Pages	B 3.1.5-1 - B 3.1.5-7	189 - Revised 08/09/00
	Pages	B 3.1.6-1 - B 3.1.6-9	Revised 09/28/01
	Pages	B 3.1.7-1 - B 3.1.7-6	189 - Revised 08/09/00
Bases 3.2	Pages	B 3.2.1-1 - B 3.2.1-11	Revised 09/28/01
	Pages	B 3.2.2-1 - B 3.2.2-3	Revised 09/28/01
	Pages	B 3.2.3-1 - B 3.2.3-3	Revised 09/28/01
	Pages	B 3.2.4-1 - B 3.2.4-3	189 - Revised 08/09/00
Bases 3.3	Pages	B 3.3.1-1 - B 3.3.1-29	189 - Revised 02/12/01
	Pages	B 3.3.1-30 - B 3.3.1-35	Revised 2/12/01
	Pages	B 3.3.2-1 - B 3.3.2-10	189 - Revised 02/12/01
	Pages	B 3.3.3-1 - B 3.3.3-24	189 - Revised 02/12/01
	Pages	B 3.3.4-1 - B 3.3.4-12	189
	Pages	B 3.3.5-1 - B 3.3.5-6	189
	Pages	B 3.3.6-1 - B 3.3.6-6	189 - Revised 02/12/01
	Pages	B 3.3.7-1 - B 3.3.7-12	Revised 05/04/01
	Pages	B 3.3.8-1 - B 3.3.8-6	189
	Pages	B 3.3.9-1 - B 3.3.9-5	189 - Revised 08/09/00
	Pages	B 3.3.10-1 - B 3.3.10-4	189
Bases 3.4	Pages	B 3.4.1-1 - B 3.4.1-5	189
	Pages	B 3.4.2-1 - B 3.4.2-2	189
	Pages	B 3.4.3-1 - B 3.4.3-7	189
	Pages	B 3.4.4-1 - B 3.4.4-4	189
	Pages	B 3.4.5-1 - B 3.4.5-5	189 - Revised 08/09/00
	Pages	B 3.4.6-1 - B 3.4.6-6	189 - Revised 02/12/01
	Pages	B 3.4.7-1 - B 3.4.7-7	189 - Revised 02/12/01
	Pages	B 3.4.8-1 - B 3.4.8-5	189 - Revised 02/12/01
	Pages	B 3.4.9-1 - B 3.4.9-6	189
	Pages	B 3.4.10-1 - B 3.4.10-4	189
	Pages	B 3.4.11-1 - B 3.4.11-7	189 - Revised 08/09/00
	Pages	B 3.4.12-1 - B 3.4.12-13	189 - Revised 08/09/00
	Pages	B 3.4.13-1 - B 3.4.13-6	189 - Revised 02/12/01
	Pages	B 3.4.14-1 - B 3.4.14-8	189 - Revised 08/09/00
	Pages	B 3.4.15-1 - B 3.4.15-6	189 - Revised 08/09/00
	Pages	B 3.4.16-1 - B 3.4.16-5	189 - Revised 08/09/00

LIST OF EFFECTIVE PAGES

Bases 3.5	Pages	B 3.5.1-1 - B 3.5.1-5	189
	Page	B 3.5.1-6	191
	Page	B 3.5.1-7	189
	Page	B 3.5.1-8	191
	Pages	B 3.5.2-1 - B 3.5.2-12	Revised 08/01/01
	Pages	B 3.5.3-1 - B 3.5.3-4	189
	Pages	B 3.5.4-1 - B 3.5.4-7	189
	Pages	B 3.5.5-1 - B 3.5.5-5	189
Bases 3.6	Pages	B 3.6.1-1 - B 3.6.1-4	Revised 03/30/01
	Pages	B 3.6.2-1 - B 3.6.2-8	Revised 03/30/01
	Pages	B 3.6.3-1 - B 3.6.3-11	Revised 03/30/01
	Pages	B 3.6.4-1 - B 3.6.4-3	Revised 04/27/01
	Pages	B 3.6.5-1 - B 3.6.5-3	Revised 04/27/01
	Pages	B 3.6.6-1 - B 3.6.6-12	Revised 08/01/01
	Pages	B 3.6.7-1 - B 3.6.7-6	189 - Revised 11/09/00
Bases 3.7	Pages	B 3.7.1-1 - B 3.7.1-4	189 - Revised 02/12/01
	Pages	B 3.7.2-1 - B 3.7.2-6	189 - Revised 08/09/00
	Pages	B 3.7.3-1 - B 3.7.3-5	189 - Revised 08/09/00
	Pages	B 3.7.4-1 - B 3.7.4-4	189 - Revised 08/09/00
	Pages	B 3.7.5-1 - B 3.7.5-9	Revised 08/01/01
	Pages	B 3.7.6-1 - B 3.7.6-4	189 - Revised 08/09/00
	Pages	B 3.7.7-1 - B 3.7.7-10	Revised 08/01/01
	Pages	B 3.7.8-1 - B 3.7.8-8	Revised 08/01/01
	Pages	B 3.7.9-1 - B 3.7.9-3	Revised 07/16/01
	Pages	B 3.7.10-1 - B 3.7.10-7	Revised 08/01/01
	Pages	B 3.7.11-1 - B 3.7.11-5	189
	Pages	B 3.7.12-1 - B 3.7.12-7	189
	Pages	B 3.7.13-1 - B 3.7.13-3	189 - Revised 08/09/00
	Pages	B 3.7.14-1 - B 3.7.14-3	189
	Pages	B 3.7.15-1 - B 3.7.15-2	189
	Pages	B 3.7.16-1 - B 3.7.16-3	189
	Pages	B 3.7.17-1 - B 3.7.17-3	189
Bases 3.8	Pages	B 3.8.1-1 - B 3.8.1-24	189 - Revised 08/09/00
	Pages	B 3.8.2-1 - B 3.8.2-4	Revised 11/06/01
	Pages	B 3.8.3-1 - B 3.8.3-7	189
	Pages	B 3.8.4-1 - B 3.8.4-9	189 - Revised 08/09/00
	Pages	B 3.8.5-1 - B 3.8.5-3	Revised 11/06/01
	Pages	B 3.8.6-1 - B 3.8.6-6	189 - Revised 08/09/00
	Pages	B 3.8.7-1 - B 3.8.7-3	189
	Pages	B 3.8.8-1 - B 3.8.8-3	Revised 11/06/01
	Pages	B 3.8.9-1 - B 3.8.9-7	Revised 11/06/01
	Pages	B 3.8.10-1 - B 3.8.10-3	Revised 11/06/01
Bases 3.9	Pages	B 3.9.1-1 - B 3.9.1-4	189 - Revised 08/09/00
	Pages	B 3.9.2-1 - B 3.9.2-3	189 - Revised 02/12/01
	Pages	B 3.9.3-1 - B 3.9.3-6	189 - Revised 08/09/00
	Pages	B 3.9.4-1 - B 3.9.4-4	189 - Revised 02/12/01
	Pages	B 3.9.5-1 - B 3.9.5-4	189 - Revised 02/12/01
	Pages	B 3.9.6-1 - B 3.9.6-3	189 - Revised 02/27/01

PALISADES PLANT

FACILITY OPERATING LICENSE DPR-20

APPENDIX A

TECHNICAL SPECIFICATIONS

BASES

As Amended Through Amendment No. 205

Revised 11/06/2001

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

The Palisades Nuclear Plant design criteria (Ref. 1) requires, and these SLs ensure, that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and Anticipated Operational Occurrences (AOOs). 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 fuel centerline temperature stays below the melting temperature.

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 primary 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. 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.

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 primary coolant.

Operation above the boundary of the nucleate boiling regime beyond onset of DNB could result in excessive cladding temperature because of the resultant sharp reduction in the heat transfer coefficient in the transition and film boiling regimes. If a steam film is allowed to form, 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 to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the primary coolant.

BASES

BACKGROUND (continued)

The Reactor Protective System (RPS), in combination with the LCOs, is designed to prevent any anticipated combination of transient conditions for Primary Coolant System (PCS) temperature, pressure, and THERMAL POWER level that would result in a violation of the reactor core SLs.

APPLICABLE SAFETY ANALYSES

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

a. There must be at least a 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 centerline fuel melting.

Palisades uses three DNB correlations; the XNB, ANFP, and HTP detailed in References 3 through 8. The DNB correlations are used solely as analytical tools to ensure that plant conditions will not degrade to the point where DNB could be challenged. The XNB correlation is used for non-High Thermal Performance (HTP) assemblies (assemblies loaded prior to cycle 9), when the non-HTP assemblies could have been limiting. The non-HTP fuel assemblies are used for vessel fluence reduction and reside on the core periphery. The core periphery locations operate at relatively low relative power fractions; therefore, they are not DNB limiting assemblies. The XNB correlation provides administrative justification for using non-HTP assemblies in Palisades low leakage core design. The ANFP and HTP correlations are used for Palisades High Thermal Performance (HTP) fuel assemblies (assemblies loaded in cycle 9 and later).

The HTP correlation can be used when the calculated reactor coolant conditions fall within the correlation's applicable coolant condition ranges. Outside of the applicable range of the HTP correlation, the ANFP correlation can be used. The ANFP correlation may be used over a broader range of coolant conditions than the HTP correlation. The HTP correlation is an extension of the ANFP correlation and incorporates the results of test sections designed to represent HTP fuel design for CE plants.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The prediction of DNB is a function of several measured parameters. The following trip functions and LCOs, limit these measured parameters to protect the Palisades reactor from approaching conditions that could lead to DNB:

Parameter	Protection
Core Flow Rate	Low PCS Flow Trip
Core Power	Variable High Power Trip
PCS Pressure/Core Power	TM/LP Trip
Core Inlet Temperature	T_{inlet} LCO
Axial Shape Index (ASI)	ASI LCO
Assembly Power	Incore Power Monitoring (LHR and F_R^T LCOs)

The RPS setpoints, LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation," in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for PCS temperature, pressure, and THERMAL POWER level that would result in a Departure from Nucleate Boiling Ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

The SL represents a design requirement for establishing the protection system trip setpoints identified previously. LCO 3.2.1, "Linear Heat Rate (LHR)," and LCO 3.2.2, "TOTAL RADIAL PEAKING FACTOR (F_R^T)," or the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

SL 2.1.1.1 and SL 2.1.1.2 ensure that the minimum DNBR is not less than the safety analyses limit and that fuel centerline temperature remains below melting.

The minimum value of the DNBR during normal operation and design basis AOOs is limited to the following DNB correlation safety limit:

Correlation	Safety Limit
XNB	1.17
ANFP	1.154
HTP	1.141

The fuel centerline melt LHR value assumed in the safety analysis is 21 kw/ft. Operation ≤ 21 kw/ft maintains the dynamically adjusted peak LHR and ensures that fuel centerline melt will not occur during normal operating conditions or design AOOs.

BASES

APPLICABILITY	SL 2.1.1.1 and SL 2.1.1.2 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. The steam generator safety valves or automatic protection actions are available to prevent PCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the plant into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1. In MODES 3, 4, 5, and 6, a reactor core SL is not required, since the reactor is not generating significant THERMAL POWER.
SAFETY LIMIT VIOLATIONS	The following violation responses are applicable to the reactor core SLs. <u>2.2.1</u> If SL 2.1.1.1 or SL 2.1.1.2 is violated, the requirement to go to MODE 3 places the plant in a MODE in which this SL is not applicable. The allowed Completion Time of 1 hour recognizes the importance of bringing the plant to a MODE where this SL is not applicable and reduces the probability of fuel damage.
REFERENCES	1. FSAR, Section 5.1 2. FSAR, Chapter 14 3. XN-NF-621(A), Rev 1 4. XN-NF-709 5. ANF-1224(A), May 1989 6. ANF-89-192, January 1990 7. XN-NF-82-21, Rev 1 8. EMF-92-153(A) and Supplement 1, March 1994

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Primary Coolant System (PCS) Pressure SL

BASES

BACKGROUND

The SL on PCS pressure protects the integrity of the PCS against overpressurization. In the event of fuel cladding failure, fission products are released into the primary coolant. The PCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on PCS pressure, continued PCS integrity is ensured. According to Palisades Nuclear Plant design criteria (Ref. 1), the Primary Coolant Pressure Boundary (PCPB) design conditions are not to be exceeded during normal operation and Anticipated Operational Occurrences (AOOs). Also, according to Palisades Nuclear Plant design criteria (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the PCPB greater than limited local yielding.

The design pressure of the PCS is 2500 psia. During normal operation and AOOs, the PCS pressure is kept from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2) and by the piping, valve, and fitting limit of 120% of design pressure (Ref. 6). The initial hydrostatic test was conducted at 125% of design pressure (3125 psia) to verify the integrity of the primary coolant system (Ref. 2). Following inception of plant operation PCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the PCS could result in a breach of the PCPB. If this 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. 4).

BASES

APPLICABLE SAFETY ANALYSES

The PCS primary safety valves, the Main Steam Safety Valves (MSSVs), and the High Pressurizer Pressure trip have settings established to ensure that the PCS pressure SL will not be exceeded.

The PCS primary 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. 2). The transient that establishes the required relief capacity, and hence the valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.

The Reactor Protective System (RPS) trip setpoints (LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation"), together with the settings of the MSSVs (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") and the primary safety valves, provide pressure protection for normal operation and AOOs. In particular, the High Pressurizer Pressure Trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). Conservative values for all system parameters, delay times and core moderator coefficient are assumed.

More specifically, for the limiting case, no credit is taken for operation of any other pressure relieving system including the following:

a. Pressurizer Power Operated Relief Valves (PORVs);

b. Turbine Bypass Control System;

c. Atmospheric Steam Dump Valves;

d. Pressurizer Level Control System; or

e. Pressurizer Pressure Control System.

SAFETY LIMITS

The maximum transient pressure allowable in the PCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the PCS piping, valves, and fittings under 120% of design pressure (Ref. 6). The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable PCS pressure is established at 2750 psia.

BASES

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, 5, and 6 because this SL could be approached or exceeded in these MODES due to overpressurization events. In MODE 6 with the reactor vessel head installed and the reactor vessel head closure bolts less than fully tensioned the potential for an over pressurization event still exists. Although overpressurization of the PCS is impossible once the reactor vessel head is removed, the requirements of this SL apply as long as fuel is in the reactor. Once all the fuel has been removed from the reactor, the requirements of SL 2.1.2 no longer apply.

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the PCS pressure SLs.

2.2.2.1

If the PCS 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.

With PCS pressure greater than the value specified in SL 2.1.2 in MODE 1 or 2, the pressure must be reduced to below this value. A pressure greater than the value specified in SL 2.1.2 exceeds 110% of the PCS design pressure and may challenge system integrity.

The allowed Completion Time of 1 hour provides the operator time to complete the necessary actions to reduce PCS pressure by terminating the cause of the pressure increase, removing mass or energy from the PCS, or a combination of these actions, and to establish MODE 3 conditions.

2.2.2.2

If the PCS pressure SL is exceeded in MODE 3, 4, 5 or 6, PCS pressure must be restored to within the SL value within 5 minutes.

Exceeding the PCS pressure SL in MODE 3, 4, 5 or 6 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.

BASES

REFERENCES

1. FSAR, Section 5.1
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000
4. 10 CFR 100
5. FSAR, Section 4.3
6. ASA B31.1-1955, Code for Pressure Piping, 1967

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCO	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 plant 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 plant 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.)

BASES

LCO 3.0.2 (continued)

The second type of Required Action specifies the remedial measures that permit continued operation of the plant 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 Specifications.

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, "PCS 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. The 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 plant 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.

BASES

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 plant 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 plant. 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.

This Specification delineates the time limits for placing the plant 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 plant 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 plant, assuming that only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the Primary 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.

BASES

LCO 3.0.3 (continued)

A plant 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 Specification 3.0.3 allow 37 hours for the plant to be in MODE 5 when a shutdown is required during MODE 1 operation. If the plant 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 29 hours, because the total time for reaching MODE 4 is not reduced from the allowable limit of 31 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.

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 plant 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 plant shutdown, in accordance with LCO 3.0.3, would not provide appropriate remedial measures for the associated condition of the plant. An example of this is in LCO 3.7.14, "Spent Fuel Pool Water Level."

BASES

LCO 3.0.3 (continued)

LCO 3.7.14 has an Applicability of "During movement of irradiated fuel assemblies in the spent fuel pool." Therefore, this LCO can be applicable in any or all MODES. If the LCO and the Required Actions of LCO 3.7.14 are not met while in MODE 1, 2, or 3, there is no safety benefit to be gained by placing the plant in a shutdown condition. The Required Action of LCO 3.7.14 of "Suspend movement of irradiated fuel assemblies in spent fuel pool" is the appropriate Required Action to complete in lieu of the actions of LCO 3.0.3. These exceptions are addressed in the individual Specifications.

LCO 3.0.4

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

a. Plant 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 plant 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 plant 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 plant 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.

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.

BASES

LCO 3.0.4 (continued)

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.

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 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.

BASES

LCO 3.0.5 (continued)

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 plant 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' LCO's Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the plant is maintained in a safe condition in the support system's Required Actions.

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 into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

BASES

LCO 3.0.6 (continued)

Specification 5.5.13, "Safety Functions 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 redundant OPERABLE support system 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.

LCO 3.0.7

Special tests and operations are required at various times over the plant's life to demonstrate performance characteristics, to perform maintenance activities, and to perform special evaluations. Because TS normally preclude these tests and operations, Special Test Exceptions (STEs) allow specified requirements to be changed or suspended under controlled conditions. STEs are included in applicable sections of the Specifications. Unless otherwise specified, all other TS requirements remain unchanged and in effect as applicable. This will ensure that all appropriate requirements of the MODE or other specified condition not directly associated with or required to be changed or suspended to perform the special test or operation will remain in effect.

The Applicability of an STE LCO represents a condition not necessarily in compliance with the normal requirements of the TS. Compliance with STE LCO is optional.

A special test may be performed under either the provisions of the appropriate STE LCO or the other applicable TS requirements. If it is desired to perform the special test under the provisions of the STE LCO, the requirements of the STE LCO shall be followed. This includes the SRs specified in the STE LCO.

BASES

LCO 3.0.7 (continued)

Some of the STE LCO require that one or more of the LCO for normal operation be met (i.e., meeting the STE LCO requires meeting the specified normal LCO). The Applicability, ACTIONS, and SRs of the specified normal LCO, however, are not required to be met in order to meet the STE LCO when it is in effect. This means that, upon failure to meet a specified normal LCO, the associated ACTIONS of the STE LCO apply, in lieu of the ACTIONS of the normal LCO. Exceptions to the above do exist.

There are instances when the Applicability of the specified normal LCO must be met, where its ACTIONS must be taken, where certain of its Surveillances must be performed, or where all of these requirements must be met concurrently with the requirements of the STE LCO.

Unless the SRs of the specified normal LCO are suspended or changed by the special test, those SRs that are necessary to meet the specified normal LCO must be met prior to performing the special test. During the conduct of the special test, those Surveillances need not be performed unless specified by the ACTIONS or SRs of the STE LCO.

ACTIONS for STE LCO provide appropriate remedial measures upon failure to meet the STE LCO. Upon failure to meet these ACTIONS, suspend the performance of the special test and enter the ACTIONS for all LCOs that are then not met. Entry into LCO 3.0.3 may possibly be required, but this determination should not be made by considering only the failure to meet the ACTIONS of the STE LCO.

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 plant 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) are only applicable when the STE is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) 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 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 plant 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.

An example of this process is:

a. High Pressure Safety Injection (HPSI) maintenance during shutdown that requires system functional tests at a specified pressure. Provided other appropriate testing is satisfactorily completed, startup can proceed with HPSI 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 plant 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. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is the Containment Leak Rate Testing Program.

BASES

SR 3.0.2 (continued)

As stated in SR 3.0.2, the 25% extension allowed by SR 3.0.2 may be applied to Required Actions whose Completion Time is stated as "once per . . ." however, the 25% extension does not apply to the initial performance of a Required Action with a periodic Completion Time that requires performance on a "once per . . ." basis. The 25% extension applies to each performance of the Required Action 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 plant 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.

BASES

SR 3.0.3 (continued)

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

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 plant.

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

SR 3.0.4 (continued)

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

The provisions of SR 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS.

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.

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 maintaining the reactor core subcritical when shut down under cold conditions, in accordance with the Palisades Nuclear Plant design criteria (Ref. 1). Maintenance of the SDM ensures that postulated reactivity events will not damage the fuel. SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown events and Anticipated Operational Occurrences (AOOs). As such, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all full-length control rods, assuming that the single control rod of highest reactivity worth remains fully withdrawn. Once all full-length control rods have been verified to be at or below the lower electrical limit, the penalty for the control rod of highest reactivity worth fully withdrawn no longer must be applied.

The Palisades Nuclear Plant design criteria requires that two separate 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 movable control rods and soluble boric acid in the Primary Coolant System (PCS). The Rod Control System provides the SDM during power operation and is capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel design limits, assuming that the control rod of highest reactivity worth remains fully withdrawn.

The soluble boron system can compensate for fuel depletion during operation and all xenon burnout reactivity changes, and maintain the reactor subcritical under cold conditions.

During MODES 1 and 2, SDM control is ensured by operating with the shutdown rods within the limits of LCO 3.1.5, "Shutdown and Part-Length Rod Group Insertion Limits," and the regulating rods within the limits of LCO 3.1.6, "Regulating Rod Group Position Limits." When the plant is in MODES 3, 4, 5, and 6 the SDM requirements are met by means of adjustments to the PCS boron concentration.

BASES

APPLICABLE SAFETY ANALYSES

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 AOOs, with the assumption that the control rod of highest reactivity worth is fully withdrawn following a reactor trip. For MODE 5, the primary safety analysis that relies on the SDM limits is the boron dilution analysis.

The acceptance criteria for the SDM requirements are that specified acceptable fuel design limits are maintained. This is done by ensuring that:

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

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits (Departure from Nucleate Boiling Ratio (DNBR), fuel centerline temperature limit AOOs, and ≤ 280 cal/gm energy deposition for the control rod ejection accident); and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The most limiting accident for the SDM requirements are based on a Main Steam Line Break (MSLB), as described in the accident analysis (Ref. 2). The increased steam flow resulting from a pipe break in the main steam system causes an increased energy removal from the affected Steam Generator (SG), and consequently the PCS. This results in a reduction of the primary coolant temperature. The resultant coolant shrinkage causes a reduction in pressure. In the presence of a negative moderator temperature coefficient, this cooldown causes an increase in core reactivity. The most limiting MSLB with respect to potential fuel damage is a guillotine break of a main steam line initiated at the end of core life. The positive reactivity addition from the moderator temperature decrease will terminate when the affected SG boils dry, thus terminating PCS heat removal and cooldown. Following the MSLB, a post trip return to power may occur; however, THERMAL POWER does not violate the Safety Limit (SL) requirement of SL 2.1.1.

In addition to the limiting MSLB transient, the SDM requirement for MODES 3 and 4 must also protect against an inadvertent boron dilution; (Ref. 3) and an uncontrolled control rod bank withdrawal from subcritical conditions (Ref. 5).

BASES

APPLICABLE SAFETY ANALYSES (continued)

Each of these events is discussed below.

In the boron dilution analysis, the required SDM defines the reactivity difference between an initial subcritical boron concentration and the corresponding critical boron concentration. These values, in conjunction with the configuration of the PCS and the assumed dilution flow rate, directly affect the results of the analysis. This event is most limiting at the beginning of core life when critical boron concentrations are highest.

The withdrawal of a control rod bank from subcritical conditions adds reactivity to the reactor core, causing both the core power level and heat flux to increase with corresponding increases in reactor coolant temperatures and pressure. The withdrawal of control rod banks also produce a time dependent redistribution of core power.

Depending on the system initial conditions and reactivity insertion rate, the uncontrolled control rod banks withdrawal transient is terminated by either a high power trip or a high pressurizer pressure trip. In all cases, power level, PCS pressure, linear heat rate, and the DNBR do not exceed allowable limits.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2).

LCO

The MSLB (Ref. 2) and the boron dilution (Ref. 3) accidents are the most limiting analyses that establish the value for SDM. For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). For the boron dilution accident, if the LCO is violated, then the minimum required time assumed for operator action to terminate dilution may no longer be applicable.

SDM is a core physics design condition that can be ensured through full-length control rod positioning (regulating and shutdown rods) and through the soluble boron concentration.

APPLICABILITY

In MODE 3, 4 and 5, the SDM requirements are applicable to provide sufficient negative reactivity to meet the assumptions of the safety analyses discussed above. In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5, and LCO 3.1.6. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

BASES

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 concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the PCS as soon as possible, the boron injection flow should be a highly concentrated solution, such as that normally found in the concentrated boric acid storage tank. The operator should borate with the best source available for the plant conditions.

In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the PCS boron concentration is at the beginning of cycle, when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of 1% $\Delta\rho$ must be recovered and a boration flow rate of 35 gpm, it is possible to increase the boron concentration of the PCS by 100 ppm in approximately 25 minutes. If a boron worth of 1.0 E-4 $\Delta\rho$/ppm is assumed, this combination of parameters will increase the SDM by 1% $\Delta\rho$. These boration parameters of 35 gpm and 100 ppm represent typical values and are provided for the purpose of offering a specific example.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

SDM is verified by a reactivity balance calculation, considering the listed reactivity effects:

a. PCS boron concentration;

b. Control rod positions;

c. PCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration; and

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

f. Isothermal Temperature Coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical and the fuel temperature will be changing at the same rate as the PCS.

Samarium is not considered in the reactivity analysis since the analysis assumes that the negative reactivity due to Samarium is offset by the positive reactivity of Plutonium built in.

SR 3.1.1.1 requires SDM to be within the limits specified in the COLR. This SDM value ensures the consequences of an MSLB, will be acceptable as a result of a cooldown of the PCS which adds positive reactivity in the presence of a negative moderator temperature coefficient as well as the other events described in the Applicable Safety Analysis. As such, the requirements of this SR must be met whenever the plant is in MODES 3, 4, and 5.

The Frequency of 24 hours for the verification of SDM 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 completing the calculation.

REFERENCES

1. FSAR, Section 5.1
2. FSAR, Section 14.14
3. FSAR, Section 14.3
4. 10 CFR 100
5. FSAR, Section 14.2

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Balance

BASES

BACKGROUND

According to the Palisades Nuclear Plant design criteria (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 anticipated operational occurrences. Therefore, reactivity balance is used as a measure of the predicted versus measured core reactivity during power operation. The periodic confirmation of core reactivity is necessary to ensure that Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, control rod worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SHUTDOWN MARGIN (SDM) or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in ensuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. 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 burnable absorbers. Excess reactivity can be inferred from the critical boron curve, which provides an indication of the soluble boron concentration in the Primary Coolant System (PCS) versus cycle burnup. Periodic measurement of the PCS boron concentration for comparison with the predicted value with other variables fixed (such as control rod height, temperature, pressure, and power) provides a convenient method of ensuring that core reactivity is within design expectations, and that the calculational models used to generate the safety analysis are adequate.

BASES

BACKGROUND (continued)

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and in 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 at RTP and moderator temperature, the excess positive reactivity is compensated by burnable poisons, full-length control rods, neutron poisons (mainly xenon and samarium) in the fuel, and the PCS boron concentration.

When the core is producing THERMAL POWER, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the PCS boron concentration is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The critical boron curve is based on steady state operation at RTP. Therefore, deviations from the predicted critical boron curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated.

APPLICABLE SAFETY ANALYSES

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation (Ref. 2) is, therefore, dependent upon accurate evaluation of core reactivity. In particular, SDM and reactivity transients, such as control rod withdrawal accidents or control rod ejection accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Monitoring reactivity balance additionally 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 PCS boron concentration requirements for reactivity control during fuel depletion.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The comparison between measured and predicted initial core reactivity provides a normalization for calculational models used to predict core reactivity. If the measured and predicted PCS boron concentrations for identical core conditions at Beginning Of Cycle (BOC) are not within design tolerances, then the assumptions used in the reload cycle design analysis or the calculational models used to predict soluble boron requirements may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the measured boron concentration from the predicted critical boron curve that develop during fuel depletion may be an indication that the calculational model is not adequate for core burnups beyond BOC, or that an unexpected change in core conditions has occurred.

The normalization of predicted PCS boron concentration to the measured value is typically performed after reaching RTP following startup from a refueling outage, with the control rods in their normal positions for power operation. 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.

The reactivity balance satisfies Criterion 2 of 10 CFR 50.36(c)(2).

LCO

The reactivity balance limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between actual and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the nuclear design methodology are larger than expected. A limit on the reactivity balance of $\pm$ 1% $\Delta\rho$ has been established, based on engineering judgment. A 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

BASES

LCO (continued)

When measured core reactivity is within ± 1% $\Delta\rho$ of the predicted value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. Since deviations from the limits are normally detected by comparing predicted and measured steady state PCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the PCS boron concentration is unlikely.

APPLICABILITY

The limits on core reactivity must be maintained during MODES 1 and 2 because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER. 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 MODE 2 because enough operating margin exists to limit the effects of a reactivity anomaly, and THERMAL POWER is low enough (≤ 5% RTP) such that reactivity anomalies are unlikely to occur. This Specification does not apply in MODES 3, 4, and 5 because the reactor is shut down and the reactivity balance is not changing.

In MODE 6, fuel loading results in a continually changing core reactivity. Boron concentration requirements (LCO 3.9.1, "Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis.

ACTIONS

A.1 and A.2

Should an imbalance develop between measured and predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input to 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.

BASES

ACTIONS

A.1 and A.2 (continued)

The required Completion Time of 7 days is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor 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 imbalance may be resolved. If the cause of the reactivity imbalance is a mismatch in core conditions at the time of PCS boron concentration sampling, then a recalculation of the PCS boron concentration requirements 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 imbalance 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 critical boron curve may be renormalized, and power operation may continue. If operational restrictions or additional SRs 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 whatever operating restrictions or Surveillances that may be required to allow continued reactor operation.

B.1

If the Required Actions for Condition A are not met within 7 days, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by periodic comparisons of measured and predicted PCS boron concentrations. The comparison is made considering that other core conditions are fixed or stable including control rod position, moderator temperature, fuel temperature, fuel depletion, and xenon concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. The SR is modified by a Note in the Surveillance column which indicates that if the normalization of predicted core reactivity to the measured value is to occur, it must take place within the first 60 Effective Full Power Days (EFPD) after each refueling. This 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. The required subsequent Frequency of 31 EFPD following the initial 60 EFPD after entering MODE 1, is acceptable, based on the slow rate of core changes due to fuel depletion and the presence of other indicators (e.g., T_q, etc.) for prompt indication of an imbalance. A second Note, "only required after initial 60 EFPD," is added to the Frequency column to allow this.

REFERENCES

1. FSAR, Section 5.1
2. FSAR, Chapter 14

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to Palisades Nuclear Plant design criteria (Ref. 1), the reactor core and its interaction with the Primary Coolant System (PCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended or rapid reactivity increases.

The MTC relates a change in core reactivity to a change in primary 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. The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, 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.

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by the measurement performed as part of startup testing following a refueling. Both initial and reload cores are designed so that the Beginning Of Cycle (BOC) MTC is less positive than that allowed by the LCO. The actual value of the MTC is dependent on core characteristics, such as fuel loading and primary coolant soluble boron concentration. The core design may require additional fixed distributed poisons (lumped burnable poison assemblies) 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. However, the safety analysis assumptions for the MTC at EOC are assumed by confirming the BOC MTC measurement is within limits which indicates the core is behaving as predicted.

BASES

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis (Ref. 2); and

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

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 important to safety, and both values must be bounded. Values used in the analyses consider worst case conditions, such as very large soluble boron concentrations, to ensure the accident results are bounding (Ref. 3).

Accidents that cause core overheating, either by decreased heat removal or increased power production, must be evaluated for results when the MTC is positive. Examples of reactivity accidents that cause increased power production include the control rod bank withdrawal transient from either partial or RATED 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. Several events discussed in Reference 2 are analyzed with a positive MTC.

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 PCS, and is therefore the most limiting event with respect to the negative MTC, is a Main Steam Line Break (MSLB) event. Following the reactor trip for the postulated EOC MSLB 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 (approximately 12% RTP) is produced.

The MTC satisfies Criterion 2 of 10 CFR 50.36(c)(2).

BASES

LCO

LCO 3.1.3 requires the MTC to be < 0.5 E-4 $\Delta\rho$/°F at ≤ 2% RTP 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 limit on a positive MTC ensures that core overheating accidents will not violate the accident analysis assumptions.

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

APPLICABILITY

In MODE 1, the MTC must be maintained to ensure that any accident initiated from THERMAL POWER operation will not violate the design assumptions of the accident analysis. In MODE 2, the limits must also be maintained to ensure startup and subcritical accidents, such as the uncontrolled full-length control rod or group withdrawal, will not violate the assumptions of the accident analysis. The measurement of MTC in MODE 2 prior to exceeding 2% RTP is used to confirm that the core is behaving as analyzed. This ensures that the MTC will remain within the analyzed range while operating in MODES 1 and 2. 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 the 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 the MTC, with temperature assumed in the safety analysis, is accepted as valid once the BOC measurement is used for normalization.

BASES

ACTIONS

A.1

MTC is a function of 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 bounds. 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, and the time for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant 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 smoothly from most positive (or least negative) to most negative value during fuel cycle operation as the PCS boron concentration is reduced to compensate for fuel depletion. The requirement for measurement prior to operation > 2% RTP satisfies the confirmatory check on the most positive (or least negative) MTC value. It also confirms that the core is behaving as analyzed which ensures that the MTC will remain within the analysis limits for the remainder of the fuel cycle.

REFERENCES

1. FSAR, Section 5.1
2. FSAR, Chapter 14
3. FSAR, Section 3.3

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Control Rod Alignment

BASES

BACKGROUND

The OPERABILITY (e.g., trippability) of the shutdown and regulating rods is an initial assumption in all safety analyses that assume full-length control rod insertion upon reactor trip. Maximum control rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The Palisades Nuclear Plant design criteria contain the applicable criteria for these reactivity and power distribution design requirements (Ref. 1).

Mechanical or electrical failures may cause a control rod to become inoperable or to become misaligned from its group. Control rod 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 in 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 at a fixed rate of approximately 46 inches per minute. Although the ability to move a full-length control rod by its drive mechanism is not an initial assumption used in the safety analyses, it is required to support OPERABILITY. As such, the inability to move a full-length control rod results in that full-length control rod being inoperable.

The control rods are arranged into groups that are radially symmetric. Therefore, movement of the control rod groups does not introduce radial asymmetries in the core power distribution. The shutdown and regulating rods provide the required reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating rods also provide reactivity (power level) control during normal operation and transients.

BASES

BACKGROUND (continued)

The axial position of shutdown and regulating rods is indicated by two separate and independent systems, which are 1) synchro based position indication system, and 2) the reed switch based position indication system.

The synchro based position indication system measures the phase angle of a synchro geared to the CRDM rack. Full control rod travel corresponds to less than 1 turn of the synchro. Each control rod has its own synchro. The Primary Information Processor (PIP) node scans and converts synchro outputs into inches of control rod withdrawal. The resolution of this system is approximately 0.5 inches. Each synchro also has cam operated limit switches which can provide positive indication of control rod position.

The reed switch based position indication system is referred to as the Secondary Position Indication (SPI) system. This system provides a highly accurate indication of actual control rod position, but at a lower precision than the synchros. The reed switches are wired so that the voltage read across the reed switch stack is proportional to rod position. The reed switches are spaced along a tube with a center to center spacing distance of 1.5 inches. The resolution of the SPI reed switch stacks is 1.5 inches. The reed switches also provide input to the matrix indication lights which provide control rod status indication for various key positions. To increase the reliability of the system, there are redundant reed switches which prevent false indication in the event an individual reed switch fails.

A control rod position deviation alarm is provided to alert the operator when any two control rods in the same group are more than 8 inches apart. This helps to ensure any control rod misalignments are minimized. The alarm can be generated by either the SPI system or PIP node since the SPI system, in conjunction with the host computer, is redundant to the PIP node in the task of control rod measurements, control rod monitoring, and limit processing.

BASES

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Refs. 3 and 4). The accident analysis defines control rod misoperation as any event, with the exception of sequential group withdrawals, which could result from a single malfunction in the reactivity control systems. For example, control rod misalignment may be caused by a malfunction of the Rod Control System, or by operator error. A stuck rod may be caused by mechanical jamming. Inadvertent withdrawal of a single control rod may be caused by an electrical or mechanical failure in the Rod Control System. A dropped control rod could be caused by an electrical or mechanical failure in the CRDM.

The acceptance criteria for addressing control rod inoperability/misalignment are that:

a. There shall be no violations of:

   1. Specified Acceptable Fuel Design Limits (SAFDL), or

   2. Primary Coolant System (PCS) pressure boundary integrity; and

b. The core must remain subcritical after accident transients.

Three types of misoperations are discussed in the safety analysis (Ref. 4). During movement of a group, one control rod may stop moving while the other control rods in the group continue. This condition may cause excessive power peaking. The second type of misoperations occurs if one control rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the remaining control rods to meet the SDM requirement with the maximum worth rod stuck fully withdrawn. If a control rod is stuck in the fully withdrawn position, its worth is added to the SDM requirement, since the safety analysis does not take two stuck rods into account. The third type of misoperations occurs when one 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).

BASES

APPLICABLE SAFETY ANALYSES (continued)

The most limiting static misalignment occurs when Bank 4 is fully inserted with one rod fully withdrawn (Bank 4 is 99 inches out of alignment with the rated Power Dependent Insertion Limit (PDIL). This event was bounded by the dropped full-length control rod event (Ref. 4).

Since the control rod drop incidents result in the most rapid approach to SAFDLs caused by a control rod misoperation, the accident analysis analyzed a single full-length control rod drop.

The above control rod misoperations may or may not result in an automatic reactor trip. In the case of the full-length rod drop, a prompt decrease in core average power and a distortion in radial power are initially produced, which, when conservatively coupled, result in a local power and heat flux increase, and a decrease in DNBR parameters.

The results of the control rod misoperation analysis show that during the most limiting misoperation events, no violations of the SAFDLs, fuel centerline temperature, or PCS pressure occur.

Control rod alignment satisfies Criteria 2 and 3 of 10 CFR 50.36(c)(2).

LCO

The limits on shutdown, regulating, and part-length rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on OPERABILITY ensure that upon reactor trip, the full-length control rods will be available and will be inserted to provide enough negative reactivity to shut down the reactor. The OPERABILITY requirements also ensure that the control rod banks maintain the correct alignment and that each full-length control rod is capable of being moved by its CRDM. The OPERABILITY requirement for the part-length rods is that they are fully withdrawn.

The requirement is to maintain the control rod alignment to within 8 inches between any control rod and all other rods in its group. To help ensure this requirement is met, the control rod position deviation alarm generated by either the PIP node or the SPI system, must be OPERABLE and provide an alarm when any control rod becomes misaligned > 8 inches from any other rod in its group. The safety analysis assumes a total misalignment from fully withdrawn to fully inserted. This case bounds the safety analysis for a single rod in any intermediate position.

BASES

LCO (continued)

The primary rod position indication system is considered OPERABLE, for purposes of this specification, if the digital position readout, the PPC display, or the cam operated position indication lights give positive indication of rod position. The secondary rod position indication system is considered OPERABLE if the magnetically operated reed switches are providing positive indication of rod position either via the plant process computer or taking direct readings of the output from the magnetic reed switches.

Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDM, any 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 neutron (or fission) power is generated, and the OPERABILITY (e.g., trippability) and alignment of control 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 not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and regulating 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 PCS. 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 refueling.

ACTIONS

LCOs 3.1.4, 3.1.5, and 3.1.6, and their ACTIONS were written to support each other. The combined intent is to assure the following:

1. There is adequate SDM available in withdrawn control rods to assure the reactor is shutdown by, and remains shutdown following, a reactor trip,

2. The control rod positioning does not cause unacceptable axial or radial flux peaking, and

3. The programmed rod withdrawal sequence and group overlap result in reactivity insertion rates within the assumptions of the Inadvertent Control Rod Bank Withdrawal Analyses.

BASES

ACTIONS (continued)

The ACTIONS for rods that are mispositioned (misaligned or inserted beyond the limit) were written assuming that an OPERABLE rod discovered to be mispositioned would simply be re-positioned correctly. While the associated Conditions would have to be entered, the rod could be re-positioned (thus exiting the LCO) without taking any other Required Action. A rod which remains mispositioned was assumed to be inoperable. The analyses account for operation with one (and only one) mispositioned rod (a dropped rod being the limiting case). With more than one mispositioned rod, the plant would be outside the bounds of the analyses and must be shutdown.

If a rod is discovered to be misaligned (ie, there is more than 8" between it and any other rod in its group, but all remaining rods in that group are within 8 inches of each other) Condition 3.1.4 C allows 2 hours to restore the rod alignment (thus exiting the LCO), perform SR 3.2.2.1 (verification that radial peaking is within limits), or reduce power to ≤ 75% RTP.

If one or more shutdown rods are inserted beyond the insertion limit, 3.1.5 A is entered; the rods are declared inoperable and 3.1.4 D (when one rod is immovable but trippable) or 3.1.4 E (when a movable rod is inserted beyond its insertion limit, or when more than one rod is inoperable for any reason) must be entered.

If the rods can be moved, they should be withdrawn and all Conditions exited.

If one rod cannot be moved (but is still considered trippable), operation may continue in accordance with 3.1.4 D (and 3.1.4 C if it is misaligned).

If more than one rod cannot be moved, Condition 3.1.4 E must also be entered. The plant must be in MODE 3 in 6 hours in accordance with ACTION 3.1.4 E.1.

If one or more part length rods are inserted beyond the limit, 3.1.5 A is entered; the rods are declared inoperable and 3.1.4 E is entered (and 3.1.4 C if it is misaligned). Condition 3.1.4 D is not applicable to Part Length rods since it only addresses full length rods.

If the rods can be moved, they should be withdrawn and all Conditions exited.

BASES

ACTIONS (continued)

If any part length rods are inserted beyond the limit and cannot be moved, the plant must be placed in MODE 3 in 6 hours in accordance with ACTION 3.1.4 E.1.

If one or more regulating rods are inserted beyond the limit, 3.1.6 A is entered;

The rods must be restored to within limits (by rod withdrawal or power reduction) within two hours.

If a rod cannot be moved, it must be considered inoperable and Condition 3.1.4 D must be entered (and 3.1.4 C if it is misaligned). Condition 3.1.4 D allows continued operation with one inoperable, but trippable, rod until the next reactor shutdown (MODE 3 entry). If more than one rod cannot be moved, Condition 3.1.4 E must be entered. The plant must be in MODE 3 in 6 hours in accordance with ACTION 3.1.4 E.1.

The analyses do not account for the possibility of more then one rod failing to insert on a trip. While boron concentration might be adjusted to restore SHUTDOWN MARGIN, if two adjacent rods fail to insert that portion of the core could remain excessively reactive. Since the analyses must assume that one rod fails to insert, operation may not continue with a known untrippable rod. A shutdown would be required by Condition 3.1.4 E.

A.1

Rod position indication is required to allow verification that the rods are positioned and aligned as assumed in the safety analysis. If one rod position indication channel is inoperable for one or more control rods then SR 3.1.4.1 (rod position verification) is required to be performed once within 15 minutes following any rod motion in that group. This ensures that the rods are positioned as required.

BASES

ACTIONS (continued)

B.1

When the control rod deviation alarm is inoperable, performing SR 3.1.4.1, once within 15 minutes of movement of any control rod, ensures improper control rod alignments are identified before unacceptable flux distributions occur. The specified Completion Times take into account other information continuously available to the operator in the control room, so that during control rod movement, deviations can be detected, and the protection provided by the control rod and deviation circuit is not required.

C.1 and C.2

Condition C addresses the situation where one rod in a group is misaligned, ie. there is more than 8" between that rod and any other rod in its group, but all remaining rods in that group are within 8 inches of each other.

A full-length control rod may become misaligned, yet remain trippable. In this condition, the control rod can still perform its required function of adding negative reactivity should a reactor trip be necessary.

Regulating rod alignment can be restored by either aligning the misaligned rod(s) to within 8 inches of all other rods in its group or, aligning the misaligned rod's group to within 8 inches of the misaligned rod if allowed by the rod group insertion limits. Shutdown rod alignment can be restored by aligning the misaligned rod to within 8 inches of all other rods in its group.

If one control rod is misaligned by > 8 inches continued operation in MODES 1 and 2 may continue, provided, within 2 hours, the TOTAL RADIAL PEAKING FACTOR has been verified acceptable in accordance with SR 3.2.2.1, or the power is reduced to ≤ 75% RTP.

Xenon redistribution in the core starts to occur as soon as a rod becomes misaligned. Reducing THERMAL POWER to ≤ 75% RTP ensures acceptable power distributions are maintained.

BASES

ACTIONS

C.1 and C.2 (continued)

For small misalignments of the control rods, there is:

a. A small effect on the time dependent long term power distributions relative to those used in generating LCOs and Limiting Safety System Settings (LSSS) setpoints;

b. A negligible effect on the available SDM; and

c. A small effect on the ejected rod worth used in the accident analysis.

With a large control rod misalignment, however, this misalignment would cause distortion of the core power distribution. This distortion may, in turn, have a significant effect on the time dependent, long term power distributions relative to those used in generating LCOs and LSSS setpoints.

The effect on the available SDM and the ejected rod worth used in the accident analysis remains small.

In both cases, a 2 hour time period is sufficient to:

a. Identify cause of a misaligned rod;

b. Take appropriate corrective action to realign the rods; and

c. Minimize the effects of xenon redistribution.

The Palisades analysis for rod misalignment is bounded by a single dropped rod. Therefore, rod misalignments are limited to one rod being misaligned from its group. If a full-length control rod is untrippable, it is not available for reactivity insertion during a reactor trip. With an untrippable full-length control rod, meeting the insertion limits of LCO 3.1.5, "Shutdown and Part-Length Rod Group Insertion Limits," and LCO 3.1.6, "Regulating Rod Group Position Limits," does not ensure that adequate SDM exists and therefore, the Actions of Condition E must be met.

BASES

ACTIONS (continued)

D.1

Condition D is entered whenever it is discovered that a single full-length control rod cannot be moved by its operator yet the control rod is still capable of being tripped (or is fully inserted). Although the ability to move a full-length control rod is not an initial assumption used in the safety analyses, it does relate to full-length control rod OPERABILITY. The inability to move a full-length control rod by its operator may be indicative of a systemic failure (other than trippability) which could potentially affect other rods. Thus, declaring a full-length control rod inoperable in this instance is conservative since it limits the number of full-length control rods which cannot be moved by their operators to only one. The Completion Time to restore an inoperable control rod to OPERABLE status is stated as prior to entering MODE 2 following next MODE 3 entry. This Completion Time allows unrestricted operation in MODES 1 and 2 while conservatively preventing a reactor startup with an immovable full-length control rod.

E.1

If the Required Action or associated Completion Time of Condition A, Condition B, Condition C, or Condition D is not met; one or more control rods are inoperable for reasons other than Condition D (ie, one full length control rod is inoperable for reasons other than being "immovable but trippable," or more than one control rod, whether full length or part length, are inoperable for any reasons) ; or two or more control rods are misaligned by > 8 inches, or two channels of control rod position indication are inoperable for one or more control rods, the plant is required to be brought to MODE 3. By being brought to MODE 3, the plant is brought outside its MODE of applicability. Continued operation is not allowed in the case of more than one control rod misaligned from any other rod in its group by > 8 inches, or two or more rods inoperable. This is because these cases may be indicative of a loss of SDM and power re-distribution, and a loss of safety function, respectively.

Also, if no rod position indication exists for one or more control rods, continued operation is not allowed because the safety analysis assumptions of rod position cannot be ensured.

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual control rod positions are within 8 inches of all other control rods in the group at a 12 hour Frequency allows the operator to detect a control 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 control rod movement, deviations can be detected. Also protection can be provided by the control rod deviation alarm.

SR 3.1.4.2

OPERABILITY of two control rod position indicator channels is required to determine control rod positions, and thereby ensure compliance with the control rod alignment and insertion limits. Performance of a CHANNEL CHECK on the primary and secondary control rod position indication channels provides confidence in the accuracy of the rod position indication systems. The control rod "full in" and "full out" lights, which correspond to the lower electrical limit and the upper electrical limit respectively, provide an additional means for determining the control rod positions when the control rods are at either their fully inserted or fully withdrawn positions.

The 12 hour Frequency takes into consideration other information continuously available to the operator in the control room, so that during control rod movement, deviations can be detected, and protection can be provided by the control rod deviation alarm.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.3

Verifying each full-length control rod is trippable would require that each full-length control rod be tripped. In MODES 1 and 2, tripping each full-length control rod would result in radial or axial power tilts, or oscillations. Therefore, individual full-length control rods are exercised every 92 days to provide increased confidence that all full-length control rods continue to be trippable, even if they are not regularly tripped. A movement of 6 inches is adequate to demonstrate motion without exceeding the alignment limit when only one control rod is being moved. The 92 day Frequency takes into consideration other information available to the operator in the control room and other surveillances being performed more frequently, which add to the determination of OPERABILITY of the control rods. At any time, if a control rod(s) is inoperable, a determination of the trippability of the control rod(s) must be made, and appropriate action taken.

SR 3.1.4.4

Demonstrating the rod position deviation alarm is OPERABLE verifies the alarm is functional. The 18 month Frequency takes into account other information continuously available to the operator in the control room, so that during control rod movement, deviations can be detected.

SR 3.1.4.5

Performance of a CHANNEL CALIBRATION of each control rod position indication channel ensures the channel is OPERABLE and capable of indicating control rod position over the entire length of the control rod's travel with the exception of the secondary rod position indicating channel dead band near the bottom of travel. This dead band exists because the control rod drive mechanism housing seismic support prevents operation of the reed switches. Since this Surveillance must be performed when the reactor is shut down, an 18 month Frequency to be coincident with refueling outage was selected. Operating experience has shown that these components usually pass this Surveillance when performed at a Frequency of once every 18 months. Furthermore, the Frequency takes into account other surveillances being performed at shorter Frequencies, which determine the OPERABILITY of the control rod position indicating systems.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.6

Verification of full-length control rod drop times determines that the maximum control rod drop time is consistent with the assumed drop time used in that safety analysis (Ref. 2). The 2.5 second acceptance criteria is measured from the time the CRDM clutch is deenergized by the reactor protection system or test switch to 90% insertion. This time is bounded by that assumed in the safety analysis (Ref.2). Measuring drop times prior to reactor criticality, after reactor vessel head reinstallation, ensures that reactor internals and CRDMs will not interfere with full-length control rod motion or drop time and that no degradation in these systems has occurred that would adversely affect full-length control rod motion or drop time. Individual full-length control rods whose drop times are greater than safety analysis assumptions are not OPERABLE. This SR is performed prior to criticality, based on the need to perform this Surveillance under the conditions that apply during a plant outage and because of the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. FSAR, Section 5.1
2. FSAR, Section 14.1
3. FSAR, Section 14.4
4. FSAR, Section 14.6

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown and Part-Length Rod Group Insertion Limits

BASES

BACKGROUND

The insertion limits of the shutdown rods are initial assumptions in all safety analyses that assume full-length 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 Palisades Nuclear Plant design criteria (Ref. 1) and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," contain the applicable criteria for these reactivity and power distribution design requirements. Limits on shutdown rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the reactivity limits, ejected rod worth, and SDM limits are preserved.

The shutdown rods are arranged into groups that are radially symmetric. Therefore, movement of the shutdown rod groups does not introduce radial asymmetries in the core power distribution. The shutdown and regulating rod groups provide the required reactivity worth for immediate reactor shutdown upon a reactor trip.

The Palisades Nuclear Plant has four part-length control rods installed. The part-length rods are required to remain completely withdrawn during power operation except during rod exercising performed in conjunction with SR 3.1.4.3. The part-length rods do not insert on a reactor trip.

The design calculations are performed with the assumption that the shutdown rod groups are withdrawn prior to the regulating rod groups. The shutdown rods can be fully withdrawn without the core going critical. This provides available negative reactivity for SDM in the event of boration errors. All control rod groups are controlled manually by the control room operator. During normal plant operation, the shutdown rod groups are fully withdrawn. The shutdown rod groups must be completely withdrawn from the core prior to withdrawing any regulating rods during an approach to criticality. The shutdown rod groups are then left in this position until the reactor is shut down.

They affect core power, burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

BASES

APPLICABLE SAFETY ANALYSES

Accident analysis assumes that the shutdown rod groups are fully withdrawn any time the reactor is critical. This ensures that:

a. The minimum SDM is maintained; and

b. The potential effects of a control rod ejection accident are limited to acceptable limits.

Control rods are considered fully withdrawn at 128 inches, since this position places them in an insignificant reactivity worth region of the integral worth curve for each bank.

On a reactor trip, all full-length control rods (shutdown and regulating), except the most reactive rod, are assumed to insert into the core. The shutdown and regulating rod groups shall be at or above their insertion limits and available to insert the required amount of negative reactivity on a reactor trip signal. The regulating rods may be partially inserted in the core as allowed by LCO 3.1.6, "Regulating Rod Group Position Limits." The shutdown rod group insertion limit is established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)) following a reactor trip from full power. The combination of regulating rod and shutdown rods (less the most reactive rod, which is assumed to remain fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 2). The shutdown rod group insertion limit also limits the reactivity worth of an ejected shutdown rod.

The acceptance criteria for addressing shutdown rods as well as regulating rod insertion limits and inoperability)or misalignment are that:

a. There be no violation of:

   1. Specified acceptable fuel design limits, or

   2. Primary Coolant System pressure boundary damage; and

b. The core remains subcritical after accident transients.

BASES

APPLICABLE SAFETY ANALYSES (continued)

As such, the shutdown and part-length rod group insertion limits affect safety analyses involving core reactivity, ejected rod worth, and SDM (Ref. 2). The part-length control rods have the potential to cause power distribution envelopes to be exceeded if inserted while the reactor is critical. Therefore, they must remain withdrawn in accordance with the limits of the LCO (Ref. 3).

The shutdown and part-length rod group insertion limits satisfy Criterion 2 of 10 CFR 50.36(c)(2).

LCO

The shutdown and part-length rod groups must be within their insertion limits any time the reactor is critical or approaching criticality. For a control rod group to be considered above its insertion limit, all OPERABLE rods in that group, which are not misaligned, must be above the insertion limit (inoperable and misaligned rods are addressed by LCO 3.1.4). Maintaining the shutdown rod groups within their insertion limits ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. Maintaining the part-length rod group within its insertion limit ensures that the power distribution envelope is maintained.

APPLICABILITY

The shutdown and part-length rod groups must be within their insertion limits, with the reactor in MODES 1 and 2. In MODE 2 the Applicability begins anytime any regulating rod is withdrawn above 5 inches. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. In MODE 4, 5, or 6, the shutdown rod groups are inserted in the core to at least the lower electrical limit and contribute to the SDM. In MODE 3 the shutdown rod groups may be withdrawn in preparation of a reactor startup. Refer to LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

The Applicability has been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.3 (rod exercise test). Control rod exercising verifies the freedom of the rods to move, and requires the individual shutdown rods to move below the LCO limits for their group. Only the full-length rods are required to be tested by SR 3.1.4.3. The part-length rods may also be moved however, if a part-length rod is moved below the limit of the associated LCO, the Required Actions of Condition A must be taken. Positioning of an individual control rod within its group is addressed by LCO 3.1.4, "Control Rod Alignment."

BASES

ACTIONS

LCOs 3.1.4, 3.1.5, and 3.1.6, and their ACTIONS were written to support each other. The combined intent is to assure the following:

1. There is adequate SDM available in withdrawn control rods to assure the reactor is shutdown by, and remains shutdown following, a reactor trip,

2. The control rod positioning does not cause unacceptable axial or radial flux peaking, and

3. The programmed rod withdrawal sequence and group overlap result in reactivity insertion rates within the assumptions of the Inadvertent Control Rod Bank Withdrawal Analyses.

The ACTIONS for rods that are mispositioned (misaligned or inserted beyond the limit) were written assuming that an OPERABLE rod discovered to be mispositioned would simply be re-positioned correctly. While the associated Conditions would have to be entered, the rod could be re-positioned (thus exiting the LCO) without taking any other Required Action. A rod which remains mispositioned was assumed to be inoperable. The analyses account for operation with one (and only one) mispositioned rod (a dropped rod being the limiting case). With more than one mispositioned rod, the plant would be outside the bounds of the analyses and must be shutdown.

If a rod is discovered to be misaligned (ie, there is more than 8" between it and any other rod in its group, but all remaining rods in that group are within 8 inches of each other) Condition 3.1.4 C allows 2 hours to restore the rod alignment (thus exiting the LCO), perform SR 3.2.2.1 (verification that radial peaking is within limits), or reduce power to ≤ 75% RTP.

If one or more shutdown rods are inserted beyond the insertion limit, 3.1.5 A is entered; the rods are declared inoperable and 3.1.4 D (when one rod is immovable but trippable) or 3.1.4 E (when a movable rod is inserted beyond its insertion limit, or when more than one rod is inoperable for any reason) must be entered.

If the rods can be moved, they should be withdrawn and all Conditions exited.

If one rod cannot be moved (but is still considered trippable), operation may continue in accordance with 3.1.4 D (and 3.1.4 C if it is misaligned).

BASES

ACTIONS (continued)

If more than one rod cannot be moved, Condition 3.1.4 E must also be entered. The plant must be in MODE 3 in 6 hours in accordance with ACTION 3.1.4 E.1.

If one or more part length rods are inserted beyond the limit, 3.1.5 A is entered; the rods are declared inoperable and 3.1.4 E is entered (and 3.1.4 C if it is misaligned). Condition 3.1.4 D is not applicable to part length rods since it only addresses full length rods.

If the rods can be moved, they should be withdrawn and all Conditions exited.

If any part length rods are inserted beyond the limit and cannot be moved, the plant must be placed in MODE 3 in 6 hours in accordance with ACTION 3.1.4 E.1.

If one or more regulating rods are inserted beyond the limit, 3.1.6 A is entered;

The rods must be restored to within limits (by rod withdrawal or power reduction) within two hours.

If a rod cannot be moved, it must be considered inoperable and Condition 3.1.4 D must be entered (and 3.1.4 C if it is misaligned). Condition 3.1.4 D allows continued operation with one inoperable, but trippable, rod until the next reactor shutdown (MODE 3 entry). If more than one rod cannot be moved, Condition 3.1.4 E must be entered. The plant must be in MODE 3 in 6 hours in accordance with ACTION 3.1.4 E.1.

The analyses do not account for the possibility of more then one rod failing to insert on a trip. While boron concentration might be adjusted to restore SHUTDOWN MARGIN, if two adjacent rods fail to insert that portion of the core could remain excessively reactive. Since the analyses must assume that one rod fails to insert, operation may not continue with a known untrippable rod. A shutdown would be required by Condition 3.1.4 E.

BASES

ACTIONS (continued)

A.1

Prior to entering this condition, the shutdown and part-length rod groups were fully withdrawn. If a shutdown rod group is then inserted into the core, its potential negative reactivity is added to the core as it is inserted.

If one or more shutdown or part-length rods are not within limits, the affected rod(s) must be declared inoperable and the applicable Conditions and Required Actions of LCO 3.1.4 entered immediately. This Required Action is based on the recognition that the shutdown and part-length rods are normally withdrawn beyond their insertion limits and are capable of being moved by their control rod drive mechanism. Although the requirements of this LCO are not applicable during performance of the control rod exercise test, the inability to restore a control rod to within the limits of the LCO following rod exercising would be indicative of a problem affecting the OPERABILITY of the control rod. Therefore, entering the applicable Conditions and Required Actions of LCO 3.1.4 is appropriate since they provide the applicable compensatory measures commensurate with the inoperability of the control rod.

B.1

When Required Action A.1 cannot be met or completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown and part-length rod groups are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown rods will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. Verification that the part-length rod groups are within their insertion limits ensures that they do not adversely affect power distribution requirements. This SR and Frequency ensure that the shutdown and part-length rod groups are withdrawn before the regulating rods are withdrawn during a plant startup.

Since control rod groups are positioned manually by the control room operator, verification of shutdown and part-length rod group position at a Frequency of 12 hours is adequate to ensure that the shutdown and part-length rod groups are within their insertion limits. Also, the 12 hour Frequency takes into account other information available to the operator in the control room for the purpose of monitoring the status of the shutdown and part-length rod groups.

REFERENCES

1. FSAR, Section 5.1
2. FSAR, Section 14.2
3. FSAR, Section 14.6

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Regulating Rod Group Position Limits

BASES

BACKGROUND

The insertion limits of the regulating rod groups are initial assumptions in all safety analyses that assume full-length rod insertion upon reactor trip. The insertion limits directly affect core power distributions, assumptions of available SDM, and initial reactivity insertion rate. The applicable criteria for these reactivity and power distribution design requirements are contained in the Palisades Nuclear Plant design criteria (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Limits on regulating rod group insertion have been established, and all regulating rod group positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking, ejected rod worth, reactivity insertion rate, and SDM limits are preserved.

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). The regulating rod groups are withdrawn and operate in a predetermined sequence. 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 manually controlled. They are capable of changing reactivity very quickly (compared to borating or diluting).

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. 2). Together, LCO 3.1.6; LCO 3.2.3, "QUADRANT POWER TILT (T_q)"; and LCO 3.2.4, "AXIAL SHAPE INDEX (ASI)," provide limits on control component operation and on monitored process variables to ensure the core operates within the linear heat rate (LCO 3.2.1, "Linear Heat Rate (LHR)") and F_R^T (LCO 3.2.2, "TOTAL RADIAL PEAKING FACTOR (F_R^T)") limits in the COLR.

BASES

BACKGROUND (continued)

Operation within the LHR limits given in the COLR prevents power peaks that would exceed the Loss Of Coolant Accident (LOCA) limits derived by the Emergency Core Cooling System analysis. Operation within the F_R^T limits given in the COLR prevents Departure from Nucleate Boiling (DNB) during a loss of forced reactor coolant flow accident. In addition to the LHR and F_R^T limits, certain reactivity limits are preserved by regulating rod insertion limits. The regulating rod group insertion limits also restrict the ejected rod worth to the values assumed in the safety analysis and preserve the minimum required SDM in MODES 1 and 2.

The ejected rod case is limited to the reactivity worth for the highest worth rod ejected from the PDIL limit, thus limiting the maximum possible reactivity excursion.

The establishment of limiting safety system settings and LCOs requires that the expected long and short term behavior of the F_R^T be determined. The long term behavior relates to the variation of the steady state F_R^T with core burnup and is affected by the amount of rod insertion assumed, the portion of a burnup cycle over which such insertion is assumed, and the expected power level variation throughout the cycle. The short term behavior relates to transient perturbations to the steady state radial peaks, due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the rods during anticipated power reductions and load maneuvering. Analyses are performed, based on the expected mode of operation of the Nuclear Steam Supply System (base loaded, maneuvering, etc.). The PDIL curve stated in the COLR dictates the acceptable regulating rod group positioning for anticipated power maneuvers and transient mitigation within the limits. The PDIL limitations stated in the COLR reflect the assumptions made in the safety analyses. This ensures that the F_R^T limits are not violated during power level maneuvering or transient mitigation.

The regulating rod group insertion and alignment limits are process variables that together characterize and control the three dimensional power distribution of the reactor core. Additionally, the regulating rod group insertion limits control the reactivity that could be added in the event of a control rod ejection accident, and the shutdown and regulating bank insertion limits ensure the required SDM is maintained.

BASES

BACKGROUND (continued)	Operation within the subject LCO limits will prevent 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, ejected rod, or other 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 (Condition I) and anticipated operational occurrences (Condition II). The acceptance criteria for the regulating rod group position, ASI, and T_q LCOs are such as to preclude core power distributions from occurring that would violate the following fuel design criteria:

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

b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at a 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. The rods must be capable of shutting down the reactor with a minimum required SDM, with the highest worth rod stuck fully withdrawn (Ref. 1).

Regulating rod group position, ASI, and T_q are process variables that together characterize and control the three dimensional power distribution of the reactor core.

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 with simultaneous violation of one or more of these LCOs. Changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SDM requirement is ensured by limiting the regulating and shutdown rod group insertion limits, so that the allowable inserted worth of the rods is such that sufficient reactivity is available to shut down the reactor to hot zero power. SDM assumes the maximum worth rod remains fully withdrawn upon trip (Ref. 4).

The most limiting SDM requirements for Mode 1 and 2 conditions at Beginning of Cycle (BOC) are determined by the requirements of several transients, e.g., Loss of Flow, etc. However, the most limiting SDM requirements for MODES 1 and 2 at End of Cycle (EOC) come from just one transient, Main Steam Line Break (MSLB). The requirements of the MSLB event at EOC for the full power and no load conditions are significantly larger than those of any other event at that time in cycle and, also, considerably larger than the most limiting requirements at BOC.

Although the most limiting SDM requirements at EOC are much larger than those at BOC, the available SDMs obtained via tripping the full-length control rods are substantially larger due to the much lower boron concentration at EOC. To verify that adequate SDMs are available throughout the cycle to satisfy the changing requirements, calculations are performed at both BOC and EOC. It has been determined that calculations at these two times in cycle are sufficient since the difference between available SDMs and the limiting SDM requirements are the smallest at these times in cycle. The measurement of full-length control rod bank worth performed as part of the Startup Testing Program demonstrates that the core has the expected shutdown capability. Consequently, adherence to LCO 3.1.5, "Shutdown and Part-Length Rod Group Insertion Limits," and LCO 3.1.6 provides assurance that the available SDM at any time in cycle will exceed the limiting SDM requirements at that time in cycle.

Operation at the insertion limits or ASI limits may approach the maximum allowable linear heat generation rate or peaking factor, with the allowed T_q present. Operation at the insertion limit may also indicate the maximum ejected rod worth could be equal to the limiting value in fuel cycles that have sufficiently high ejected rod worth.

The regulating and shutdown rod insertion limits ensure that safety analyses assumptions for reactivity insertion rate, SDM, ejected rod worth, and peaking factors are preserved.

The regulating rod group position limits satisfy Criterion 2 of 10 CFR 50.36(c)(2).

BASES

LCO

The limits on regulating rod group sequence, overlap, and physical insertion, as defined in the COLR, must be maintained because they serve the function of preserving power distribution, ensuring that the SDM is maintained, ensuring that ejected rod worth is maintained, and ensuring adequate negative reactivity insertion on trip. The overlap between regulating rod groups provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during regulating rod group motion. For a control rod group to be considered above its insertion limit, all OPERABLE rods in that group, which are not misaligned, must be above the insertion limit (inoperable and misaligned rods are addressed by LCO 3.1.4).

The Power Dependent Insertion Limit (PDIL) alarm circuit is required to be OPERABLE for notification that the regulating rod groups are outside the required insertion limits. The Control Rod Out Of Sequence (CROOS) alarm circuit is required to be OPERABLE for notification that the rods are not within the required sequence and overlap limits. When the PDIL or the CROOS alarm circuit is inoperable, the verification of rod group positions is increased to ensure improper rod alignment is identified before unacceptable flux distribution occurs. The PDIL and CROOS alarms can be generated by either the synchro based Primary Indication Processor (PIP) node, or the reed switch based Secondary Position Indication (SPI) system since the SPI system, in conjunction with the host computer, is redundant to the PIP node in the task of control rod measurement, control rod monitoring and limit processing.

APPLICABILITY

The regulating rod group sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits must be maintained, since they preserve the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions. Applicability in MODES 3, 4, and 5 is not required, since neither the power distribution nor ejected rod worth assumptions would be exceeded in these MODES. SDM is preserved in MODES 3, 4, and 5 by adjustments to the soluble boron concentration.

The Applicability has been modified by a Note indicating the LCO requirement is suspended while performing SR 3.1.4.3 (rod exercise test). Control rod exercising verifies the freedom of the rods to move, and requires the individual regulating rods to move below the LCO limits which could violate the LCO for their group.

BASES

ACTIONS

LCOs 3.1.4, 3.1.5, and 3.1.6, and their ACTIONS were written to support each other. The combined intent is to assure the following:

1. There is adequate SDM available in withdrawn control rods to assure the reactor is shutdown by, and remains shutdown following, a reactor trip,

2. The control rod positioning does not cause unacceptable axial or radial flux peaking, and

3. The programmed rod withdrawal sequence and group overlap result in reactivity insertion rates within the assumptions of the Inadvertent Control Rod Bank Withdrawal Analyses.

The ACTIONS for rods that are mispositioned (misaligned or inserted beyond the limit) were written assuming that an OPERABLE rod discovered to be mispositioned would simply be re-positioned correctly. While the associated Conditions would have to be entered, the rod could be re-positioned (thus exiting the LCO) without taking any other Required Action. A rod which remains mispositioned was assumed to be inoperable. The analyses account for operation with one (and only one) mispositioned rod (a dropped rod being the limiting case). With more than one mispositioned rod, the plant would be outside the bounds of the analyses and must be shutdown.

If a rod is discovered to be misaligned (ie, there is more than 8" between it and any other rod in its group, but all remaining rods in that group are within 8 inches of each other) Condition 3.1.4 C allows 2 hours to restore the rod alignment (thus exiting the LCO), perform SR 3 2.2.1 (verification that radial peaking is within limits), or reduce power to ≤ 75% RTP.

If one or more shutdown rods are inserted beyond the insertion limit, 3.1.5 A is entered; the rods are declared inoperable and 3.1.4 D (when one rod is immovable but trippable) or 3.1.4 E (when a movable rod is inserted beyond its insertion limit, or when more than one rod is inoperable for any reason) must be entered.

If the rods can be moved, they should be withdrawn and all Conditions exited.

If one rod cannot be moved (but is still considered trippable), operation may continue in accordance with 3.1.4 D (and 3.1.4 C if it is misaligned).

BASES

ACTIONS (continued)

If more than one rod cannot be moved, Condition 3.1.4 E must also be entered. The plant must be in MODE 3 in 6 hours in accordance with ACTION 3.1.4 E.1.

If one or more part length rods are inserted beyond the limit, 3.1.5 A is entered; the rods are declared inoperable and 3.1.4 E is entered (and 3.1.4 C if it is misaligned). Condition 3.1.4 D is not applicable to Part Length rods since it only addresses full length rods.

If the rods can be moved, they should be withdrawn and all Conditions exited.

If any part length rods are inserted beyond the limit and cannot be moved, the plant must be placed in MODE 3 in 6 hours in accordance with ACTION 3.1.4 E.1.

If one or more regulating rods are inserted beyond the limit, 3.1.6 A is entered;

The rods must be restored to within limits (by rod withdrawal or power reduction) within two hours.

If a rod cannot be moved, it must be considered inoperable and Condition 3.1.4 D must be entered (and 3.1.4 C if it is misaligned). Condition 3.1.4 D allows continued operation with one inoperable, but trippable, rod until the next reactor shutdown (MODE 3 entry). If more than one rod cannot be moved, Condition 3.1.4 E must be entered. The plant must be in MODE 3 in 6 hours in accordance with ACTION 3.1.4 E.1.

The analyses do not account for the possibility of more then one rod failing to insert on a trip. While boron concentration might be adjusted to restore SHUTDOWN MARGIN, if two adjacent rods fail to insert that portion of the core could remain excessively reactive. Since the analyses must assume that one rod fails to insert, operation may not continue with a known untrippable rod. A shutdown would be required by Condition 3.1.4 E.

BASES

ACTIONS (continued)

A.1 and A.2

Operation beyond the insertion limit may result in a loss of SDM and excessive peaking factors. The insertion limit should not be violated during normal operation; this violation, however, may occur during transients when the operator is manually controlling the regulating rods in response to changing plant conditions.

When the regulating groups are inserted beyond the insertion limits, actions must be taken to either withdraw the regulating groups beyond the limits or to reduce THERMAL POWER to less than or equal to that allowed for the actual rod group position limit. Two hours provides a reasonable time to accomplish this, allowing the operator to deal with current plant conditions while limiting peaking factors to acceptable levels.

B.1

Operating outside the regulating rod group sequence and overlap limits specified in the COLR may result in excessive peaking factors. If the sequence and overlap limits are exceeded, the regulating rod groups must be restored to within the appropriate sequence and overlap. Two hours provides adequate time for the operator to restore the regulating rod group to within the appropriate sequence and overlap limits.

C.1

When the PDIL or the CROOS alarm circuit is inoperable, performing SR 3.1.6.1 once within 15 minutes following any rod motion ensures improper rod alignments are identified before unacceptable flux distributions occur.

D.1

When a Required Action cannot be completed within the required Completion Time, a controlled shutdown should be commenced. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

With the PDIL alarm circuit OPERABLE, verification of each regulating rod group position every 12 hours is sufficient to detect rod positions that may approach the acceptable limits, and to provide the operator with time to undertake the Required Action(s) should the sequence or insertion limits be found to be exceeded.

The 12 hour Frequency also takes into account the indication provided by the PDIL alarm circuit and other information about rod group positions available to the operator in the control room.

SR 3.1.6.2

Demonstrating the PDIL alarm circuit OPERABLE verifies that the PDIL alarm circuit is functional. The 31 day Frequency takes into account other Surveillances being performed at shorter Frequencies that identify improper control rod alignments.

SR 3.1.6.3

Demonstrating the CROOS alarm circuit OPERABLE verifies that the CROOS alarm circuit is functional. The 31 day Frequency takes into account other Surveillances being performed at shorter Frequencies that identify improper control rod alignment.

REFERENCES

1. FSAR, Section 5.1
2. 10 CFR 50.46
3. FSAR, Section 14.16
4. FSAR, Section 14.4

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Special Test Exceptions (STE)

BASES

BACKGROUND

The primary purpose of this STE is to permit relaxation of existing LCOs to allow the performance of certain PHYSICS TESTS. These tests are conducted to determine control rod worths, SHUTDOWN MARGIN (SDM), and specific reactor core characteristics.

Section XI of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants" (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. Testing is required as an integral part of the design, fabrication, construction, and operation of the power plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59, "Changes, tests, and experiments" (Ref. 2).

The key objectives of a test program are to (Ref. 3):

a. Ensure that the facility has been adequately designed;

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

c. Verify assumptions used for predicting plant response;

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

e. Verify that operating and emergency procedures are adequate.

To accomplish these objectives, testing is required during startup and low power operation after each shutdown that involved an alteration of the fuel assemblies in the reactor core. 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.

BASES

BACKGROUND (continued)

PHYSICS TESTS procedures are written and approved in accordance with the administrative processes for procedure controls. The procedures include all information necessary to permit a detailed execution of testing required to ensure that design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to power escalation.

Examples of PHYSICS TESTS include determination of critical boron concentration, full-length control rod group and individual control rod worths, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because fuel damage criteria are not exceeded. Even if an accident occurs during a PHYSICS TEST with one or more LCOs suspended, fuel damage criteria are preserved because the limits on power distribution and shutdown capability are maintained during PHYSICS TESTS.

Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1997 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. As long as the Linear Heat Rate (LHR) remains within its limit, fuel design criteria are preserved.

In this test, the following LCOs are suspended:

a. LCO 3.1.4, "Control Rod Alignment";

b. LCO 3.1.5, "Shutdown and Part-Length Rod Group Insertion Limits";

c. LCO 3.1.6, "Regulating Rod Group Position Limits"; and

d. LCO 3.4.2, "PCS Minimum Temperature for Criticality."

This STE places limits on allowable THERMAL POWER during PHYSICS TESTS assuring the LHR and the Departure from Nucleate Boiling (DNB) parameters will be maintained within limits. It also places limits on the amount of control rod worth required to be available for reactivity control when control rod worth measurements are performed.

APPLICABLE SAFETY ANALYSES (continued)

SRs are conducted as necessary to ensure that reactor power and shutdown capability remain within limits during PHYSICS TESTS. Requiring ≥ 1% shutdown reactivity, based on predicted control rod worths, be available for trip insertion from the OPERABLE full-length control rod provides a high degree of assurance that shutdown capability is maintained for the most challenging postulated accident assuming all full-length control rods are inserted in the core. Since LCOs 3.1.5 and 3.1.6 are suspended, however, there is not the same degree of assurance during this test that the reactor would always be shut down if the highest worth full-length control rod was stuck out and calculational uncertainties or the estimated highest rod worth was not as expected (the single failure criterion is not met). This situation is judged acceptable, however, because specified acceptable fuel damage limits are still met. The risk of experiencing a stuck rod and subsequent criticality is reduced during this PHYSICS TEST exception by the requirement that ≥ 1% shutdown reactivity is available based on predicted control rod worths.

PHYSICS TESTS include measurement of core parameters or exercise of control components. Also involved are the shutdown and regulating rods, which affect power peaking and are required for shutdown of the reactor. The limits for insertion of these rod groups are specified for each fuel cycle in the COLR.

As described in LCO 3.0.7, compliance with Special Test Exceptions LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2) apply. Special Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

This LCO relaxes the minimum primary coolant temperature at which the reactor may be made critical, permits individual full-length control rods and full-length control rod groups to be positioned outside of their normal alignment and insertion limits during the performance of PHYSICS TESTS such as those required to:

a. Measure control rod worths;

b. Measure control rod shadowing factors; and

c. Measure temperature and power coefficients.

BASES

LCO (continued)

This LCO specifies that a minimum amount of rod worth is immediately available for reactivity control when rod worth measurement tests are performed. This portion of the STE permits the periodic verification of the actual versus predicted control rod group worths.

The requirements of LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended during the performance of PHYSICS TESTS, provided:

a. THERMAL POWER is ≤ 2% RTP;

b. ≥ 1% shutdown reactivity, based on predicted control rod worth, is available for trip insertion; and

c. T_{ave} is ≥ 500°F.

APPLICABILITY

This LCO is applicable in MODE 2 because the reactor must be critical to perform the PHYSICS TESTS described in the LCO section.

ACTIONS

A.1

If THERMAL POWER exceeds 2% RTP, THERMAL POWER must be reduced to restore the additional thermal margin provided by the reduction. The 15 minute Completion Time ensures that prompt action shall be taken to reduce THERMAL POWER to within acceptable limits.

B.1

If < 1% shutdown reactivity is available for trip insertion, 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 plant conditions. Boration will be continued until ≥ 1% shutdown reactivity is achieved.

C.1

If the T_{ave} requirement is not met, T_{ave} must be restored. The 15 minutes Completion Time ensures that prompt action shall be taken to raise T_{ave} within the required limit.

ACTIONS (continued)

D.1

If Required Actions of Condition A, Condition B, or Condition C cannot be completed within the required Completion Time, PHYSICS TESTS must be suspended within 1 hour. Allowing 1 hour for suspending PHYSICS TESTS allows the operator sufficient time to change any abnormal rod configuration back to within the limits of LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6, or to restore Primary Coolant System (PCS) temperature to within the limits of LCO 3.4.2.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

Verifying that THERMAL POWER is ≤ 2% RTP as specified in the PHYSICS TEST procedure and required by the safety analysis, ensures that adequate LHR and DNB parameter margins are maintained while LCOs are suspended. The 1 hour Frequency is sufficient, based on the slow rate of power change and increased operational controls in place during PHYSICS TESTS.

SR 3.1.7.2

Verifying T_{ave} ≥ 500°F during the PHYSICS TEST ensures that T_{ave} remains in an analyzed range while the LCOs are suspended. The 1 hour Frequency is sufficient, based on the slow rate of change and increased operational controls in place during PHYSICS TESTS.

SR 3.1.7.3

Verification that ≥ 1% shutdown reactivity is available for trip insertion is performed by a reactivity balance calculation, considering the following reactivity effects:

a. PCS boron concentration;

b. Control rod group position;

c. PCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

BASES

SURVEILLANCE REQUIREMENTS	<u>SR 3.1.7.3</u> (continued) e. Xenon concentration; and f. Isothermal Temperature Coefficient (ITC). Using the ITC accounts for Doppler reactivity in this calculation because reactor power is maintained below 2% RTP, and for most of the PHYSIC TESTS below the point of adding heat the fuel temperature will be changing at the same rate as the PCS. The Frequency of 24 hours is based on the generally slow change in boron concentration and on the low probability of an accident occurring without the SDM established by LCO 3.1.5.
REFERENCES	1. 10 CFR 50, Appendix B, Section XI 2. 10 CFR 50.59 3. Regulatory Guide 1.68, Revision 2, August 1978 4. ANSI/ANS-19.6.1-1997, August 22, 1997

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Linear Heat Rate (LHR)

BASES

BACKGROUND

The purpose of this LCO is to limit the core power distribution to the initial values assumed in the accident analyses. Operation within the limits imposed by this LCO either limits or prevents potential fuel cladding failures that could breach the primary fission product barrier and release fission products to the primary coolant in the event of a Loss Of Coolant Accident (LOCA), loss of flow accident, ejected control rod 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 ensuring that the plant is operating within acceptable bounding conditions at the onset of a transient.

Methods of controlling the power distribution include:

a. Using control rods to alter the axial power distribution;

b. Decreasing control rod insertion by boration, thereby improving the radial power distribution; and

c. Correcting off optimum conditions (e.g., a control rod drop or misoperation of the plant) that cause margin degradations.

The core power distribution is controlled so that, in conjunction with other core operating parameters (e.g., control rod insertion and alignment limits), the power distribution satisfies this LCO. The limiting safety system settings and this LCO are based on the accident analyses (Refs. 1 and 2), so that specified acceptable fuel design limits are not exceeded as a result of Anticipated Operational Occurrences (AOOs), and the limits of acceptable consequences are not exceeded for other postulated accidents.

Limiting power distribution changes over time also minimizes the xenon distribution changes, which is a significant factor in controlling the axial power distribution.

Power distribution is a product of multiple parameters, various combinations of which may produce acceptable power distributions.

BASES

BACKGROUND (continued)

The limits on LHR, TOTAL RADIAL PEAKING FACTOR (F_R^T), QUADRANT POWER TILT (T_q), and AXIAL SHAPE INDEX (ASI), which are obtained directly from the core reload analysis, ensure compliance with the safety limits on LHR and Departure from Nucleate Boiling Ratio (DNBR).

Either of the two core power distribution monitoring systems, the Incore Alarm portion of the Incore Monitoring System or the Excore Monitoring System, provides adequate monitoring of the core power distribution and is capable of verifying that the LHR is within its limits. The Incore Alarm System performs this function by continuously monitoring the local power at many points throughout the core and comparing the measurements to predetermined setpoints above which the limit on LHR could be exceeded. The Excore Monitoring System performs this function by providing comparison of the measured core ASI with predetermined ASI limits based on incore measurements. An Excore Monitoring System Allowable Power Level (APL), which may be less than RATED THERMAL POWER, and an additional restriction on T_q, are applied when using the Excore Monitoring System to ensure that the ASI limits adequately restrict the LHR to less than the limiting values.

In conjunction with the use of the Excore Monitoring System for monitoring LHR and in establishing ASI limits, the following assumptions are made:

a. The control rod insertion limits of LCO 3.1.5, "Shutdown and Part-Length Rod Group Insertion Limits," and LCO 3.1.6, "Regulating Rod Group Position Limits," are satisfied;

b. The additional T_q restriction of SR 3.2.1.6 is satisfied; and

c. F_R^T, does not exceed the limits of LCO 3.2.2.

The limitations on the TOTAL RADIAL PEAKING FACTOR provided in the COLR ensure that the assumptions used in the analysis for establishing the LHR limits and Limiting Safety System Settings (LSSS) remain valid during operation at the various allowable control rod group insertion limits.

BASES

BACKGROUND (continued)

The Incore Monitoring System continuously provides a direct indication of the core power distribution. It also provides alarms that have been established for the individual incore detector segments, ensuring that the peak LHRs are maintained within the limits specified in the COLR. The setpoints for these alarms include tolerances, set in conservative directions, for:

a. A measurement calculational uncertainty factor (as identified in the COLR);

b. An engineering uncertainty factor of 1.03; and

c. A THERMAL POWER measurement uncertainty factor of 1.02.

The measurement uncertainties associated with LHR and F_R^T are based on a statistical analysis performed on power distribution benchmarking results. The COLR includes the applicable measurement uncertainties for incore detector usage. The engineering and THERMAL POWER uncertainties are incorporated in the power distribution calculation performed by the fuel vendor.

The excore power distribution monitoring system consists of Power Range Channels 5 through 8. The power range channels monitor neutron flux from 0 to 125 percent full power. They are arranged symmetrically around the reactor core to provide information on the radial and axial flux distributions.

The power range detector assembly consists of two uncompensated ion chambers for each channel. One detector extends axially along the lower half of the core while the other, which is located directly above it, monitors flux from the upper half of the core. The DC current signal from each of the ion chambers is fed directly to the control room drawer assembly without pre-amplification. Each excore detector supplies data to a Thermal Margin Monitor (TMM). Each TMM uses these excore signals to calculate Axial Shape Index (ASI) on a continuous basis.

ASI can be defined as the compensated ratio of power developed in the upper and lower sections of the core. The TMM takes the excore detector signals and develops a power ratio (YE) that describes the distribution of neutron flux developed in the core by the formula:

$$YE = (L - U)/(L + U)$$

Where L is the lower excore segment flux, and U is the upper excore segment flux.

BASES

BACKGROUND (continued)

The excore detectors which are located within the concrete biological shield of the reactor must be compensated for the phenomenon of shape annealing. Shape annealing factors are developed to correct the excore readings for neutron attenuation from the core periphery to the excore detector locations. This accounts for any material that would cause neutron attenuation within the detector path such as: concrete, structural steel and so forth. This allows the excore detectors to represent an accurate measurement of the core power distribution. Shape annealing has been found to be a linear relationship which can be correlated to the Axial Offset (AO) as determined by an Incore Detector System to the raw readings seen by the excore detectors.

Reactor Engineering has developed shape annealing factors for each individual Excore detector. The TMM uses the above calculated power ratio and the appropriate shape annealing factor to determine the ASI value for an individual excore detector channel.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) or AOOs (Condition 2) (Ref. 3). The power distribution and control rod insertion and alignment LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a LOCA, peak cladding temperature must not exceed 2200°F (Ref. 4);

b. During a loss of 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 (Ref. 3).

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

d. The full-length control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The power density at any point in the core must be limited to maintain the fuel design criteria (Ref. 4). This is accomplished by maintaining the power distribution and primary coolant conditions so that the peak LHR and DNB parameters are within operating limits supported by accident analyses (Ref. 1), with due regard for the correlations between measured quantities, the power distribution, and uncertainties in determining the power distribution.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Fuel cladding failure during a LOCA is limited by restricting the maximum linear heat generation rate so that the peak cladding temperature does not exceed 2200°F (Ref. 4). High peak cladding temperatures are assumed to cause severe cladding failure by oxidation due to a Zircaloy water reaction.

The LCOs governing LHR, ASI, and the Primary Coolant System Operation ensure that these criteria are met as long as the core is operated within the LHR, ASI, F_R^T, and T_q limits. The latter are process variables that characterize the three dimensional power distribution of the reactor core. Operation within the limits for these variables ensures that their actual values are within the ranges used in the accident analyses.

Fuel cladding damage does not necessarily occur while the plant is operating at conditions outside the limits of these LCOs during normal operation. Fuel cladding damage could result, however, if an accident occurs from initial conditions outside the limits of these LCOs. The potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and can correspondingly increase local LHR.

The Incore Monitoring System provides for monitoring of LHR, F_R^T, and QUADRANT POWER TILT to ensure that fuel design conditions and safety analysis assumptions are maintained. The Incore Monitoring System is also utilized to determine the target AXIAL OFFSET (AO) and to determine the Allowable Power Level (APL) when using the excore detectors.

The Excore Monitoring System provides for monitoring of ASI and QUADRANT POWER TILT to ensure that fuel design conditions and safety analysis assumptions are maintained.

LHR satisfies Criterion 2 of 10 CFR 50.36(c)(2).

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The power distribution LCO limits, except T_q, are provided in the COLR. The limitation on the LHR in the peak power fuel rod at the peak power elevation Z ensures that, in the event of a LOCA, the peak temperature of the fuel cladding does not exceed 2200°F.

BASES

LCO (continued)

The LCO requires that LHR be maintained within the limits specified in the COLR and either the Incore Alarm System or Excore Monitoring System be OPERABLE to monitor LHR. When using the Incore Alarm System, the LHR is not considered to be out of limits until there are four or more incore detectors simultaneously in alarm. When using the Excore Monitoring System, LHR is considered within limits when the conditions are acceptable for use of the Excore Monitoring System and the associated ASI and T_q limits specified in the SRs are met.

To be considered OPERABLE, the Incore Alarm System must have at least 90 of the 180 incore detectors OPERABLE and 2 incore detectors per axial level per core quadrant OPERABLE. In addition, the plant process computer must be OPERABLE and the required alarm setpoints entered into the plant computer. Only 36 of the 45 instrument locations are included in the Incore Alarm System Uncertainty Analysis (180 of the possible 215 detectors). Instrument locations 1, 4, 13, 34, 41, 42 and 45 are not included, and instrument locations 7 and 44 are used by the Reactor Vessel Level Monitoring System (RVLMS).

To be considered OPERABLE, the Excore Monitoring System must have been calibrated with OPERABLE incore detectors, the ASI must not have been out of limits for the last 24 hours, and THERMAL POWER must be less than the APL.

APPLICABILITY

In MODE 1 with THERMAL POWER > 25% RTP, power distribution must be maintained within the limits assumed in the accident analysis to ensure that fuel damage does not result following an AOO. In MODE 1 with THERMAL POWER ≤ 25% RTP, and in other MODES, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on the core power distribution, and because ample thermal margin exists to ensure that the fuel integrity is not jeopardized and safety analysis assumptions remain valid.

ACTIONS

A.1

There are three acceptable methods for verifying that LHR is within limits. The LCO requires monitoring by either an OPERABLE Incore Alarm System or an OPERABLE Excore Monitoring System. When both of the required systems are inoperable, Condition B allows for monitoring by taking manual readings of the incore detectors. Any of these three methods may indicate that the LHR is not within limits. With the LHR exceeding its limit, excessive fuel damage could occur following an accident. In this Condition, prompt action must be taken to restore the LHR to within the specified limits. One hour to restore the LHR to within its specified limits is reasonable and ensures that the core does not continue to operate in this Condition. The 1 hour

BASES

ACTIONS (Continued)

A.1

Completion Time also allows the operator sufficient time for evaluating core conditions and for initiating proper corrective actions.

ACTIONS (continued)

B.1 and B.2

With the Incore Alarm System inoperable for monitoring LHR and the Excore Monitoring System inoperable for monitoring LHR, THERMAL POWER must be reduced to ≤ 85% RTP within 2 hours. Operation at ≤ 85% RTP ensures that ample thermal margin is maintained. A 2 hour Completion Time is adequate to achieve the required plant condition without challenging plant systems. Additionally, with the Incore Alarm and Excore Monitoring Systems inoperable, LHR must be verified to be within limits within 4 hours, and every 2 hours thereafter by manually collecting incore detector readings at the terminal blocks in the control room utilizing a suitable signal detector. The manual readings shall be taken on a minimum of 10 individual detectors per quadrant (to include a total of 90 detectors in a 10 hour period). The time interval of 2 hours and the minimum of 10 detectors per quadrant are sufficient to maintain adequate surveillance of the power distribution to detect significant changes until the monitoring systems are returned to service.

As stated in SR 3.0.2, the 25% extension allowed by SR 3.0.2 may be applied to Required Actions whose Completion Time is stated as "once per . . ." however, the 25% extension does not apply to the initial performance of a Required Action with a periodic Completion Time that requires performance on a "once per. . ." basis. The 25% extension applies to each performance of the Required Action after the initial performance. Therefore, while Required Action 3.2.1 B.2 must be initially performed within 4 hours without any SR 3.0.2 extension, subsequent performances at the "Once per 2 hours" interval may utilize the 25% SR 3.0.2 extension.

C.1

If the Required Action and associated Completion Time are not met, THERMAL POWER must be reduced to ≤ 25% RTP. This reduced power level ensures that the core is operating within its thermal limits and places the core in a conservative condition. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reach ≤ 25% RPT from full power MODE 1 conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

The Incore Alarm portion of the Incore Monitoring System provides continuous monitoring of LHR through the plant computer. The PIDAL computer program is used to generate alarm setpoints for the plant computer that are based on measured margin to allowed LHR. As the incore detectors are read by the plant computer, they are continuously compared to the alarm setpoints. If the Incore Alarm System LHR monitoring function is inoperable, excore detectors or manual recordings of the incore detector readings may be used to monitor LHR. Periodically monitoring LHR ensures that the assumptions made in the Safety Analysis are maintained. This SR is modified by a Note that states that the SR is only required to be met when the Incore Alarm System is being used to monitor LHR. The 12 hour Frequency is consistent with an SR which is to be performed each shift.

SR 3.2.1.2

Continuous monitoring of the LHR is provided by the Incore Alarm System which provides adequate monitoring of the core power distribution and is capable of verifying that the LHR does not exceed its specified limits.

Performance of this SR verifies the Incore Alarm System can accurately monitor LHR by ensuring the alarm setpoints are based on a measured power distribution. Therefore, they are only applicable when the Incore Alarm System is being used to determine the LHR.

The alarm setpoints must be initially adjusted following each fuel loading prior to operation above 50% RTP, and periodically adjusted every 31 Effective Full Power Days (EFPD) thereafter. A 31 EFPD Frequency is consistent with the historical testing frequency of the reactor monitoring system. The SR is modified by a Note which requires the SR to be met only when the Incore Alarm System is being used to determine LHR.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.3

SR 3.2.1.3 requires, prior to initial use of the excore LHR monitoring function, verification that the absolute difference of the measured ASI and the target ASI has been ≤ 0.05 for each OPERABLE channel for the last 24 hours using the previous 24 hourly recorded values. Performance of this SR verifies that plant conditions are acceptable for the Excore Monitoring System to accurately monitor the LHR (Ref. 5). The prior to initial use verification identifies that there have been no significant power distribution anomalies while using other monitoring methods, e.g., the incore detectors, which may affected the ability of the excore detectors to monitor LHR.

The SR is modified by a Note that states that the SR is only required to be met when the Excore Monitoring System is being used to monitor LHR. Failure of this SR prevents the Excore Monitoring System from being considered OPERABLE for monitoring of LHR.

SR 3.2.1.4

SR 3.2.1.4 requires verification that THERMAL POWER is less than or equal to the Allowable Power Level (APL) which is limited to not more than 10% greater than the THERMAL POWER at which the APL was last determined. Performance of this SR also verifies that plant conditions are acceptable for the Excore Monitoring System to accurately monitor the LHR (Ref. 5). The 1 hour Frequency is based on engineering judgement and the need to assure that conditions remain acceptable for use of the Excore Monitoring System to monitor LHR.

The SR is modified by a Note that states that the SR is only required to be met when the Excore Monitoring System is being used to monitor LHR. Failure of this SR prevents the Excore Monitoring System from being considered OPERABLE for monitoring of LHR.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.1.5

SR 3.2.1.5 requires verification that the absolute difference of the measured ASI and the target ASI is ≤ 0.05 every hour. This must be verified on at least 3 of the 4, 2 of the 3, or 2 of the 2 OPERABLE channels, whichever is the applicable case. However, any otherwise OPERABLE channel which indicates an absolute difference of > 0.05 must be considered out of limits. Performance of this SR verifies that plant conditions are acceptable for the Excore Monitoring System to be used to assure LHR is within limits (Ref. 5). The 1 hour Frequency is appropriate because the excore detectors input neutron flux information into the ASI calculation which is normally performed automatically and continuously.

The SR is modified by a Note that states that the SR is only required to be met when the Excore Monitoring System is being used to monitor LHR. Failure of this SR (when using an OPERABLE Excore Monitoring System) is a failure to verify that LHR is within limits and is therefore considered a failure to meet the LCO due to LHR not within limits as determined by the Excore Monitoring System.

SR 3.2.1.6

SR 3.2.1.6 requires verification that the QUADRANT POWER TILT is ≤ 0.03. Performance of this SR also verifies that plant conditions are acceptable for the Excore Monitoring System to be used to assure LHR is within limits (Ref. 5). The 24 hour Frequency is based on engineering judgement and the need to identify adverse trends in these parameters prior to their affecting the ability of the Excore Monitoring System to monitor LHR.

The SR is modified by a Note that states that the SR is only required to be met when the Excore Monitoring System is being used to monitor LHR. Failure of this SR (when using an OPERABLE Excore Monitoring System) is a failure to verify that LHR is within limits and is therefore considered a failure to meet the LCO due to LHR not within limits as determined by the Excore Monitoring System.

BASES

REFERENCES

1. FSAR, Chapter 14

2. FSAR, Chapter 6

3. FSAR, Section 5.1

4. 10 CFR 50.46

5. Safety Evaluation Report for Palisades Nuclear Plant Operating License Amendment No. 68, Section 4, dated December 8, 1981

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 TOTAL RADIAL PEAKING FACTOR (F_R^T)

BASES

BACKGROUND	The Background section of Bases B 3.2.1, "Linear Hear Rate," is applicable to these Bases.
APPLICABLE SAFETY ANALYSES	The Applicable Safety Analyses section of Bases B 3.2.1 is applicable to these Bases. The TOTAL RADIAL PEAKING FACTOR satisfies Criterion 2 of 10 CFR 50.36(c)(2).
LCO	The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. The power distribution LCO limits, except T_q, are provided in the COLR. The limitations on F_R^T are provided to ensure that assumptions used in the analysis for establishing DNB margin, LHR limit and the thermal margin/low pressure and variable high power trip setpoints remain valid during operation. Data from the incore detectors are used for determining the measured F_R^T.
APPLICABILITY	In MODE 1 with THERMAL POWER > 25% RTP, power distribution must be maintained within the limits assumed in the accident analyses to ensure that fuel damage does not result following an AOO. In MODE 1 with THERMAL POWER ≤ 25% RTP, and in other MODES, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on the core power distribution, and because ample thermal margin exists to ensure that the fuel integrity is not jeopardized and safety analysis assumptions remain valid.

BASES

ACTIONS

A.1

If F_R^T exceeds its limits, F_R^T must be restored to within the limits as identified in the COLR within 6 hours. Restoration may be either by correcting the source of the peaking or by a reduction in THERMAL POWER. The THERMAL POWER typically necessary to achieve restoration is identified by the equation:

$$P = \left[1 - 3.33\left((F_R/F_L) - 1\right)\right](RTP)$$

Where F_R is the measured value of F_R^T; and F_L is the corresponding limit provided in the COLR. Operating at or below this power level, P, is typically sufficient to restore F_R^T within limits. If power reductions do not restore F_R^T to within limits within 6 hours, Condition B is applicable.

Six hours to restore F_R^T to within limit(s) is reasonable and ensures that the core does not continue to operate in this condition for an extended period. The 6 hour Completion Time also allows the operator sufficient time for evaluating core conditions and for initiating proper corrective actions.

B.1

If the Required Action and associated Completion Time are not met, THERMAL POWER must be reduced to ≤ 25% RTP. This reduced power level ensures that the core is operating within its thermal limits and places the core in a conservative condition. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reach ≤ 25% RTP from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The periodic Surveillance to determine F_R^T ensures that F_R^T remains within the range assumed in the analysis throughout the fuel cycle. Determining F_R^T using the incore detectors after each fuel loading prior to the reactor exceeding 50% RTP ensures that the core is properly loaded.

Performance of the Surveillance every 31 Effective Full Power Days (EFPD) ensures that unacceptable changes in F_R^T are promptly detected.

REFERENCES

None

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 QUADRANT POWER TILT (T_q)

BASES

BACKGROUND

The Background section for Bases B 3.2.1, "Linear Heat Rate," is applicable to these Bases, with the following addition:

The power range monitoring system provides alarms when T_q exceeds predetermined values. The average of the four power range signals is developed by a single "Comparitor Averager." Each power range channel compares its output signal to this average signal. Two channel deviation alarm bistables, set at different levels, are provided in each power range channel. The deviation alarms will annunciate when the associated channel signal is either above or below the average, however, only a signal above the average is of concern with regard to T_q.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases B 3.2.1 is applicable to these Bases.

The T_q satisfies Criterion 2 of 10 CFR 50.36(c)(2).

LCO

The power distribution LCO limits are based on correlations between power peaking and the measured variables used as inputs to the LHR and DNBR operating limits. The power distribution LCO limits, except T_q, are provided in the COLR. The limits on T_q ensure that assumptions used in the analysis for establishing LHR limits and DNB margin remain valid during operation.

APPLICABILITY

In MODE 1 with THERMAL POWER > 25% RTP, power distribution must be maintained within the limits assumed in accident analysis to ensure that fuel damage does not result following an AOO. In MODE 1 with THERMAL POWER ≤ 25% RTP, and in other MODES, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on core power distribution, and because ample thermal margin exists to ensure that the fuel integrity is not jeopardized and safety analysis assumptions remain valid.

BASES

ACTIONS

A.1

If the measured T_q is > 0.05, T_q must be restored within 2 hours or F_R^T must be determined to be within the limits of LCO 3.2.2, and determined to be within these limits every 8 hours thereafter, as long as T_q is out of limits. Two hours is sufficient time to allow the operator to reposition control rods, and significant radial xenon redistribution cannot occur within this time. The 8 hour Completion Time ensures changes in F_R^T can be identified before the limits of LCO 3.2.2 are exceeded.

As stated in SR 3.0.2, the 25% extension allowed by SR 3.0.2 may be applied to Required Actions whose Completion Time is stated as "once per . . ." however, the 25% extension does not apply to the initial performance of a Required Action with a periodic Completion Time that requires performance on a "once per. . ." basis. The 25% extension applies to each performance of the Required Action after the initial performance. Therefore, while Required Action 3.2.3 A.1 must be initially performed within 2 hours without any SR 3.0.2 extension, subsequent performances at the "Once per 8 hours" interval may utilize the 25% SR 3.0.2 extension.

B.1

With the measured T_q > 0.10, power must be reduced to < 50% RTP within 4 hours, F_R^T must be within specified limits to ensure that acceptable flux peaking factors are maintained as required by Condition A (which continues to be applicable). Based on operating experience, 4 hours is sufficient time for evaluation of these factors. If F_R^T is within limits, operation may proceed while attempts are made to restore T_q to within its limit. If the tilt is generated due to a control rod misalignment, continued operation at < 50% RTP allows for realignment; if the cause is other than control rod misalignment, continued operation may be necessary to discover the cause of the tilt. Reducing THERMAL POWER to < 50% RTP, and the more frequent measurement of F_R^T required by ACTION A.1, provide conservative protection from potential increased peaking due to xenon redistribution.

BASES

ACTIONS (continued)

C.1

If T_q is > 0.15, or if Required Actions and associated Completion Times are not met, THERMAL POWER must be reduced to ≤ 25% RTP. This requirement ensures that the core is operating within its thermal limits and places the core in a conservative condition. Four hours is a reasonable time to reach 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

QUADRANT POWER TILT (T_q) is determined from excore detector readings which are calibrated using incore detector measurements (Ref. 1). Calibration factors are determined using incore measurements and an incore analysis computer program (Ref. 2). Each power range channel provides alarms if T_q exceeds its limits. Therefore, with all power range channels OPERABLE, this SR only requires verification that the channel deviation alarms do not indicate an excessive T_q. If the Excore Monitoring System T_q deviation alarm monitoring function is inoperable, excore detector readings or symmetric incore detector readings may be used to monitor T_q at 12 hour intervals. The 12 hour Frequency prevents significant xenon redistribution between Surveillances.

REFERENCES

1. FSAR, Section 7.6.2.2
2. FSAR, Section 7.6.2.4

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 AXIAL SHAPE INDEX (ASI)

BASES

BACKGROUND

The Background section for Bases B 3.2.1, "Linear Heat Rate," is applicable to these Bases, with the following addition:

The Excore Monitoring System ASI alarm function consists of four channels. At least two channels of the ASI alarm function are necessary to verify that ASI is within limits. With one or more excore monitoring channels measured ASI differing from the incore measured AO by > 0.02 under steady state operating conditions, the ASI monitoring channel alarm setpoint may be adjusted to compensate for this deviation. This ensures that fuel design parameters can continue to be accurately monitored and not exceeded when the incore/excore alignment is not within normal tolerances. This may occur when the calibration cannot be performed or the alignment problem exists after the calibration.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section for Bases B 3.2.1 is applicable to these Bases.

The ASI satisfies Criterion 2 of 10 CFR 50.36(c)(2).

LCO

The power distribution LCO limits are based on correlations between power peaking and certain measured variables used as inputs to the LHR and DNBR operating limits. These power distribution LCO limits, except T_q, are provided in the COLR. The limitation on ASI ensures that the assumed axial power profiles used in the development of the inlet temperature LCO bound the measured axial power profile.

The limitation on ASI, along with the limitations of LCO 3.3.1, "Reactor Protective System Instrumentation," represents a conservative envelope of operating conditions consistent with the assumptions that have been analytically demonstrated adequate for maintaining an acceptable minimum DNBR throughout all AOOs. Operation of the core with conditions within the specified limits ensures that an acceptable minimum margin from DNB conditions is maintained in the event of any AOO.

BASES

APPLICABILITY

In MODE 1 with THERMAL POWER > 25% RTP, power distribution must be maintained within the limits assumed in the accident analyses to ensure that fuel damage does not result following an AOO. In MODE 1 with THERMAL POWER ≤ 25% RTP, and in other MODES, this LCO does not apply because there is not sufficient THERMAL POWER to require a limit on the core power distribution, and because ample thermal margin exists to ensure that the fuel integrity is not jeopardized and safety analysis assumptions remain valid.

ACTIONS

A.1

Operating the core within ASI limits specified in the COLR and within the limits of LCO 3.3.1 ensures an acceptable margin for DNB and for maintaining local power density in the event of an AOO. Maintaining ASI within limits also ensures that the limits of 10 CFR 50.46 are not exceeded during accidents. The Required Actions to restore ASI must be completed within 2 hours to limit the duration the plant is operated outside the initial conditions assumed in the accident analyses. In addition, this Completion Time is sufficiently short that the xenon distribution in the core cannot change significantly.

B.1

If the Required Action and associated Completion Time are not met, core power must be reduced. Reducing THERMAL POWER to ≤ 25% RTP ensures that the core is operating farther from thermal limits and places the core in a conservative condition. Four hours is a reasonable amount of time, based on operating experience, to reduce THERMAL POWER to ≤ 25% RTP in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

Verifying that the ASI is within the limits specified in the COLR ensures that the core is not approaching DNB conditions. ASI is determined from excore detector readings which are calibrated using incore detector measurements (Ref. 1). Calibration factors are determined using incore measurements and an incore analysis computer program (Ref. 2). ASI is normally calculated and compared to the alarm setpoints continuously and automatically. Therefore, this SR only requires verification that alarms do not indicate an excessive ASI. If the Excore Monitoring System ASI Alarm function is inoperable, excore detector or incore indications may be used to monitor ASI. A Frequency of 12 hours is adequate for the operator to identify trends in conditions that result in an approach to the ASI limits, because the mechanisms that affect the ASI, such as xenon redistribution or control rod drive mechanism malfunctions, cause the ASI to change slowly and should be discovered before the limits are exceeded.

REFERENCES

1. FSAR, Section 7.6.2.2

2. FSAR, Section 7.6.2.4

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Protective System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a reactor trip to protect against violating the acceptable fuel design limits and breaching the reactor coolant pressure boundary during Anticipated Operational Occurrences (AOOs). (As defined in 10 CFR 50, Appendix A, "Anticipated operational occurances mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.") By tripping the reactor, the RPS also assists the Engineered Safety Features (ESF) systems in mitigating accidents.

The protection and monitoring systems have been designed to ensure safe operation of the reactor. This is achieved by specifying Limiting Safety System Settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this Specification as the Allowable Values, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).

During AOOs, which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling;
- Fuel centerline melting shall not occur; and
- The Primary Coolant System (PCS) pressure SL of 2750 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 50 (Ref. 1) and 10 CFR 100 (Ref. 2) criteria during AOOs.

BASES

BACKGROUND (continued)

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 (Ref. 2) limits. Different accident categories allow a different fraction of these limits based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The RPS is segmented into four interconnected modules. These modules are:

- Measurement channels;
- RPS trip units;
- Matrix Logic; and
- Trip Initiation Logic.

This LCO addresses measurement channels and RPS trip units. It also addresses the automatic bypass removal feature for those trips with Zero Power Mode bypasses. The RPS Logic and Trip Initiation Logic are addressed in LCO 3.3.2, "Reactor Protective System (RPS) Logic and Trip Initiation." The role of the measurement channels, RPS trip units, and RPS Bypasses is discussed below.

Measurement Channels

Measurement channels, consisting of pressure switches, field transmitters, or process sensors and associated instrumentation, provide a measurable electronic signal based upon the physical characteristics of the parameter being measured.

With the exception of Hi Startup Rate, which employs two instrument channels, and Loss of Load, which employs a single pressure sensor, four identical measurement channels with electrical and physical separation are provided for each parameter used in the direct generation of trip signals. These are designated channels A through D. Some measurement channels provide input to more than one RPS trip unit within the same RPS channel. In addition, some measurement channels may also be used as inputs to Engineered Safety Features (ESF) bistables, and most provide indication in the control room.

BASES

BACKGROUND (continued)

Measurement Channels (continued)

In the case of Hi Startup Rate and Loss of Load, where fewer than four sensor channels are employed, the reactor trips provided are not relied upon by the plant safety analyses. The sensor channels do however, provide trip input signals to all four RPS channels.

When a channel monitoring a parameter exceeds a predetermined setpoint, indicating an abnormal condition, the bistable monitoring the parameter in that channel will trip. Tripping two or more channels of bistable trip units monitoring the same parameter de-energizes Matrix Logic, (addressed by LCO 3.3.2) which in turn de-energizes the Trip Initiation Logic. This causes all four DC clutch power supplies to de-energize, interrupting power to the control rod drive mechanism clutches, allowing the full length control rods to insert into the core.

For those trips relied upon in the safety analyses, three of the four measurement and trip unit channels can meet the redundancy and testability of GDC 21 in 10 CFR 50, Appendix A (Ref. 1). This LCO requires, however, that four channels be OPERABLE. The fourth channel provides additional flexibility by allowing one channel to be removed from service (trip channel bypassed) for maintenance or testing while still maintaining a minimum two-out-of-three logic.

Since no single failure will prevent a protective system actuation, this arrangement meets the requirements of IEEE Standard 279-1971 (Ref. 3).

Most of the RPS trips are generated by comparing a single measurement to a fixed bistable setpoint. Two trip Functions, Variable High Power Trip and Thermal Margin Low Pressure Trip, make use of more than one measurement to provide a trip.

The required RPS Trip Functions utilize the following input instrumentation:

- Variable High Power Trip (VHPT)

  The VHPT uses Q Power as its input. Q Power is the higher of NI power from the power range NI drawer and primary calorimetric power (ΔT power) based on PCS hot leg and cold leg temperatures. The measurement channels associated with the VHPT are the power range excore channels, and the PCS hot and cold leg temperature channels.

BASES

BACKGROUND (continued)

Measurement Channels

- Variable High Power Trip (VHPT) (continued)

  The Thermal Margin Monitors provide the complex signal processing necessary to calculate the TM/LP trip setpoint, VHPT trip setpoint and trip comparison, and Q Power calculation. On power decreases the VHPT setpoint tracks power levels downward so that it is always within a fixed increment above current power, subject to a minimum value.

  On power increases, the trip setpoint remains fixed unless manually reset, at which point it increases to the new setpoint, a fixed increment above Q Power at the time of reset, subject to a maximum value. Thus, during power escalation, the trip setpoint must be repeatedly reset to avoid a reactor trip.

- High Startup Rate Trip

  The High Startup Rate trip uses the wide range Nuclear Instruments (NIs) to provide an input signal. There are only two wide range NI channels. The wide range channel signal processing electronics are physically mounted in RPS cabinet channels C (NI-1/3) and D (NI-2/4). Separate bistable trip units mounted within the NI-1/3 wide range channel drawer supply High Startup Rate trip signals to RPS channels A and C. Separate bistable trip units mounted within the NI-2/4 wide range channel drawer provide High Startup Rate trip signals to RPS channels B and D.

- Low Primary Coolant Flow Trip

  The Low Primary Coolant Flow Trip utilizes 16 flow measurement channels which monitor the differential pressure across the primary side of the steam generators. Each RPS channel, A, B, C, and D, receives a signal which is the sum of four differential pressure signals. This totalized signal is compared with a setpoint in the RPS Low Flow bistable trip unit for that RPS channel.

BASES

BACKGROUND (continued)

Measurement Channels (continued)

- Low Steam Generator Level Trips

  There are two separate Low Steam Generator Level trips, one for each steam generator. Each Low Steam Generator Level trip monitors four level measurement channels for the associated steam generator, one for each RPS channel.

- Low Steam Generator Pressure Trips

  There are also two separate Low Steam Generator Pressure trips, one for each steam generator. Each Low Steam Generator Pressure trip monitors four pressure measurement channels for the associated steam generator, one for each RPS channel.

- High Pressurizer Pressure Trip

The High Pressurizer Pressure Trip monitors four pressurizer pressure channels, one for each RPS channel.

- Thermal Margin Low Pressure (TM/LP) Trip

  The TM/LP Trip utilizes bistable trip units. Each of these bistable trip units receives a calculated trip setpoint from the Thermal Margin Monitor (TMM) and compares it to the measured pressurizer pressure signal. The TM/LP setpoint is based on Q power (the higher of NI power from the power range NI drawer, or ΔT power, based on PCS hot leg and cold leg temperatures) pressurizer pressure, PCS cold leg temperature, and Axial Shape Index. The TMM provide the complex signal processing necessary to calculate the TM/LP trip setpoint, TM/LP trip comparison signal, and Q Power.

BASES

BACKGROUND (continued)

Measurement Channels (continued)

- Loss of Load Trip

  The Loss of Load trip uses a single pressure switch, 63/AST-2, in the turbine auto stop oil circuit to sense a turbine trip for input to all four RPS auxiliary trip units. The Loss of Load Trip is actuated by turbine auxiliary relays 305L and 305R. Relay 305L provides input to RPS channels A and C; 305R to channels B and D. Relays 305L and 305R are energized on a turbine trip. Their inputs are the same as the inputs to the turbine solenoid trip valve, 20ET.

  If a turbine trip is generated by loss of auto stop oil pressure, auto stop oil pressure switch 63/AST-2 will actuate relays 305L and 305R and generate a reactor trip. If a turbine trip is generated by an input to the solenoid trip valve, relays 305L and 305R, which are wired in parallel, will also be actuated and will generate a reactor trip.

- Containment High Pressure Trip

  The Containment High Pressure Trip is actuated by four pressure switches, one for each RPS channel.

- Zero Power Mode Bypass Automatic Removal

  The Zero Power Bypass allows manually bypassing (i.e., disabling) four reactor trip functions, Low PCS Flow, Low SG A Pressure, Low SG B Pressure, and TM/LP (low PCS pressure), when reactor power (as indicated by the wide range nuclear instrument channels) is below 10^{-4}%. This bypassing is necessary to allow RPS testing and control rod drive mechanism testing when the reactor is shutdown and plant conditions would cause a reactor trip to be present.

  The Zero Power Mode Bypass removal interlock uses the wide range nuclear instruments (NIs) as measurement channels. There are only two wide range NI channels. Separate bistables are provided to actuate the bypass removal for each RPS channel. Bistables in the NI-1/3 channel provide the bypass removal function for RPS channels A and C; bistables in the NI-2/4 channel for RPS channels B and D.

BASES

BACKGROUND (continued)

Several measurement instrument channels provide more than one required function. Those sensors shared for RPS and ESF functions are identified in Table B 3.3.1-1. That table provides a listing of those shared channels and the Specifications which they affect.

RPS Trip Units

Two types of RPS trip units are used in the RPS cabinets; bistable trip units and auxiliary trip units:

> A bistable trip unit receives a measured process signal from its instrument channel and compares it to a setpoint; the trip unit actuates three relays, with contacts in the Matrix Logic channels, when the measured signal is less conservative than the setpoint. They also provide local trip indication and remote annunciation.
>
> An auxiliary trip unit receives a digital input (contacts open or closed); the trip unit actuates three relays, with contacts in the Matrix Logic channels, when the digital input is received. They also provide local trip indication and remote annunciation.

Each RPS channel has four auxiliary trip units and seven bistable trip units.

The contacts from these trip unit relays are arranged into six coincidence matrices, comprising the Matrix Logic. If bistable trip units monitoring the same parameter in at least two channels trip, the Matrix Logic will generate a reactor trip (two-out-of-four logic).

Four of the RPS measurement channels provide contact outputs to the RPS, so the comparison of an analog input to a trip setpoint is not necessary. In these cases, the bistable trip unit is replaced with an auxiliary trip unit. The auxiliary trip units provide contact multiplication so the single input contact opening can provide multiple contact outputs to the coincidence logic as well as trip indication and annunciation.

BASES

BACKGROUND (continued)

RPS Trip Units (continued)

Trips employing auxiliary trip units include the VHPT, which receives contact inputs from the Thermal Margin Monitors; the High Startup Rate trip which employs contact inputs from bistables mounted in the two wide range drawers; the Loss of Load Trip which receives contact inputs from one of two auxiliary relays which are operated by a single switch sensing turbine auto stop oil pressure; and the Containment High Pressure (CHP) trip, which employs containment pressure switch contacts.

There are four RPS trip units, designated as channels A through D, each channel having eleven trip units, one for each RPS Function. Trip unit output relays de-energize when a trip occurs.

All RPS Trip Functions, with the exception of the Loss of Load and CHP trips, generate a pretrip alarm as the trip setpoint is approached.

The Allowable Values are specified for each safety related RPS trip Function which is credited in the safety analysis. Nominal trip setpoints are specified in the plant procedures. The nominal setpoints are selected to ensure plant parameters do not exceed the Allowable Value if the instrument loop is performing as required. The methodology used to determine the nominal trip setpoints is also provided in plant documents. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Each Allowable Value specified is more conservative than the analytical limit determined in the safety analysis in order to account for uncertainties appropriate to the trip Function. These uncertainties are addressed as described in plant documents. A channel is inoperable if its actual setpoint is not within its Allowable Value.

Setpoints in accordance with the Allowable Value will ensure that SLs of Chapter 2.0 are not violated during AOOs and the consequences of DBAs will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed.

Note that in the accompanying LCO 3.3.1, the Allowable Values of Table 3.3.1-1 are the LSSS.

BASES

BACKGROUND (continued)

Reactor Protective System Bypasses

Three different types of trip bypass are utilized in the RPS, Operating Bypass, Zero Power Mode Bypass, and Trip Channel Bypass. The Operating Bypass or Zero Power Mode Bypass prevent the actuation of a trip unit or auxiliary trip unit; the Trip Channel Bypass prevents the trip unit output from affecting the Logic Matrix. A channel which is bypassed, other than as allowed by the Table 3.3.1-1 footnotes, cannot perform its specified safety function and must be considered to be inoperable.

Operating Bypasses

The Operating Bypasses are initiated and removed automatically during startup and shutdown as power level changes. An Operating Bypass prevents the associated RPS auxiliary trip unit from receiving a trip signal from the associated measurement channel. With the bypass in place, neither the pre-trip alarm nor the trip will actuate if the measured parameter exceeds the set point. An annunciator is provided for each Operating Bypass. The RPS trips with Operating Bypasses are:

a. High Startup Rate Trip bypass. The High Startup Rate trip is automatically bypassed when the associated wide range channel indicates below 1E-4% RTP, and when the associated power range excore channel indicates above 13% RTP. These bypasses are automatically removed between 1E-4% RTP and 13% RTP.

b. Loss of Load bypass. The Loss of Load trip is automatically bypassed when the associated power range excore channel indicates below 17% RTP. The bypass is automatically removed when the channel indicates above the set point. The same power range excore channel bistable is used to bypass the High Startup Rate trip and the Loss of Load trip for that RPS channel.

BASES

BACKGROUND (continued)

Operating Bypasses (continued)

Each wide range channel contains two bistables set at 1E-4% RTP, one bistable unit for each associated RPS channel. Each of the two wide range channels affect the Operating Bypasses for two RPS channels; wide range channel NI-1/3 for RPS channels A and C, wide range channel NI-2/4 for RPS channels B and D. Each of the four power range excore channel affects the Operating Bypasses for the associated RPS channel. The power range excore channel bistables associated with the Operating Bypasses are set at a nominal 15%, and are required to actuate between 13% RTP and 17% RTP.

Zero Power Mode (ZPM) Bypass

The ZPM Bypass is used when the plant is shut down and it is desired to raise the control rods for control rod drop testing with PCS flow, pressure or temperature too low for the RPS trips to be reset. ZPM bypasses may be manually initiated and removed when wide range power is below 1E-4% RTP, and are automatically removed if the associated wide range NI indicated power exceeds 1E-4% RTP. A ZPM bypass prevents the RPS trip unit from actuating if the measured parameter exceeds the set point. Operation of the pretrip alarm is unaffected by the zero power mode bypass. An annunciator indicates the presence of any ZPM bypass. The RPS trips with ZPM bypasses are:

a. Low Primary Coolant System Flow.

b. Low Steam Generator Pressure.

c. Thermal Margin/Low Pressure.

The wide range NI channels provide contact closure permissive signals when indicated power is below 1E-4% RTP. The ZPM bypasses may then be manually initiated or removed by actuation of key-lock switches. One key-lock switch located on each RPS cabinet controls the ZPM Bypass for the associated RPS trip channels. The bypass is automatically removed if the associated wide range NI indicated power exceeds 1E-4% RTP. The same wide range NI channel bistables that provide the ZPM Bypass permissive and removal signals also provide the high startup rate trip Operating Bypass actuation and removal.

BASES

BACKGROUND (continued)

Trip Channel Bypass

A Trip Channel Bypass is used when it is desired to physically remove an individual trip unit from the system, or when calibration or servicing of a trip channel could cause an inadvertent trip. A trip Channel Bypass may be manually initiated or removed at any time by actuation of a key-lock switch. A Trip Channel Bypass prevents the trip unit output from affecting the RPS logic matrix. A light above the bypass switch indicates that the trip channel has been bypassed. Each RPS trip unit has an associated trip channel bypass:

The key-lock trip channel bypass switch is located above each trip unit. The key cannot be removed when in the bypass position. Only one key for each trip parameter is provided, therefore the operator can bypass only one channel of a given parameter at a time. During the bypass condition, system logic changes from two-out-of-four to two-out-of-three channels required for trip.

APPLICABLE SAFETY ANALYSES

Each of the analyzed accidents and transients can be detected by one or more RPS Functions. The accident analysis contained in Reference 4 takes credit for most RPS trip Functions. The High Startup Rate and Loss of Load Functions, which are not specifically credited in the accident analysis are part of the NRC approved licensing basis for the plant. The High Startup Rate and Loss of Load trips are purely equipment protective, and their use minimizes the potential for equipment damage.

The specific safety analyses applicable to each protective Function are identified below.

1. Variable High Power Trip (VHPT)

   The VHPT provides reactor core protection against positive reactivity excursions.

   The safety analysis assumes that this trip is OPERABLE to terminate excessive positive reactivity insertions during power operation and while shut down.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

2. High Startup Rate Trip

There are no safety analyses which take credit for functioning of the High Startup Rate Trip. The High Startup Rate trip is used to trip the reactor when excore wide range power indicates an excessive rate of change. The High Startup Rate trip minimizes transients for events such as a continuous control rod withdrawal or a boron dilution event from low power levels. The trip may be operationally bypassed when THERMAL POWER is < 1E-4% RTP, when poor counting statistics may lead to erroneous indication. It may also be operationally bypassed at > 13% RTP, where moderator temperature coefficient and fuel temperature coefficient make high rate of change of power unlikely.

There are only two wide range drawers, with each supplying contact input to auxiliary trip units in two RPS channels.

3. Low Primary Coolant System Flow Trip

The Low PCS Flow trip provides DNB protection during events which suddenly reduce the PCS flow rate during power operation, such as loss of power to, or seizure of, a primary coolant pump.

Flow in each of the four PCS loops is determined from pressure drop from inlet to outlet of the SGs. The total PCS flow is determined, for the RPS flow channels, by summing the loop pressure drops across the SGs and correlating this pressure sum with the sum of SG differential pressures which exist at 100% flow (four pump operation at full power T_{ave}). Full PCS flow is that flow which exists at RTP, at full power T_{ave}, with four pumps operating.

4, 5. Low Steam Generator Level Trip

The Low Steam Generator Level trips are provided to trip the reactor in the event of excessive steam demand (to prevent overcooling the PCS) and loss of feedwater events (to prevent overpressurization of the PCS).

The Allowable Value assures that there will be sufficient water inventory in the SG at the time of trip to allow a safe and orderly plant shutdown and to prevent SG dryout assuming minimum AFW capacity.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

4, 5. Low Steam Generator Level Trip (continued)

Each SG level is sensed by measuring the differential pressure in the upper portion of the downcomer annulus in the SG. These trips share four level sensing channels on each SG with the AFW actuation signal.

6, 7. Low Steam Generator Pressure Trip

The Low Steam Generator Pressure trip provides protection against an excessive rate of heat extraction from the steam generators, which would result in a rapid uncontrolled cooldown of the PCS. This trip provides a mitigation function in the event of an MSLB.

The Low SG Pressure channels are shared with the Low SG Pressure signals which isolate the steam and feedwater lines.

8. High Pressurizer Pressure Trip

The High Pressurizer Pressure trip, in conjunction with pressurizer safety valves and Main Steam Safety Valves (MSSVs), provides protection against overpressure conditions in the PCS when at operating temperature. The safety analyses assume the High Pressurizer Pressure trip is OPERABLE during accidents and transients which suddenly reduce PCS cooling (e.g., Loss of Load, Main Steam Isolation Valve (MSIV) closure, etc.) or which suddenly increase reactor power (e.g., rod ejection accident).

The High Pressurizer Pressure trip shares four safety grade instrument channels with the TM/LP trip, Anticipated Transient Without Scram (ATWS) and PORV circuits, and the Pressurizer Low Pressure Safety Injection Signal.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

9. Thermal Margin/Low Pressure (TM/LP) Trip

The TM/LP trip is provided to prevent reactor operation when the DNBR is insufficient. The TM/LP trip protects against slow reactivity or temperature increases, and against pressure decreases.

The trip is initiated whenever the PCS pressure signal drops below a minimum value (P_{min}) or a computed value (P_{var}) as described below, whichever is higher.

The TM/LP trip uses Q Power, ASI, pressurizer pressure, and cold leg temperature (T_c) as inputs.

Q Power is the higher of core THERMAL POWER (ΔT Power) or nuclear power. The ΔT power uses hot leg and cold leg RTDs as inputs. Nuclear power uses the power range excore channels as inputs. Both the ΔT and excore power signals have provisions for calibration by calorimetric calculations.

The ASI is calculated from the upper and lower power range excore detector signals, as explained in Section 1.1, "Definitions." The signal is corrected for the difference between the flux at the core periphery and the flux at the detectors.

The T_c value is the higher of the two cold leg signals.

The Low Pressurizer Pressure trip limit (P_{var})is calculated using the equations given in Table 3.3.1-2.

The calculated limit (P_{var}) is then compared to a fixed Low Pressurizer Pressure trip limit (P_{min}). The auctioneered highest of these signals becomes the trip limit (P_{trip}). P_{trip} is compared to the measured PCS pressure and a trip signal is generated when the measured pressure for that channel is less than or equal to P_{trip}. A pre-trip alarm is also generated when P is less than or equal to the pre-trip setting, $P_{trip} + \Delta P$.

The TM/LP trip setpoint is a complex function of these inputs and represents a minimum acceptable PCS pressure for the existing temperature and power conditions. It is compared to actual PCS pressure in the TM/LP trip unit.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

10. Loss of Load Trip

There are no safety analyses which take credit for functioning of the Loss of Load Trip.

The Loss of Load trip is provided to prevent lifting the pressurizer and main steam safety valves in the event of a turbine generator trip while at power. The trip is equipment protective. The safety analyses do not assume that this trip functions during any accident or transient. The Loss of Load trip uses a single pressure switch in the turbine auto stop oil circuit to sense a turbine trip for input to all four RPS auxiliary trip units.

11. Containment High Pressure Trip

The Containment High Pressure trip provides a reactor trip in the event of a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB). The Containment High Pressure trip shares sensors with the Containment High Pressure sensing logic for Safety Injection, Containment Isolation, and Containment Spray. Each of these sensors has a single bellows which actuates two microswitches. One microswitch on each of four sensors provides an input to the RPS.

12. Zero Power Mode Bypass Removal

The only RPS bypass considered in the safety analyses is the Zero Power Mode (ZPM) Bypass. The ZPM Bypass is used when the plant is shut down and it is desired to raise the control rods for control rod drop testing with PCS flow or temperature too low for the RPS Low PCS Flow, Low SG Pressure, or Thermal Margin/Low Pressure trips to be reset. ZPM bypasses are automatically removed if the wide range NI indicated power exceeds 1E-4% RTP.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

12. Zero Power Mode Bypass Removal (continued)

The safety analyses take credit for automatic removal of the ZPM Bypass if reactor criticality due to a Continuous Control Rod Bank Withdrawal should occur with the affected trips bypassed and PCS flow, pressure, or temperature below the values at which the RPS could be reset. The ZPM Bypass would effectively be removed when the first wide range NI channel indication reached 1E-4% RTP. With the ZPM Bypass for two RPS channels removed, the RPS would trip on one of the un-bypassed trips. This would prevent the reactor reaching an excessive power level.

If a reactor criticality due to a Continuous Control Rod Bank Withdrawal should occur when PCS flow, steam generator pressure, and PCS pressure (TM/LP) were above their trip setpoints, a trip would terminate the event when power increased to the minimum setting (nominally 30%) of the Variable High Power Trip. In this case, the monitored parameters are at or near their normal operational values, and a trip initiated at 30% RTP provides adequate protection.

The RPS design also includes automatic removal of the Operating Bypasses for the High Startup Rate and Loss of Load trips. The safety analyses do not assume functioning of either these trips or the automatic removal of their bypasses.

The RPS instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2).

LCO

The LCO requires all instrumentation performing an RPS Function to be OPERABLE. Failure of the trip unit (including its output relays), any required portion of the associated instrument channel, or both, renders the affected channel(s) inoperable and reduces the reliability of the affected Functions. Failure of an automatic ZPM bypass removal channel may also impact the associated instrument channel(s) and reduce the reliability of the affected Functions.

BASES

LCO (continued)

Actions allow Trip Channel Bypass of individual channels, but the bypassed channel must be considered to be inoperable. The bypass key used to bypass a single channel cannot be simultaneously used to bypass that same parameter in other channels. This interlock prevents operation with more than one channel of the same Function trip channel bypassed. The plant is normally restricted to 7 days in a trip channel bypass, or otherwise inoperable condition before either restoring the Function to four channel operation (two-out-of-four logic) or placing the channel in trip (one-out-of-three logic).

The Allowable Values are specified for each safety related RPS trip Function which is credited in the safety analysis. Nominal trip setpoints are specified in the plant procedures. The nominal setpoints are selected to ensure plant parameters do not exceed the Allowable Value if the instrument loop is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Each Allowable Value specified is more conservative than the analytical limit determined in the safety analysis in order to account for uncertainties appropriate to the trip Function. These uncertainties are addressed as described in plant documents. Neither Allowable Values nor setpoints are specified for the non-safety related RPS Trip Functions, since no safety analysis assumptions would be violated if they are not set at a particular value.

The following Bases for each trip Function identify the above RPS trip Function criteria items that are applicable to establish the trip Function OPERABILITY.

1. <u>Variable High Power Trip (VHPT)</u>

   This LCO requires all four channels of the VHPT Function to be OPERABLE.

   The Allowable Value is high enough to provide an operating envelope that prevents unnecessary VHPT trips during normal plant operations. The Allowable Value is low enough for the system to function adequately during reactivity addition events.

BASES

LCO (continued)

1. Variable High Power Trip (VHPT) (continued)

The VHPT is designed to limit maximum reactor power to its maximum design and to terminate power excursions initiating at lower powers without power reaching this full power limit. During plant startup, the VHPT trip setpoint is initially at its minimum value, $\leq$ 30%. Below 30% RTP, the VHPT setpoint is not required to "track" with Q Power, i.e., be adjusted to within 15% RTP. It remains fixed until manually reset, at which point it increases to $\leq$ 15% above existing Q Power.

The maximum allowable setting of the VHPT is 106.5% RTP. Adding to this the possible variation in trip setpoint due to calibration and instrument error, the maximum actual steady state power at which a trip would be actuated is 115%, which is the value assumed in the safety analysis.

2. High Startup Rate Trip

This LCO requires four channels of High Startup Rate Trip Function to be OPERABLE in MODES 1 and 2.

The High Startup Rate trip serves as a backup to the administratively enforced startup rate limit. The Function is not credited in the accident analyses; therefore, no Allowable Value for the trip or operating bypass Functions is derived from analytical limits and none is specified.

The four channels of the High Startup Rate trip are derived from two wide range NI signal processing drawers. Thus, a failure in one wide range channel could render two RPS channels inoperable. It is acceptable to continue operation in this condition because the High Startup Rate trip is not credited in any safety analyses.

The requirement for this trip Function is modified by a footnote, which allows the High Startup Rate trip to be bypassed when the wide range NI indicates below 10E-4% or when THERMAL POWER is above 13% RTP. If a High Startup Rate trip is bypassed when power is between these limits, it must be considered to be inoperable.

BASES

LCO (continued)

3. Low Primary Coolant System Flow Trip

This LCO requires four channels of Low PCS Flow Trip Function to be OPERABLE.

This trip is set high enough to maintain fuel integrity during a loss of flow condition. The setting is low enough to allow for normal operating fluctuations from offsite power.

The Low PCS Flow trip setpoint of 95% of full PCS flow insures that the reactor cannot operate when the flow rate is less than 93% of the nominal value considering instrument errors. Full PCS flow is that flow which exists at RTP, at full power T_{ave}, with four pumps operating.

The requirement for this trip Function is modified by a footnote, which allows use of the ZPM bypass when wide range power is below 1E-4% RTP. That bypass is automatically removed when the associated wide range channel indicates 1E-4% RTP. If a trip channel is bypassed when power is above 1E-4% RTP, it must be considered to be inoperable.

4, 5. Low Steam Generator Level Trip

This LCO requires four channels of Low Steam Generator Level Trip Function per steam generator to be OPERABLE.

The 25.9% Allowable Value assures that there is an adequate water inventory in the steam generators when the reactor is critical and is based upon narrow range instrumentation. The 25.9% indicated level corresponds to the location of the feed ring.

6, 7. Low Steam Generator Pressure Trip

This LCO requires four channels of Low Steam Generator Pressure Trip Function per steam generator to be OPERABLE.

The Allowable Value of 500 psia is sufficiently below the full load operating value for steam pressure so as not to interfere with normal plant operation, but still high enough to provide the required protection in the event of excessive steam demand. Since excessive steam demand causes the PCS to cool down, resulting in positive reactivity addition to the core, a reactor trip is required to offset that effect.

BASES

LCO
(continued)

8. High Pressurizer Pressure Trip

This LCO requires four channels of High Pressurizer Pressure Trip Function to be OPERABLE.

The Allowable Value is set high enough to allow for pressure increases in the PCS during normal operation (i.e., plant transients) not indicative of an abnormal condition. The setting is below the lift setpoint of the pressurizer safety valves and low enough to initiate a reactor trip when an abnormal condition is indicated.

9. Thermal Margin/Low Pressure (TM/LP) Trip

This LCO requires four channels of TM/LP Trip Function to be OPERABLE.

The TM/LP trip setpoints are derived from the core thermal limits through application of appropriate allowances for measurement uncertainties and processing errors. The allowances specifically account for instrument drift in both power and inlet temperatures, calorimetric power measurement, inlet temperature measurement, and primary system pressure measurement.

Other uncertainties including allowances for assembly power tilt, fuel pellet manufacturing tolerances, core flow measurement uncertainty and core bypass flow, inlet temperature measurement time delays, and ASI measurement, are included in the development of the TM/LP trip setpoint used in the accident analysis.

The requirement for this trip Function is modified by a footnote, which allows use of the ZPM bypass when wide range power is below 1E-4% RTP. That bypass is automatically removed when the associated wide range channel indicates 1E-4% RTP. If a trip channel is bypassed when power is above 1E-4% RTP, it must be considered to be inoperable.

BASES

LCO (continued)

10. Loss of Load Trip

The LCO requires four Loss of Load Trip Function channels to be OPERABLE in MODE 1 with THERMAL POWER $\geq$ 17% RTP.

The Loss of Load trip may be bypassed or be inoperable with THERMAL POWER < 17% RTP, since it is no longer needed to prevent lifting of the pressurizer safety valves or steam generator safety valves in the event of a Loss of Load. Loss of Load Trip unit must be considered inoperable if it is bypassed when THERMAL POWER is above 17% RTP.

This LCO requires four RPS Loss of Load auxiliary trip units, relays 305L and 305R, and pressure switch 63/AST-2 to be OPERABLE. With those components OPERABLE, a turbine trip will generate a reactor trip. The LCO does not require the various turbine trips, themselves, to be OPERABLE.

The Nuclear Steam Supply System and Steam Dump System are capable of accommodating the Loss of Load without requiring the use of the above equipment.

The Loss of Load Trip Function is not credited in the accident analysis; therefore, an Allowable Value for the trip cannot be derived from analytical limits, and is not specified.

11. Containment High Pressure Trip

This LCO requires four channels of Containment High Pressure Trip Function to be OPERABLE.

The Allowable Value is high enough to allow for small pressure increases in containment expected during normal operation (i.e., plant heatup) that are not indicative of an abnormal condition. The setting is low enough to initiate a reactor trip to prevent containment pressure from exceeding design pressure following a DBA and ensures the reactor is shutdown before initiation of safety injection and containment spray.

BASES

LCO
(continued)

12. ZPM Bypass

The LCO requires that four channels of automatic Zero Power Mode (ZPM) Bypass removal instrumentation be OPERABLE. Each channel of automatic ZPM Bypass removal includes a shared wide range NI channel, an actuating bistable in the wide range drawer, and a relay in the associated RPS cabinet. Wide Range NI channel 1/3 is shared between ZPM Bypass removal channels A and C; Wide Range NI channel 2/4, between ZPM Bypass removal channels B and D. An operable bypass removal channel must be capable of automatically removing the capability to bypass the affected RPS trip channels with the ZPM Bypass key switch at the proper setpoint.

APPLICABILITY

This LCO requires all safety related trip functions to be OPERABLE in accordance with Table 3.3.1-1.

Those RPS trip Functions which are assumed in the safety analyses (all except High Startup Rate and Loss of Load), are required to be operable in MODES 1 and 2, and in MODES 3, 4, and 5 with more than one full-length control rod capable of being withdrawn and PCS boron concentration less than REFUELING BORON CONCENTRATION.

These trip Functions are not required while in MODES 3, 4, or 5, if PCS boron concentration is at REFUELING BORON CONCENTRATION, or when no more than one full-length control rod is capable of being withdrawn, because the RPS Function is already fulfilled. REFUELING BORON CONCENTRATION provides sufficient negative reactivity to assure the reactor remains subcritical regardless of control rod position, and the safety analyses assume that the highest worth withdrawn full-length control rod will fail to insert on a trip. Therefore, under these conditions, the safety analyses assumptions will be met without the RPS trip Function.

The High Startup Rate Trip Function is required to be OPERABLE in MODES 1 and 2, but may be bypassed when the associated wide range NI channel indicates below 1E-4% power, when poor counting statistics may lead to erroneous indication. In MODES 3, 4, 5, and 6, the High Startup Rate trip is not required to be OPERABLE. Wide range channels are required to be OPERABLE in MODES 3, 4, and 5, by LCO 3.3.9, "Neutron Flux Monitoring Channels," and in MODE 6, by LCO 3.9.2, "Nuclear Instrumentation."

BASES

APPLICABILITY (continued)

The High Startup Rate Trip Function is required to be OPERABLE in MODES 1 and 2, but may be bypassed when the associated wide range NI channel indicates below 1E-4% power, when poor counting statistics may lead to erroneous indication. In MODES 3, 4, 5, and 6, the High Startup Rate trip is not required to be OPERABLE. Wide range channels are required to be OPERABLE in MODES 3, 4, and 5, by LCO 3.3.9, "Neutron Flux Monitoring Channels," and in MODE 6, by LCO 3.9.2, "Nuclear Instrumentation."

The Loss of Load trip is required to be OPERABLE with THERMAL POWER at or above 17% RTP. Below 17% RTP, the ADVs are capable of relieving the pressure due to a Loss of Load event without challenging other overpressure protection.

The trips are designed to take the reactor subcritical, maintaining the SLs during AOOs and assisting the ESF in providing acceptable consequences during accidents.

ACTIONS

The most common causes of channel inoperability are outright failure of loop components or drift of those loop components which is sufficient to exceed the tolerance provided in the plant setpoint analysis. Loop component failures are typically identified by the actuation of alarms due to the channel failing to the "safe" condition, during CHANNEL CHECKS (when the instrument is compared to the redundant channels), or during the CHANNEL FUNCTIONAL TEST (when an automatic component might not respond properly). Typically, the drift of the loop components is found to be small and results in a delay of actuation rather than a total loss of function. Excessive loop component drift would, most likely, be identified during a CHANNEL CHECK (when the instrument is compared to the redundant channels) or during a CHANNEL CALIBRATION (when instrument loop components are checked against reference standards).

In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or RPS bistable trip unit is found inoperable, all affected Functions provided by that channel must be declared inoperable, and the plant must enter the Condition for the particular protection Functions affected.

BASES

ACTIONS (continued)

When the number of inoperable channels in a trip Function exceeds that specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 is immediately entered if applicable in the current MODE of operation.

A Note has been added to the ACTIONS to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function. The Completion Times of each inoperable Function will be tracked separately for each Function, starting from the time the Condition was entered.

A.1

Condition A applies to the failure of a single channel in any required RPS Function, except High Startup Rate, Loss of Load, or ZPM Bypass Removal. (Condition A is modified by a Note stating that this Condition does not apply to the High Startup Rate, Loss of Load, or ZPM Bypass Removal Functions. The failure of one channel of those Functions is addressed by Conditions B, C, or D.)

If one RPS bistable trip unit or associated instrument channel is inoperable, operation is allowed to continue. Since the trip unit and associated instrument channel combine to perform the trip function, this Condition is also appropriate if both the trip unit and the associated instrument channel are inoperable. Though not required, the inoperable channel may be bypassed. The provision of four trip channels allows one channel to be bypassed (removed from service) during operations, placing the RPS in two-out-of-three coincidence logic. The failed channel must be restored to OPERABLE status or placed in trip within 7 days.

Required Action A.1 places the Function in a one-out-of-three configuration. In this configuration, common cause failure of dependent channels cannot prevent trip.

The Completion Time of 7 days is based on operating experience, which has demonstrated that a random failure of a second channel occurring during the 7 day period is a low probability event.

BASES

ACTIONS (continued)

A.1 (continued)

The Completion Time of 7 days is based on operating experience, which has demonstrated that a random failure of a second channel occurring during the 7 day period is a low probability event.

B.1

Condition B applies to the failure of a single High Startup Rate trip unit or associated instrument channel.

If one trip unit or associated instrument channel fails, it must be restored to OPERABLE status prior to entering MODE 2 from MODE 3. A shutdown provides the appropriate opportunity to repair the trip function and conduct the necessary testing. The Completion Time is based on the fact that the safety analyses take no credit for the functioning of this trip.

C.1

Condition C applies to the failure of a single Loss of Load or associated instrument channel.

If one trip unit or associated instrument channel fails, it must be restored to OPERABLE status prior to THERMAL POWER ≥ 17% RTP following a shutdown. If the plant is shutdown at the time the channel becomes inoperable, then the failed channel must be restored to OPERABLE status prior to THERMAL POWER ≥ 17% RTP. For this Completion Time, "following a shutdown" means this Required Action does not have to be completed until prior to THERMAL POWER ≥ 17% RTP for the first time after the plant has been in MODE 3 following entry into the Condition. The Completion Time trip assures that the plant will not be restarted with an inoperable Loss of Load trip channel.

BASES

ACTIONS (continued)

D.1 and D.2

Condition D applies when one or more automatic ZPM Bypass removal channels are inoperable. If the ZPM Bypass removal channel cannot be restored to OPERABLE status, the affected ZPM Bypasses must be immediately removed, or the bypassed RPS trip Function channels must be immediately declared to be inoperable. Unless additional circuit failures exist, the ZPM Bypass may be removed by placing the associated "Zero Power Mode Bypass" key operated switch in the normal position.

A trip channel which is actually bypassed, other than as allowed by the Table 3.3.1-1 footnotes, cannot perform its specified safety function and must immediately be declared to be inoperable.

E.1 and E.2

Condition E applies to the failure of two channels in any RPS Function, except ZPM Bypass Removal Function. (The failure of ZPM Bypass Removal Functions is addressed by Condition D.).

Condition E is modified by a Note stating that this Condition does not apply to the ZPM Bypass Removal Function.

The Required Actions are modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODES even though two channels are inoperable, with one channel tripped. MODE changes in this configuration are allowed because two trip channels for the affected function remain OPERABLE. A trip occurring in either or both of those channels would cause a reactor trip.

In this configuration, the protection system is in a one-out-of-two logic, and the probability of a common cause failure affecting both of the OPERABLE channels during the 7 days permitted is remote.

Required Action E.1 provides for placing one inoperable channel in trip within the Completion Time of 1 hour. Though not required, the other inoperable channel may be (trip channel) bypassed.

BASES

ACTIONS (continued)

E.1 and E.2 (continued)

This Completion Time is sufficient to allow the operator to take all appropriate actions for the failed channels while ensuring that the risk involved in operating with the failed channels is acceptable. With one channel of protective instrumentation bypassed or inoperable in an untripped condition, the RPS is in a two-out-of-three logic for that function; but with another channel failed, the RPS may be operating in a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, one of the inoperable channels is placed in trip. This places the RPS in a one-out-of-two for that function logic. If any of the other unbypassed channels for that function receives a trip signal, the reactor will trip.

Action E.2 is modified by a Note stating that this Action does not apply to (is not required for) the High Startup Rate and Loss of Load Functions.

One channel is required to be restored to OPERABLE status within 7 days for reasons similar to those stated under Condition A. After one channel is restored to OPERABLE status, the provisions of Condition A still apply to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action E.2 must be placed in trip if more than 7 days have elapsed since the initial channel failure.

F.1

The power range excore channels are used to generate the internal ASI signal used as an input to the TM/LP trip. They also provide input to the Thermal Margin Monitors for determination of the Q Power input for the TM/LP trip and the VHPT. If two power range excore channels cannot be restored to OPERABLE status, power is restricted or reduced during subsequent operations because of increased uncertainty associated with inoperable power range excore channels which provide input to those trips.

The Completion Time of 2 hours is adequate to reduce power in an orderly manner without challenging plant systems.

BASES

ACTIONS (continued)

G.1, G.2.1, and G.2.2

Condition G is entered when the Required Action and associated Completion Time of Condition A, B, C, D, E, or F are not met, or if the control room ambient air temperature exceeds 90°F.

If the control room ambient air temperature exceeds 90°F, all Thermal Margin Monitor channels are rendered inoperable because their operating temperature limit is exceeded. In this condition, or if the Required Actions and associated Completion Times are not met, the reactor must be placed in a condition in which the LCO does not apply. To accomplish this, the plant must be placed in MODE 3, with no more than one full-length control rod capable of being withdrawn or with the PCS boron concentration at REFUELING BORON CONCENTRATION in 6 hours.

The Completion Time is reasonable, based on operating experience, for placing the plant in MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The Completion Time is also reasonable to ensure that no more than one full-length control rod is capable of being withdrawn or that the PCS boron concentration is at REFUELING BORON CONCENTRATION.

SURVEILLANCE REQUIREMENTS

The SRs for any particular RPS Function are found in the SR column of Table 3.3.1-1 for that Function. Most Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION.

SR 3.3.1.1

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1 (continued)

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

The Containment High Pressure and Loss of Load channels are pressure switch actuated. As such, they have no associated control room indicator and do not require a CHANNEL CHECK.

The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. 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 3.3.1.2

This SR verifies that the control room ambient air temperature is within the environmental qualification temperature limits for the most restrictive RPS components, which are the Thermal Margin Monitors. These monitors provide input to both the VHPT Function and the TM/LP Trip Function. The 12 hour Frequency is reasonable based on engineering judgement and plant operating experience.

SR 3.3.1.3

A daily calibration (heat balance) is performed when THERMAL POWER is ≥ 15%. The daily calibration consists of adjusting the "nuclear power calibrate" potentiometers to agree with the calorimetric calculation if the absolute difference is ≥ 1.5%. Nuclear power is adjusted via a potentiometer, or THERMAL POWER is adjusted via a Thermal Margin Monitor bias number, as necessary, in accordance with the daily calibration (heat balance) procedure. Performance of the daily calibration ensures that the two inputs to the Q power measurement are indicating accurately with respect to the much more accurate secondary calorimetric calculation.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.3 (continued)

The Frequency of 24 hours is based on plant operating experience and takes into account indications and alarms located in the control room to detect deviations in channel outputs.

The Frequency is modified by a Note indicating this Surveillance must be performed within 12 hours after THERMAL POWER is ≥ 15% RTP. The secondary calorimetric is inaccurate at lower power levels. The 12 hours allows time requirements for plant stabilization, data taking, and instrument calibration.

SR 3.3.1.4

It is necessary to calibrate the power range excore channel upper and lower subchannel amplifiers such that the measured ASI reflects the true core power distribution as determined by the incore detectors. ASI is utilized as an input to the TM/LP trip function where it is used to ensure that the measured axial power profiles are bounded by the axial power profiles used in the development of the T_{inlet} limitation of LCO 3.4.1. An adjustment of the excore channel is necessary only if reactor power is greater than 25% RTP and individual excore channel ASI differs from AXIAL OFFSET, as measured by the incores, outside the bounds of the following table:

Allowed Reactor Power	Group 4 Rods≥128" withdrawn	Group 4 Rods <128" withdrawn
≤100%	-0.020 ≤ (AO-ASI) ≤ 0.020	-0.040 ≤ (AO-ASI) ≤ 0.040
< 95	-0.033 ≤ (AO-ASI) ≤ 0.020	-0.053 ≤ (AO-ASI) ≤ 0.040
< 90	-0.046 ≤ (AO-ASI) ≤ 0.020	-0.066 ≤ (AO-ASI) ≤ 0.040
< 85	-0.060 ≤ (AO-ASI) ≤ 0.020	-0.080 ≤ (AO-ASI) ≤ 0.040
< 80	-0.133 ≤ (AO-ASI) ≤ 0.080	-0.153 ≤ (AO-ASI) ≤ 0.100
< 75	-0.146 ≤ (AO-ASI) ≤ 0.080	-0.166 ≤ (AO-ASI) ≤ 0.100
< 70	-0.159 ≤ (AO-ASI) ≤ 0.080	-0.179 ≤ (AO-ASI) ≤ 0.100
< 65	-0.163 ≤ (AO-ASI) ≤ 0.080	-0.183 ≤ (AO-ASI) ≤ 0.100
< 60	-0.168 ≤ (AO-ASI) ≤ 0.080	-0.183 ≤ (AO-ASI) ≤ 0.100
< 55	-0.172 ≤ (AO-ASI) ≤ 0.080	-0.183 ≤ (AO-ASI) ≤ 0.100
< 50	-0.176 ≤ (AO-ASI) ≤ 0.080	-0.183 ≤ (AO-ASI) ≤ 0.100
< 45	-0.181 ≤ (AO-ASI) ≤ 0.080	-0.183 ≤ (AO-ASI) ≤ 0.100
< 40	-0.183 ≤ (AO-ASI) ≤ 0.080	-0.183 ≤ (AO-ASI) ≤ 0.100
< 35	-0.183 ≤ (AO-ASI) ≤ 0.080	-0.183 ≤ (AO-ASI) ≤ 0.100
< 30	-0.183 ≤ (AO-ASI) ≤ 0.080	-0.183 ≤ (AO-ASI) ≤ 0.100
< 25	Below 25% RTP any AO/ASI difference is acceptable	

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.4 (continued)

Below 25% RTP any difference between ASI and AXIAL OFFSET is acceptable. A Note indicates the Surveillance is not required to have been performed until 12 hours after THERMAL POWER is ≥ 25% RTP. Uncertainties in the excore and incore measurement process make it impractical to calibrate when THERMAL POWER is < 25% RTP. The 12 hours allows time for plant stabilization, data taking, and instrument calibration.

The 31 day Frequency is adequate, based on operating experience of the excore linear amplifiers and the slow burnup of the detectors. 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. Slow changes in neutron flux during the fuel cycle can also be detected at this Frequency.

SR 3.3.1.5

A CHANNEL FUNCTIONAL TEST is performed on each RPS instrument channel, except Loss of Load and High Startup Rate, every 92 days to ensure the entire channel will perform its intended function when needed. For the TM/LP Function, the constants associated with the Thermal Margin Monitors must be verified to be within tolerances.

A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

Any setpoint adjustment must be consistent with the assumptions of the current setpoint analysis.

The Frequency of 92 days is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 5).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.6

A calibration check of the power range excore channels using the internal test circuitry is required every 92 days. This SR uses an internally generated test signal to check that the 0% and 50% levels read within limits for both the upper and lower detector, both on the analog meter and on the TMM screen. This check verifies that neither the zero point nor the amplifier gain adjustment have undergone excessive drift since the previous complete CHANNEL CALIBRATION.

The Frequency of 92 days is acceptable, based on plant operating experience, and takes into account indications and alarms available to the operator in the control room.

SR 3.3.1.7

A CHANNEL FUNCTIONAL TEST on the Loss of Load and High Startup Rate channels is performed prior to a reactor startup to ensure the entire channel will perform its intended function.

A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

The High Startup Rate trip is actuated by either of the Wide Range Nuclear Instrument Startup Rate channels. NI-1/3 sends a trip signal to RPS channels A and C; NI-2/4 to channels B and D. Since each High Startup Rate channel would cause a trip on two RPS channels, the High Startup Rate trip is not tested when the reactor is critical.

The four Loss of Load Trip channels are all actuated by a single pressure switch monitoring turbine auto stop oil pressure which is not tested when the reactor is critical. Operating experience has shown that these components usually pass the Surveillance when performed at a Frequency of once per 7 days prior to each reactor startup.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.8

SR 3.3.1.8 is the performance of a CHANNEL CALIBRATION every 18 months.

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor (except neutron detectors). The Surveillance 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 between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be consistent with the setpoint analysis.

The bistable setpoints must be found to trip within the Allowable Values specified in the LCO and left set consistent with the assumptions of the setpoint analysis. The Variable High Power Trip setpoint shall be verified to reset properly at several indicated power levels during (simulated) power increases and power decreases.

The as-found and as-left values must also be recorded and reviewed for consistency with the assumptions of the setpoint analysis.

As part of the CHANNEL CALIBRATION of the wide range Nuclear Instrumentation, automatic removal of the ZPM Bypass for the Low PCS Flow, TM/LP must be verified to assure that these trips are available when required.

The Frequency is based upon the assumption of an 18 month calibration interval for the determination of the magnitude of equipment drift.

This SR is modified by a Note which states that it is not necessary to calibrate neutron detectors because they are passive devices with minimal drift and because of the difficulty of simulating a meaningful signal. Slow changes in power range excore neutron detector sensitivity are compensated for by performing the daily calorimetric calibration (SR 3.3.1.3) and the monthly calibration using the incore detectors (SR 3.3.1.4). Sudden changes in detector performance would be noted during the required CHANNEL CHECKS (SR 3.3.1.1).

BASES

REFERENCES	1.	10 CFR 50, Appendix A, GDC 21
	2.	10 CFR 100
	3.	IEEE Standard 279-1971, April 5, 1972
	4.	FSAR, Chapter 14
	5.	CEN-327, June 2, 1986, including Supplement 1, March 3, 1989

Table B 3.3.1-1 (page 1 of 1)
Instruments Affecting Multiple Specifications

REQUIRED INSTRUMENT CHANNELS	AFFECTED SPECIFICATIONS
Source Range NI-1/3 & 2/4, Count Rate Signal	3.3.9 3.9.2
Source Range NI-1/3, Count Rate Indication @ C-150 Panel	3.3.8 #1
Wide Range NI-1/3 & 2/4, Flux Level 10^{-4} Bypass	3.3.1 #3,6,7,9,&12
Wide Range NI-1/3 & 2/4, Startup Rate	3.3.1 #2
Wide Range NI-1/3 & 2/4, Flux Level Indication	3.3.7 #3 3.3.9
Power Range NI-5, 6, 7, & 8, Tq	3.2.1 3.2.3
Power Range NI-5, 6, 7, & 8, Q Power	3.3.1 #1 & 9
Power Range NI-5, 6, 7, & 8, ASI	3.3.1 #9 3.2.1 3.2.4
Power Range NI-5, 6, 7, & 8, Loss of Load/High Startup Rate Bypass	3.3.1 #2 & 10
PCS TC TT-0112 & 0122 CC & CD, Temperature Signal (SMM)	3.3.7 #5
PCS TC TT-0112 & 0122 CA, CB, CC, & CD, Temperature Signal (Q Power & TMM)	3.3.1 #1 & 9 3.4.1.b
PCS TC TT-0112CA & 0122CB, Temperature Signal (LTOP)	3.4.12.b.1
PCS TC TT-0112CC & 0122CD (PTR-0112 & 0122) Temperature Indication	3.3.7 #2
PCS TC TT-0112CA, Temperature Signal (SPI ΔT Power for PDIL Alarm Circuit)	3.1.6
PCS TC TT-0122CB, Temperature Signal (PIP ΔT Power for PDIL Alarm Circuit)	3.1.6
PCS TH TT-0112 & 0122 HC & HD, Temperature Signal (SMM)	3.3.7 #5
PCS TH TT-0112HC & 0122HD (PTR-0112 & 0122) Temperature Indication	3.3.7 #1
PCS TH TT-0112 & 0122 HA, HB, HC, & HD, Temperature Signal (Q Power)	3.3.1 #1 & 9
PCS TH TT-0112HA, Temperature Signal (SPI ΔT Power for PDIL Alarm Circuit)	3.1.6
PCS TH TT-0122HB, Temperature Signal (PIP ΔT Power for PDIL Alarm Circuit)	3.1.6
Thermal Margin Monitor PY-0102A, B, C, & D	3.3.1 #1 & 9
Pressurizer Pressure PT-0105A & B, Pressure Signal (WR Indication & LTOP)	3.3.7 #5 3.3.4.12.b.1
Pressurizer Pressure PT-0102A, B, C, & D, Pressure Signal (RPS & SIS)	3.3.1 #8 & 9 3.3.3 #1.a&7a
Pressurizer Pressure PT-0104A & B, Pressure Signal (LTOP & SDC Interlock)	3.4.12.b.1 3.4.14
Pressurizer Pressure PI-0110, Pressure Indication @ C-150 Panel	3.3.8 #2
SG Level LT-0751 & 0752 A, B, C, & D, Level Signal (RPS & AFAS)	3.3.1 #4 & 5 3.3.3 #4.a & 4.b
SG Level LI-0757C & 0758C, Wide Range Level Indication @ C-150 Panel	3.3.8 #10 & 11
SG Level LI-0757 & 0758 A & B, Wide Range Level Indication	3.3.7 #11 & 12
SG Pressure PT-0751 & 0752 A, B, C, & D, Pressure Signal (RPS & SG Isolation)	3.3.1 #6 & 7 3.3.3 #2a, 2b,7b,7c
SG Pressure PIC-0751 & 0752 A, B, C, & D, Pressure Indication	3.3.7 #13 & 14
SG Pressure PI-0751E & 0752E, Pressure Indication @ C-150 Panel	3.3.8 #8 & 9
Containment Pressure PS-1801, 1802, 1803, & 1804, Switch Output (RPS)	3.3.1 #11
Containment Pressure PS-1801, 1802A, 1803, & 1804A, Switch Output (ESF)	3.3.3 #5.a
Containment Pressure PS-1801A, 1802, 1803A, & 1804, Switch Output (ESF)	3.3.3 #5.b

Note: The information provided in this table is intended for use as an aid to distinguish those instrument channels which provide more than one required function and to describe which specifications they affect. The information in this table should not be taken as inclusive for all instruments nor affected specifications.

B 3.3 INSTRUMENTATION

B 3.3.2 Reactor Protective System (RPS) Logic and Trip Initiation

BASES

BACKGROUND

The RPS initiates a reactor trip to protect against violating the acceptable fuel design limits and reactor coolant pressure boundary integrity during Anticipated Operational Occurrences (AOOs). (As defined in 10 CFR 50, Appendix A, "Anticipated operational occurances mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.") By tripping the reactor, the RPS also assists the Engineered Safety Features (ESF) systems in mitigating accidents.

The protection and monitoring systems have been designed to ensure safe operation of the reactor. This is achieved by specifying Limiting Safety System Settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this Specification as the Allowable Value, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).

During AOOs, which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling;
- Fuel centerline melting shall not occur; and
- The Primary Coolant System (PCS) pressure SL of 2750 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 50 (Ref. 1) and 10 CFR 100 (Ref. 2) criteria during AOOs.

BASES

BACKGROUND (continued)

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 (Ref. 2) limits. Different accident categories allow a different fraction of these limits based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The RPS is segmented into four interconnected modules. These modules are:

- Measurement channels (or pressure switches);
- Bistable trip units;
- Matrix Logic; and
- Trip Initiation Logic.

This LCO addresses the RPS Logic (Matrix Logic and Trip Initiation Logic), including Manual Trip capability. LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation," provides a description of the role of the measurement channels and associated bistable trip units in the RPS. The RPS Logic is summarized below:

RPS Logic

The RPS Logic, consisting of Matrix Logic and Trip Initiation Logic, employs a scheme that provides a reactor trip when trip units in any two of the four channels sense the same input parameter trip. This is called a two-out-of-four trip logic. This logic and the clutch power supply configuration are shown in FSAR Figure 7-1 (Ref. 3).

Bistable trip unit relay contact outputs from the four channels are configured into six logic matrices. Each logic matrix checks for a coincident trip in the same parameter in two trip unit channels. The matrices are designated the AB, AC, AD, BC, BD, and CD matrices to reflect the bistable trip unit channels being monitored. Each logic matrix contains four normally energized matrix relays. When a coincidence is detected, consisting of a trip in the same Function in the two channels being monitored by the logic matrix, all four matrix relay coils de-energize.

BASES

BACKGROUND (continued)

RPS Logic (continued)

The matrix relay contacts are arranged into trip paths, with one of the four matrix relays in each matrix opening contacts in one of the four trip paths. Each trip path provides power to one of the four normally energized clutch power supply "M-contactors" (M1, M2, M3, and M4). The trip paths thus each have six contacts in series, one from each matrix, and perform a logical OR function, de-energizing the M-contactors if any one or more of the six logic matrices indicate a coincidence condition.

When a coincidence occurs in two RPS channels, all four matrix relays in the affected matrix de-energize. This in turn de-energizes all four M-contactors, which interrupt AC input power to the four clutch power supplies, allowing the full-length control rods to insert by gravity.

Manual reactor trip capability is afforded by two main control panel-mounted pushbuttons. One of these (on Control Panel CO-2) opens contacts in series with each of the four trip paths, de-energizing all M-contactors. The other pushbutton (on Control Panel CO-6) opens circuit breakers which provide AC input power to the M-contactor contacts and downstream clutch power supplies. Thus depressing either pushbutton will cause a reactor trip.

De-energizing the M-contactors removes AC power to the four clutch power supply inputs. Contacts from M-contactors M1 and M2 are in series with each other and in the AC power supply path to clutch power supplies PS1 and PS2 (these constitute a "trip leg"). M3 and M4 are similarly arranged with respect to clutch power supplies PS3 and PS4 (these constitute a second "trip leg"). Approximately half of the control rod clutches receive power from auctioneered clutch power supplies 1 and 3. The remaining control rod clutches receive clutch power from auctioneered clutch power supplies 2 and 4.

Matrix Logic refers to the matrix power supplies, trip channel bypass contacts, and interconnecting RPS cabinet matrix wiring between bistable and auxiliary trip unit relay contacts, including the matrix relays. Contacts in the bistable and auxiliary trip units are excluded from the Matrix Logic definition, since they are addressed as part of the instrumentation channel.

The Trip Initiation Logic consists of the M-contactor isolation transformers, all interconnecting wiring, and the M-contactors.

BASES

BACKGROUND (continued)

RPS Logic (continued)

Manual trip circuitry includes both manual reactor trip pushbuttons C0-2 and C0-6, and the interconnecting wiring necessary to effect deenergization of the clutch power supplies.

Neither the clutch power supplies nor the AC input power source to these supplies is considered as safety related. Operation may continue with one or two selective clutch power supplies de-energized.

It is possible to change the two-out-of-four RPS Logic to a two-out-of-three logic for a given input parameter in one channel at a time by Trip Channel Bypassing the RPS Trip unit output contacts in the Matrix Logic "Ladder." Trip Channel Bypassing a trip unit effectively shorts the trip unit relay contacts in the three matrices associated with that channel. Thus, the bypassed trip units will function normally, producing normal channel trip indication and annunciation, but a reactor trip will not occur unless two additional channels indicate a trip condition. Trip Channel Bypassing can be simultaneously performed on any number of parameters in any number of channels, providing each parameter is bypassed in only one channel at a time. A single bypass key for each trip function interlock prevents simultaneous Trip Channel Bypassing of the same parameter in more than one channel. Trip Channel Bypassing is normally employed during maintenance or testing.

Functional testing of the entire RPS, from trip unit input through the de-energizing of individual sets of clutch power supplies, can be performed either at power or during shutdown and is normally performed on a quarterly basis. FSAR Section 7.2 (Ref. 4) explains RPS testing in more detail.

APPLICABLE SAFETY ANALYSES

Reactor Protective System (RPS) Logic

The RPS Logic provides for automatic trip initiation to avoid exceeding the SLs during AOOs and to assist the ESF systems in ensuring acceptable consequences during accidents. All transients and accidents that call for a reactor trip assume the RPS Logic is functioning as designed.

BASES

APPLICABLE SAFETY ANALYSIS (continued)

Manual Trip

There are no accident analyses that take credit for the Manual Trip; however, the Manual Trip is part of the RPS circuitry. It is used by the operator to shut down the reactor whenever any parameter is rapidly trending toward its trip setpoint. A Manual Trip accomplishes the same results as any one of the automatic trip Functions.

The RPS Logic and Trip Initiation satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

Reactor Protective System (RPS) Logic

Failures of individual trip unit relays and their contacts are addressed in LCO 3.3.1. This Specification addresses failures of the Matrix Logic not addressed in the above, such as the failure of matrix relay power supplies or the failure of the trip channel bypass contact in the bypass condition.

Loss of a single preferred AC bus will de-energize one of the two power supplies in each of three matrices. Because of power supply auctioneering, all four matrix relays will remain energized in each affected matrix.

Each of the four Trip Initiation Logic channels de-energizes one set of clutch power supplies if any of the six coincidence matrices de-energize their associated matrix relays. They thus perform a logical OR function. Trip Initiation Logic channels 1 and 2 receive AC power from preferred AC bus Y-30. Trip Initiation Logic channels 3 and 4 receive AC input power from preferred AC bus Y-40. Because of clutch power supply output auctioneering, it is possible to de-energize either input bus without de-energizing control rod clutches.

1. Matrix Logic

   This LCO requires six channels of Matrix Logic to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when more than one full-length control rod is capable of being withdrawn and the PCS boron concentration is less than REFUELING BORON CONCENTRATION.

BASES

LCO (continued)

2. Trip Initiation Logic

This LCO requires four channels of Trip Initiation Logic to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when more than one full-length control rod is capable of being withdrawn and the PCS boron concentration is less than REFUELING BORON CONCENTRATION.

3. Manual Trip

The LCO requires both Manual Trip channels to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when more than one full-length control rod is capable of being withdrawn and the PCS boron concentration is less than REFUELING BORON CONCENTRATION.

Two independent pushbuttons are provided. Each pushbutton is considered to be a channel. Depressing either pushbutton interrupts power to all four clutch power supplies, tripping the reactor.

APPLICABILITY

The RPS Matrix Logic, Trip Initiation Logic, and Manual Trip are required to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when more than one full-length control rod capable of being withdrawn and the PCS boron concentration is less than REFUELING BORON CONCENTRATION. This ensures the reactor can be tripped when necessary, but allows for maintenance and testing when the reactor trip is not needed.

In MODES 3, 4, and 5 with no more than one full-length control rod capable of being withdrawn or the PCS boron concentration at REFUELING BORON CONCENTRATION, these Functions do not have to be OPERABLE. However, LCO 3.3.9, "Neutron Flux Monitoring Channels," does require neutron flux monitoring capability under these conditions.

BASES

ACTIONS

When the number of inoperable channels in a trip Function exceeds that specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 is immediately entered if applicable in the current MODE of operation.

A.1

Condition A applies if one Matrix Logic channel is inoperable. The channel must be restored to OPERABLE status within 48 hours. The Completion Time of 48 hours provides the operator time to take appropriate actions and still ensures that any risk involved in operating with a failed channel is acceptable. Operating experience has demonstrated that the probability of a random failure of a second Matrix Logic channel is low during any given 48 hour interval. If the channel cannot be restored to OPERABLE status within 48 hours, Condition E is entered.

B.1

Condition B applies if one Trip Initiation Logic channel is inoperable. The Required Action require de-energizing the affected clutch power supplies. This removes the need for the affected channel by performing its associated safety function. With the clutch power supplies associated with one initiation logic channel de-energized, the remaining two clutch power supplies prevent control rod clutches from de-energizing. The remaining clutch power supplies are in a one-out-of-two logic with respect to the remaining initiation logic channels in the clutch power supply path. This meets redundancy requirements, but testing on the OPERABLE channels cannot be performed without causing a reactor trip.

Required Action B.1 provides for de-energizing the affected clutch power supplies associated with the inoperable channel within a Completion Time of 1 hour.

BASES

ACTIONS (continued)

C.1

Condition C applies to the failure of one Manual Trip channel. With one manual reactor trip channel inoperable operation may continue until the reactor is shut down for other reasons. Repair during operation is not required because one OPERABLE channel is all that is required for safe operation. No safety analyses assume operation of the Manual trip.

The Manual Trip channels are not testable without actually causing a reactor trip, so even if the difficulty were corrected, the post maintenance testing necessary to declare the channel OPERABLE could not be completed during operation. Because of this, the Required Action is to restore the inoperable channel to OPERABLE status prior to entering MODE 2 from MODE 3 during the next plant startup.

D.1

Condition D applies to the failure of both Trip Initiation Logic channels affecting the same trip leg. The affected control rod drive clutch power supplies must be de-energized immediately. With both channels inoperable, the RPS Function is lost if the affected clutch power supplies are not de-energized. Therefore, immediate action is required to de-energize the affected clutch power supplies. The immediate Completion Time is appropriate since there could be a loss of safety function if the associated clutch power supplies are not de-energized.

E.1, E.2.1 and E.2.2

Condition E is entered if Required Actions associated with Condition A, B, C, or D are not met within the required Completion Time or if for one or more Functions more than one Manual Trip, Matrix Logic, or Trip Initiation Logic channel is inoperable for reasons other than Condition D.

In Condition E the reactor must be placed in a MODE in which the LCO does not apply. The Completion Time of 6 hours to be in MODE 3 is reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

E.1, E.2.1 and E.2.2 (continued)

Required Actions E.2.1 and E.2.2 allow 6 hours to verify that no more than one full-length control rod is capable of being withdrawn or to verify that PCS boron concentration is at REFUELING BORON CONCENTRATION. The Completion Time is reasonable to place the plant in an operating condition in which the LCO does not apply.

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1

A CHANNEL FUNCTIONAL TEST on each RPS Logic channel is performed every 92 days to ensure the entire channel will perform its intended function when needed. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

This SR addresses the two tests associated with the RPS Logic: Matrix Logic and Trip Initiation Logic.

Matrix Logic Tests

These tests are performed one matrix at a time. They verify that a coincidence in the two input channels for each Function removes power from the matrix relays. During testing, power is applied to the matrix relay test coils and prevents the matrix relay contacts from assuming their de-energized state. The Matrix Logic tests will detect any short circuits around the bistable contacts in the coincidence logic such as may be caused by faulty bistable relay or trip channel bypass contacts.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1 (continued)

Trip Initiation Logic Tests

These tests are similar to the Matrix Logic tests, except that test power is withheld from one matrix relay at a time, allowing the initiation circuit to de-energize, de-energizing the affected set of clutch power supplies.

The Frequency of 92 days is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 5).

SR 3.3.2.2

A CHANNEL FUNCTIONAL TEST on the Manual Trip channels is performed prior to a reactor startup to ensure the entire channel will perform its intended function if required. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

The Manual Trip Function is not tested at power. However, the simplicity of this circuitry and the absence of drift concern makes this Frequency adequate. Additionally, operating experience has shown that these components usually pass the Surveillance when performed once within 7 days prior to each reactor startup.

REFERENCES

1. 10 CFR 50, Appendix A
2. 10 CFR 100
3. FSAR, Figure 7-1
4. FSAR, Section 7.2
5. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989

B 3.3 INSTRUMENTATION

B 3.3.3 Engineered Safety Features (ESF) Instrumentation

BASES

BACKGROUND

The ESF Instrumentation initiates necessary safety systems, based upon the values of selected plant parameters, to protect against violating core design limits and the Primary Coolant System (PCS) pressure boundary and to mitigate accidents.

The ESF circuitry generates the signals listed below when the monitored variables reach levels that are indicative of conditions requiring protective action. The inputs to each ESF actuation signal are also listed.

1. Safety Injection Signal (SIS).
    a. Containment High Pressure (CHP)
    b. Pressurizer Low Pressure
2. Steam Generator Low Pressure (SGLP);
    a. Steam Generator A Low Pressure
    b. Steam Generator B Low Pressure
3. Recirculation Actuation Signal (RAS);
    a. Safety Injection Refueling Water Tank (SIRWT) Low Level
4. Auxiliary Feedwater Actuation Signal (AFAS);
    a. Steam Generator A Low Level
    b. Steam Generator B Low Level
5. Containment High Pressure Signal (CHP);
    a. Containment High Pressure - Left Train
    b. Containment High Pressure - Right Train

BASES

BACKGROUND (continued)

6. Containment High Radiation Signal (CHR);

    a. Containment High Radiation

7. Automatic Bypass Removal

    a. Pressurizer Pressure Low Bypass

    b. Steam Generator A Low Pressure Bypass

    c. Steam Generator B Low Pressure Bypass

In the above list of actuation signals, the CHP and RAS are derived from pressure and level switches, respectively.

Equipment actuated by each of the above signals is identified in the FSAR, Chapter 7. (Ref. 1).

The ESF circuitry, with the exception of RAS, employs two-out-of-four logic. Four independent measurement channels are provided for each function used to generate ESF actuation signals. When any two channels of the same function reach their setpoint, actuating relays are energized which, in turn, initiate the protective actions. Two separate and redundant trains of actuating relays, each powered from separate power supplies, are utilized. These separate relay trains operate redundant trains of ESF equipment.

RAS logic consists of output contacts of the relays actuated by the SIRWT level switches arranged in a "one-out-of-two taken twice" logic. The contacts are arranged so that at least one low level signal powered from each station battery is required to initiate RAS. Loss of a single battery, therefore, cannot either cause or prevent RAS initiation.

The ESF logic circuitry contains the capability to manually block the SIS actuation logic and the SGLP action logic during normal plant shutdowns to avoid undesired actuation of the associated equipment. In each case, when three of the four associated measurement channels are below the block setpoint, pressing a manual pushbutton will block the actuation signal for that train. If two of the four of the measurement channels increase above the block setpoint, the block will automatically be removed.

BASES

BACKGROUND (continued)

7. Automatic Bypass Removal (continued)

The sensor subsystems, including individual channel actuation bistables, is addressed in this LCO. The actuation logic subsystems, manual actuation, and downstream components used to actuate the individual ESF components are addressed in LCO 3.3.4.

Measurement Channels

Measurement channels, consisting of pressure switches, field transmitters, or process sensors and associated instrumentation, provide a measurable electronic signal based upon the physical characteristics of the parameter being measured.

Four identical measurement channels are provided for each parameter used in the generation of trip signals. These are designated Channels A through D. Measurement channels provide input to ESF bistables within the same ESF channel. In addition, some measurement channels may also be used as inputs to Reactor Protective System (RPS) bistables, and most provide indication in the control room.

When a channel monitoring a parameter indicates an abnormal condition, the bistable monitoring the parameter in that channel will trip. In the case of RAS and CHP, the sensors are latching auxiliary relays from level and pressure switches, respectively, which do not develop an analog input to separate bistables. Tripping two or more channels monitoring the same parameter will actuate both channels of Actuation Logic of the associated ESF equipment.

Three of the four measurement and bistable channels are necessary to meet the redundancy and testability of GDC 21 in Appendix A to 10 CFR 50 (Ref. 2). The fourth channel provides additional flexibility by allowing one channel to be removed from service for maintenance or testing while still maintaining a minimum two-out-of-three logic.

Since no single failure will prevent a protective system actuation and no protective channel feeds a control channel, this arrangement meets the requirements of IEEE Standard 279 -1971 (Ref. 3).

BACKGROUND (continued)

Measurement Channels (continued)

The ESF Actuation Functions are generated by comparing a single measurement to a fixed bistable setpoint. The ESF Actuation Functions utilize the following input instrumentation:

- Safety Injection Signal (SIS)

  The Safety Injection Signal can be generated by any of three inputs: Pressurizer Low Pressure, Containment High Pressure, or Manual Actuation. Manual Actuation is addressed by LCO 3.3.4; Containment High Pressure is discussed below. Four instruments (channels A through D), monitor Pressurizer Pressure to develop the SIS actuation. Each of these instrument channels has two individually adjustable ESF bistable trip devices, one for the bypass removal circuit (discussed below) and one for SIS. Each ESF bistable trip device actuates two auxiliary relays, one for each actuation train. The output contacts from these auxiliary relays form the logic circuits addressed in LCO 3.3.4. The instrument channels associated with each Pressurizer Low Pressure SIS actuation bistable include the pressure measurement loop, the SIS actuation bistable, and the two auxiliary relays associated with that bistable. The bistables associated with automatic removal of the Pressurizer Low Pressure Bypass are discussed under Function 7.a, below.

- Low Steam Generator Pressure Signal (SGLP)

  There are two separate Low Steam Generator Pressure signals, one for each steam generator. For each steam generator, four instruments (channels A through D) monitor pressure to develop the SGLP actuation. Each of these instrument channels has two individually adjustable ESF bistable trip devices, one for the bypass removal circuit (discussed below) and one for SGLP. Each Steam SGLP bistable trip device actuates an auxiliary relay. The output contacts from these auxiliary relays form the SGLP logic circuits addressed in LCO 3.3.4. The instrument channels associated with each Steam Generator Low Pressure Signal bistable include the pressure measurement loop, the SGLP actuation bistable, and the auxiliary relay associated with that bistable. The bistables associated with automatic removal of the SGLP Bypass are discussed under Function 7.a, below.

BASES

BACKGROUND (continued)

Measurement Channels (continued)

- Recirculation Actuation Signal (RAS)

  There are four Safety Injection Refueling Water (SIRW) Tank level instruments used to develop the RAS signal. Each of these instrument channels actuates two auxiliary relays, one for each actuation train. The output contacts from these auxiliary relays form the logic circuits addressed in LCO 3.3.4. The SIRW Tank Low Level instrument channels associated with each RAS actuation bistable include the level instrument and the two auxiliary relays associated with that instrument.

- Auxiliary Feedwater Actuation Signal (AFAS)

  There are two separate AFAS signals (AFAS channels A and B), each one actuated on low level in either steam generator. For each steam generator, four level instruments (channels A through D) monitor level to develop the AFAS actuation signals. The output contacts from the bistables on these level channels form the SGLP logic circuits addressed in LCO 3.3.4. The instrument channels associated with each Steam Generator Low Level Signal bistable include the level measurement loop and the Low Level AFAS bistable.

- Containment High Pressure Actuation (CHP)

  The Containment High Pressure signal is actuated by two sets of four pressure switches, one set for each train. The output contacts from these pressure switches form the CHP logic circuits addressed in LCO 3.3.4.

BASES

BACKGROUND (continued)

Measurement Channels (continued)

- Containment High Radiation Actuation (CHR)

  The CHR signal can be generated by either of two inputs: High Radiation or Manual Actuation. Manual Actuation is addressed by LCO 3.3.4. Four radiation monitor instruments (channels A through D), monitor containment area radiation level to develop the CHR signal. Each CHR monitor bistable device actuates one auxiliary relay which has contacts in each CHR logic train addressed in LCO 3.3.4. The instrument channels associated with each CHR actuation bistable include the radiation monitor itself and the associated auxiliary relay.

- Automatic Bypass Removal Functions

  Pressurizer Low Pressure and Steam Generator Low Pressure logic circuits have the capability to be blocked to avoid undesired actuation when pressure is intentionally lowered during plant shutdowns. In each case these bypasses are automatically removed when the measured pressure exceeds the bypass permissive setpoint. The measurement channels which provide the bypass removal signal are the same channels which provide the actuation signal. Each of these pressure measurement channels has two bistables, one for actuation and one for the bypass removal Function. The pressurizer pressure channels include an auxiliary relay actuated by the bypass removal bistable. The logic circuits for Automatic Bypass Removal Functions are addressed by LCO 3.3.4.

Several measurement instrument channels provide more than one required function. Those sensors shared for RPS and ESF functions are identified in Table B 3.3.1-1. That table provides a listing of those shared channels and the Specifications which they affect.

BASES

BACKGROUND (continued)

Bistable Trip Units

There are four channels of bistables, designated A through D, for each ESF Function, one for each measurement channel. The bistables for all required Functions, except CHP and RAS, receive an analog input from the measurement device, compare the analog input to trip setpoints, and provide contact output to the Actuation Logic. CHP and RAS are actuated by pressure switches and level switches respectively.

The Allowable Values are specified for each safety related ESF trip Function which is credited in the safety analysis. Nominal trip setpoints are specified in the plant procedures. The nominal setpoints are selected to ensure plant parameters do not exceed the Allowable Value if the instrument loop is performing as required. The methodology used to determine the nominal trip setpoints is also provided in plant documents. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Each Allowable Value specified is more conservative than the analytical limit determined in the safety analysis in order to account for uncertainties appropriate to the trip Function. These uncertainties are addressed as described in plant documents. A channel is inoperable if its actual setpoint is not within its Allowable Value.

Setpoints in accordance with the Allowable Value will ensure that Safety Limits of Chapter 2.0, "SAFETY LIMITS (SLs)," are not violated during Anticipated Operational Occurrences (AOOs) and that the consequences of Design Basis Accidents (DBAs) will be acceptable, providing the plant is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed. (As defined in 10 CFR 50, Appendix A, "Anticipated operational occurances mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.")

ESF Instrument Channel Bypasses

The only ESF instrument channels with built-in bypass capability are the Low SG Level AFAS bistables. Those bypasses are effected by a key operated switch, similar to the RPS Trip Channel Bypasses. A bypassed Low SG Level channel AFAS bistable cannot perform its specified function and must be considered inoperable.

BASES

BACKGROUND (continued)

ESF Instrument Channel Bypasses (continued)

While there are no other built-in provisions for instrument channel bypasses in the ESF design (bypassing any other channel output requires opening a circuit link, lifting a lead, or using a jumper), this LCO includes requirements for OPERABILITY of the instrument channels and bistables which provide input to the Automatic Bypass Removal Logic channels required by LCO 3.3.4, "ESF Logic and Manual Initiation."

The Actuation Logic channels for Pressurizer Pressure and Steam Generator Low Pressure, however, have the ability to be manually bypassed when the associated pressure is below the range where automatic protection is required. These actuation logic channel bypasses may be manually initiated when three-out-of-four bypass permissive bistables indicate below their setpoint. When two-out-of-four of these bistables are above their bypass permissive setpoint, the actuation logic channel bypass is automatically removed. The bypass permissive bistables use the same four measurement channels as the blocked ESF function for their inputs.

APPLICABLE SAFETY ANALYSES

Each of the analyzed accidents can be detected by one or more ESF Functions. One of the ESF Functions is the primary actuation signal for that accident. An ESF Function may be the primary actuation signal for more than one type of accident. An ESF Function may also be a secondary, or backup, actuation signal for one or more other accidents. Functions not specifically credited in the accident analysis, serve as backups and are part of the NRC approved licensing basis for the plant.

BASES

APPLICABLE SAFETY ANALYSES (continued)

ESF protective Functions are as follows.

1. Safety Injection Signal (SIS)

The SIS ensures acceptable consequences during Loss of Coolant Accident (LOCA) events, including steam generator tube rupture, and Main Steam Line Breaks (MSLBs) or Feedwater Line Breaks (FWLBs) (inside containment). To provide the required protection, SIS is actuated by a CHP signal, or by two-out-of-four Pressurizer Low Pressure channels decreasing below the setpoint. SIS initiates the following actions:

a. Start HPSI & LPSI pumps;

b. Start component cooling water and service water pumps;

c. Initiate service water valve operations;

d. Initiate component cooling water valve operations;

e. Start containment cooling fans (when coincident with a loss of offsite power);

f. Enable Containment Spray Pump Start on CHP; and

g. Initiate Safety Injection Valve operations.

Each SIS logic train is also actuated by a contact pair on one of the CHP initiation relays for the associated CHP train.

2. Steam Generator Low Pressure Signal (SGLP)

The SGLP ensures acceptable consequences during an MSLB or FWLB by isolating the steam generator if it indicates a low steam generator pressure. The SGLP concurrent with or following a reactor trip, minimizes the rate of heat extraction and subsequent cooldown of the PCS during these events.

BASES

APPLICABLE SAFETY ANALYSES (continued)

2. Steam Generator Low Pressure Signal (SGLP) (continued)

One SGLP circuit is provided for each SG. Each SGLP circuit is actuated by two-out-of-four pressure channels on the associated SG reaching their setpoint. SGLP initiates the following actions:

a. Close the associated Feedwater Regulating valve and its bypass; and

b. Close both Main Steam Isolation Valves.

3. Recirculation Actuation Signal

At the end of the injection phase of a LOCA, the SIRWT will be nearly empty. Continued cooling must be provided by the ECCS to remove decay heat. The source of water for the ECCS pumps is automatically switched to the containment recirculation sump. Switchover from SIRWT to the containment sump must occur before the SIRWT empties to prevent damage to the ECCS pumps and a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sump to support pump suction.

Furthermore, early switchover must not occur to ensure sufficient borated water is injected from the SIRWT to ensure the reactor remains shut down in the recirculation mode. An SIRWT Low Level signal initiates the RAS.

RAS initiates the following actions:

a. Trip LPSI pumps (this trip can be manually bypassed);

b. Switch HPSI and containment spray pump suction from SIRWT to Containment Sump by opening sump CVs and closing SIRWT CVs; and

c. Adjust cooling water to component cooling heat exchangers.

BASES

APPLICABLE SAFETY ANALYSES (continued)

3. Recirculation Actuation Signal (continued)

The RAS signal is actuated by separate sensors from those which provide tank level indication. The allowable range of 21" to 27" above the tank floor corresponds to 1.1% to 3.3% indicated level. Typically the actual setting is near the midpoint of the allowable range.

4. Auxiliary Feedwater Actuation Signal

An AFAS initiates feedwater flow to both steam generators if a low level is indicated in either steam generator.

The AFAS maintains a steam generator heat sink during the following events:

- MSLB;
- FWLB;
- LOCA; and
- Loss of feedwater.

5. Containment High Pressure Signal (CHP)

The CHP signal closes all containment isolation valves not required for ESF operation and starts containment spray (if SIS enabled), ensuring acceptable consequences during LOCAs, control rod ejection events, MSLBs, or FWLBs (inside containment).

CHP is actuated by two-out-of-four pressure switches for the associated train reaching their setpoints. CHP initiates the following actions:

a. Containment Spray;

b. Safety Injection Signal;

c. Main Feedwater Isolation;

BASES

APPLICABLE SAFETY ANALYSIS (continued)

5. Containment High Pressure Signal (CHP) (continued)

   d. Main Steam Line Isolation;

   e. Control Room HVAC Emergency Mode; and

   f. Containment Isolation Valve Closure.

6. Containment High Radiation Signal (CHR)

   CHR is actuated by two-out-of-four radiation monitors exceeding their setpoints. CHR initiates the following actions to ensure acceptable consequences following a LOCA or control rod ejection event:

   a. Control Room HVAC Emergency Mode;

   b. Containment Isolation Valve Closure; and

   c. Block automatic starting of ECCS pump room sump pumps.

   During refueling operations, separate switch-selectable radiation monitors initiate CHR, as addressed by LCO 3.3.6.

7. Automatic Bypass Removal Functions

   The logic circuitry provides automatic removal of the Pressurizer Pressure Low and Steam Generator Pressure Low actuation signal bypasses. There are no assumptions in the safety analyses which assume operation of these automatic bypass removal circuits, and no analyzed events result in conditions where the automatic removal would be required to mitigate the event. The automatic removal circuits are required to assure that logic circuit bypasses will not be overlooked during a plant startup.

The ESF Instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2).

BASES

LCO

The LCO requires all channel components necessary to provide an ESF actuation to be OPERABLE.

The Bases for the LCO on ESF Functions are addressed below.

1. Safety Injection Signal (SIS)

This LCO requires four channels of SIS Pressurizer Low Pressure to be OPERABLE in MODES 1, 2, and 3.

The setpoint was chosen so as to be low enough to avoid actuation during plant operating transients, but to be high enough to be quickly actuated by a LOCA or MSLB. The settings include an uncertainty allowance which is consistent with the settings assumed in the MSLB analysis (which bounds the settings assumed in the LOCA analysis).

2. Steam Generator Low Pressure Signal (SGLP)

This LCO requires four channels of Steam Generator Low Pressure Instrumentation for each SG to be OPERABLE in MODES 1, 2, and 3. However, as indicated in Table 3.3.3-1, Note (a), the SGLP Function is not required to be OPERABLE in MODES 2 or 3 if all Main Steam Isolation Valves (MSIVs) are closed and deactivated and all Main Feedwater Regulating Valves (MFRVs) and MFRV bypass valves are either closed and deactivated or isolated by closed manual valves.

The setpoint was chosen to be low enough to avoid actuation during plant operation, but be close enough to full power operating pressure to be actuated quickly in the event of a MSLB. The setting includes an uncertainty allowance which is consistent with the setting used in the Reference 4 analysis.

Each SGLP logic is made up of output contacts from four pressure bistables from the associated SG. When the logic circuit is satisfied, two relays are energized to actuate steam and feedwater line isolation.

BASES

LCO (continued)

2. Steam Generator Low Pressure Signal (SGLP) (continued)

This LCO applies to failures in the four sensor subsystems, including sensors, bistables, and associated equipment. Failures in the actuation subsystems are considered Actuation Logic failures and are addressed in LCO 3.3.4.

3. Recirculation Actuation Signal (RAS)

This LCO requires four channels of SIRWT Low Level to be OPERABLE in MODES 1, 2, and 3.

The setpoint was chosen to provide adequate water in the containment sump for HPSI pump net positive suction head following an accident, but prevent the pumps from running dry during the switchover.

The upper limit on the Allowable Value for this trip is set low enough to ensure RAS does not initiate before sufficient water is transferred to the containment sump. Premature recirculation could impair the reactivity control Function of safety injection by limiting the amount of boron injection. Premature recirculation could also damage or disable the recirculation system if recirculation begins before the sump has enough water.

The lower limit on the SIRWT Low Level trip Allowable Value is high enough to transfer suction to the containment sump prior to emptying the SIRWT.

4. Auxiliary Feedwater Actuation Signal (AFAS)

The AFAS logic actuates AFW to each SG on a SG Low Level in either SG.

The Allowable Value was chosen to assure that AFW flow would be initiated while the SG could still act as a heat sink and steam source, and to assure that a reactor trip would not occur on low level without the actuation of AFW.

BASES

LCO (continued)

4. Auxiliary Feedwater Actuation Signal (AFAS) (continued)

This LCO requires four channels for each steam generator of Steam Generator Low Level to be OPERABLE in MODES 1, 2, and 3.

5. Containment High Pressure Signal (CHP)

This LCO requires four channels of CHP to be OPERABLE for each of the associated ESF trains (left and right) in MODES 1, 2, 3 and 4.

The setpoint was chosen so as to be high enough to avoid actuation by containment temperature or atmospheric pressure changes, but low enough to be quickly actuated by a LOCA or a MSLB in the containment.

6. Containment High Radiation Signal (CHR)

This LCO requires four channels of CHR to be OPERABLE in MODES 1, 2, 3, and 4.

The setpoint is based on the maximum primary coolant leakage to the containment atmosphere allowed by LCO 3.4.13 and the maximum activity allowed by LCO 3.4.16. N^{16} concentration reaches equilibrium in containment atmosphere due to its short half-life, but other activity was assumed to build up. At the end of a 24 hour leakage period the dose rate is approximately 20 R/h as seen by the area monitors. A large leak could cause the area dose rate to quickly exceed the 20 R/h setting and initiate CHR.

7. Automatic Bypass Removal

The automatic bypass removal logic removes the bypasses which are used during plant shutdown periods, for Pressurizer Low Pressure and Steam Generator Low Pressure actuation signals.

The setpoints were chosen to be above the setpoint for the associated actuation signal, but well below the normal operating pressures.

BASES

LCO (continued)

7. Automatic Bypass Removal (continued)

This LCO requires four channels of Pressurizer Low Pressure bypass removal and four channels for each steam generator of Steam Generator Low Pressure bypass removal, to be OPERABLE in MODES 1, 2, and 3.

APPLICABILITY

All ESF Functions are required to be OPERABLE in MODES 1, 2, and 3. In addition, Containment High Pressure and Containment High Radiation are required to be operable in MODE 4.

In MODES 1, 2, and 3 there is sufficient energy in the primary and secondary systems to warrant automatic ESF System responses to:

- Close the main steam isolation valves to preclude a positive reactivity addition and containment overpressure;
- Actuate AFW to preclude the loss of the steam generators as a heat sink (in the event the normal feedwater system is not available);
- Actuate ESF systems to prevent or limit the release of fission product radioactivity to the environment by isolating containment and limiting the containment pressure from exceeding the containment design pressure during a design basis LOCA or MSLB; and
- Actuate ESF systems to ensure sufficient borated inventory to permit adequate core cooling and reactivity control during a design basis LOCA or MSLB accident.

The CHP and CHR Functions are required to be OPERABLE in MODE 4 to limit leakage of radioactive material from containment and limit operator exposure during and following a DBA.

The SGLP Function is not required to be OPERABLE in MODES 2 and 3, if all MSIVs are closed and deactivated and all MFRVs and MFRV bypass valves are either closed and deactivated or isolated by closed manual valves, since the SGLP Function is not required to perform any safety functions under these conditions.

BASES

APPLICABILITY (continued)

In lower MODES, automatic actuation of ESF Functions is not required, because adequate time is available for plant operators to evaluate plant conditions and respond by manually operating the ESF components.

LCO 3.3.6 addresses automatic Refueling CHR isolation during CORE ALTERATIONS or during movement of irradiated fuel.

In MODES 5 and 6, ESFAS initiated systems are either reconfigured or disabled for shutdown cooling operation. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components.

ACTIONS

The most common causes of channel inoperability are outright failure of loop components or drift of those loop components which is sufficient to exceed the tolerance provided in the plant setpoint analysis. Loop component failures are typically identified by the actuation of alarms due to the channel failing to the "safe" condition, during CHANNEL CHECKS (when the instrument is compared to the redundant channels), or during the CHANNEL FUNCTIONAL TEST (when an automatic component might not respond properly). Typically, the drift of the loop components is found to be small and results in a delay of actuation rather than a total loss of function. Excessive loop component drift would, most likely, be identified during a CHANNEL CHECK (when the instrument is compared to the redundant channels) or during a CHANNEL CALIBRATION (when instrument loop components are checked against reference standards).

Typically, the drift is small and results in a delay of actuation rather than a total loss of function. Determination of setpoint drift is generally made during the performance of a CHANNEL FUNCTIONAL TEST when the process instrument is set up for adjustment to bring it to within specification. If the actual trip setpoint is not within the Allowable Value in Table 3.3.3-1, the channel is inoperable and the appropriate Condition(s) are entered.

In the event a channel's trip setpoint is found nonconservative with respect to the Allowable Value in Table 3.3.3-1, or the sensor, instrument loop, signal processing electronics, or ESF bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the plant must enter the Condition statement for the particular protection Function affected.

BASES

ACTIONS (continued)

When the number of inoperable channels in a trip Function exceeds those specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered if applicable in the current MODE of operation.

A Note has been added to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function in Table 3.3.3-1. Completion Times for the inoperable channel of a Function will be tracked separately.

A.1

Condition A applies to the failure of a single bistable or associated instrumentation channel of one or more input parameters in each ESF Function except the RAS Function. Since the bistable and associated instrument channel combine to perform the actuation function, the Condition is also appropriate if both the bistable and associated instrument channel are inoperable.

ESF coincidence logic is normally two-out-of-four. If one ESF channel is inoperable, startup or power operation is allowed to continue as long as action is taken to restore the design level of redundancy.

If one ESF channel is inoperable, startup or power operation is allowed to continue, providing the inoperable channel actuation bistable is placed in trip within 7 days. The provision of four trip channels allows one channel to be inoperable in a non-trip condition up to the 7 day Completion Time allotted to place the channel in trip. Operating with one failed channel in a non-trip condition during operations, places the ESF Actuation Logic in a two-out-of-three coincidence logic.

If the failed channel cannot be restored to OPERABLE status in 7 days, the associated bistable is placed in a tripped condition. This places the function in a one-out-of-three configuration.

BASES

ACTIONS (continued)

A.1 (continued)

In this configuration, common cause failure of the dependent channel cannot prevent ESF actuation. The 7 day Completion Time is based upon operating experience, which has demonstrated that a random failure of a second channel occurring during the 7 day period is a low probability event.

Condition A is modified by a Note which indicates it is not applicable to the SIRWT Low Level Function.

B.1 and B.2

Condition B applies to the failure of two channels in any of the ESF Functions except the RAS Function.

With two inoperable channels, one channel actuation device must be placed in trip within the 8 hour Completion Time. Eight hours is allowed for this action since it must be accomplished by a circuit modification, or by removing power from a circuit component. With one channel of protective instrumentation inoperable, the ESF Actuation Logic Function is in two-out-of-three logic, but with another channel inoperable the ESF may be operating with a two-out-of-two logic. This is outside the assumptions made in the analyses and should be corrected. To correct the problem, the second channel is placed in trip. This places the ESF in a one-out-of-two logic. If any of the other OPERABLE channels receives a trip signal, ESF actuation will occur.

One of the failed channels must be restored to OPERABLE status within 7 days, and the provisions of Condition A still applied to the remaining inoperable channel. Therefore, the channel that is still inoperable after completion of Required Action B.2 must be placed in trip if more than 7 days has elapsed since the channel's initial failure.

BASES

ACTIONS (continued)

B.1 and B.2 (continued)

Condition B is modified by a Note which indicates that it is not applicable to the SIRWT Low Level Function. The Required Action is also modified by a Note stating that LCO 3.0.4 is not applicable. The Note was added to allow the changing of MODES even though two channels are inoperable, with one channel tripped. MODE changes in this configuration are allowed, to permit maintenance and testing on one of the inoperable channels. In this configuration, the protection system is in a one-out-of-two logic, and the probability of a common cause failure affecting both of the OPERABLE channels during the 7 days permitted is remote.

C.1 and C.2

Condition C applies to one RAS SIRWT Low Level channel inoperable. The SIRWT low level circuitry is arranged in a "1-out-of-2 taken twice" logic rather than the more frequently used 2-out-of-4 logic. Therefore, Required Action C.1 differs from other ESF functions. With a bypassed SIRWT low level channel, an additional failure might disable automatic RAS, but would not initiate a premature RAS. With a tripped channel, an additional failure could cause a premature RAS, but would not disable the automatic RAS.

Since considerable time is available after initiation of SIS until RAS must be initiated, and since a premature RAS could damage the ESF pumps, it is preferable to bypass an inoperable channel and risk loss of automatic RAS than to trip a channel and risk a premature RAS.

The Completion Time of 8 hours allowed is reasonable because the Required Action involves a circuit modification.

Required Action C.2 requires that the inoperable channel be restored to OPERABLE status within 7 days. The Completion Time is reasonable based upon operating experience, which has demonstrated that a random failure of a second channel occurring during the 7 day period is a low probability event.

BASES

ACTIONS (continued)

D.1 and D.2

If the Required Actions and associated Completion Times of Condition A, B, or C are not met for Functions 1, 2, 3, 4, or 7, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1 and E.2

If the Required Actions and associated Completion Times of Condition A, B, or C are not met for Functions 5 or 6, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The SRs for any particular ESF Function are found in the SRs column of Table 3.3.3-1 for that Function. Most functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION.

SR 3.3.3.1

A CHANNEL CHECK is performed once every 12 hours on each ESF input channel which is provided with an indicator to provide a qualitative assurance that the channel is working properly and that its readings are within limits. A CHANNEL CHECK is not performed on the CHP and SIRWT Low Level channels because they have no associated control room indicator.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.1 (continued)

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

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the 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 verified to be reading at the bottom of the range and not failed downscale.

The Frequency of 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 displays associated with the LCO required channels.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.2

A CHANNEL FUNCTIONAL TEST is performed every 92 days to ensure the entire channel will perform its intended function when needed. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

This test is required to be performed each 92 days on ESF input channels provided with on-line testing capability. It is not required for the SIRWT Low Level channels since they have no built in test capability. The CHANNEL FUNCTIONAL TEST for SIRWT Low Level channels is performed each 18 months as part of the required CHANNEL CALIBRATION.

The CHANNEL FUNCTIONAL TEST tests the individual channels using an analog test input to each bistable.

Any setpoint adjustment shall be consistent with the assumptions of the current setpoint analysis.

The Frequency of 92 days is based on the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Reference 5).

SR 3.3.3.3

CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The Surveillance 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 between successive calibrations to ensure that the channel remains operational between successive surveillances. CHANNEL CALIBRATIONS must be performed consistent with the setpoint analysis.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.3 (continued)

The as found and as left values must also be recorded and reviewed for consistency with the assumptions of the extension analysis. The requirements for this review are outlined in Reference 5.

The Frequency is based upon the assumption of an 18 month calibration interval for the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. FSAR, Chapter 7
2. 10 CFR 50, Appendix A
3. IEEE Standard 279-1971
4. FSAR, Chapter 14
5. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989

B 3.3 INSTRUMENTATION

B 3.3.4 Engineered Safety Features (ESF) Logic and Manual Initiation

BASES

BACKGROUND

The ESF Instrumentation initiates necessary safety systems, based upon the values of selected plant parameters, to protect against violating core design limits and the Primary Coolant System (PCS) pressure boundary and to mitigate accidents.

The ESF circuitry generates the following signals listed below when the monitored variables reach levels that are indicative of conditions requiring protective action. The inputs to each ESF Actuation Signal are also listed.

1. Safety Injection Signal (SIS);
   a. Containment High Pressure (CHP)
   b. Pressurizer Low Pressure
2. Steam Generator Low Pressure Signal (SGLP)
   a. Steam Generator A Low Pressure
   b. Steam Generator B Low Pressure
3. Recirculation Actuation Signal (RAS);
   a. Safety Injection Refueling Water Tank (SIRWT) Low Level
4. Auxiliary Feedwater Actuation Signal (AFAS)
   a. Steam Generator A Low Level
   b. Steam Generator B Low Level

BASES

BACKGROUND (continued)

5. Containment High Pressure Signal (CHP);

   a. Containment High Pressure - Left Train

   b. Containment High Pressure - Right Train

6. Containment High Radiation Signal (CHR)

   a. Containment High Radiation

In the above list of actuation signals, the CHP and RAS are derived from pressure and level switches, respectively.

Equipment actuated by each of the above signals is identified in the FSAR, Chapter 7 (Ref. 1).

The ESF circuitry, with the exception of RAS, employs two-out-of-four logic. Four independent measurement channels are provided for each function used to generate ESF actuation signals. When any two channels of the same function reach their setpoint, actuating relays initiate the protective actions. Two separate and redundant trains of actuating relays, each powered from separate power supplies, are utilized. These separate relay trains operate redundant trains of ESF equipment. The actuation relays are considered part of the actuation logic addressed by this LCO.

RAS logic consists of output contacts of the relays actuated by the SIRWT Low Level switches arranged in a "one-out-of-two taken twice" logic. The contacts are arranged so that at least one low level signal powered from each station battery is required to initiate RAS. Loss of a single battery, therefore, cannot either cause or prevent RAS initiation.

The sensor subsystem, including individual channel bistables, is addressed in LCO 3.3.3, "Engineered Safety Features (ESF) Instrumentation." This LCO addresses the actuation subsystem manual actuation, and downstream components used to actuate the individual ESF functions, as defined in the following section.

BASES

BACKGROUND (continued)

ESF Logic

Each of the six ESF actuation signals in Table 3.3.4-1 operates two trains of actuating relays. Each train is capable of initiating the ESF equipment to meet the minimum requirements to provide all functions necessary to operate the system associated with the plant's capability to cope with abnormal events.

The SGLP logic circuitry includes provisions such that the SGLP automatic actuation Function may be bypassed if three-out-of-four Steam Generator (SG) pressure channels are below a bypass permissive setpoint. Similarly, the SIS automatic actuation on Pressurizer Low Pressure may be bypassed when three-out-of-four channels are below a permissive setpoint. This actuation bypassing is performed when the ESF Functions are no longer required for protection. These actuation bypasses are enabled manually when the permissive conditions are satisfied .

All actuation bypasses are automatically removed when enabling conditions are no longer satisfied. If an SIS or SGLP automatic actuation channel is bypassed, other than as allowed by Table 3.3.4-1, the channel cannot perform its required safety function and must be considered to be inoperable.

Testing of a major portion of the ESF circuits is accomplished while the plant is at power. More extensive sequencer and load testing may be done with the reactor shut down. The test circuits are designed to test the redundant circuits separately such that the correct operation of each circuit may be verified by either equipment operation or by sequence lights.

Manual Initiation

Manual ESF initiation capability is provided to permit the operator to manually actuate an ESF System when necessary.

Two control room mounted manual actuation switches are provided for SIS actuation, one for each train. Each SIS manual actuation switch affects one actuation channel, which actuates one train of SIS equipment.

There are no single manual controls provided to actuate CHP, however, CHP may be manually initiated using individual component controls.

BASES

BACKGROUND

Manual Initiation (continued)

Two control room mounted manual actuation switches are provided for CHR actuation, each switch affects both actuation channels, which actuates both CHR trains.

There are no single manual controls provided to actuate SGLP, however, SGLP may be manually initiated using individual component controls.

RAS is actuated by manually actuating the circuit "Test" switch, however, RAS may also be manually initiated using individual component controls.

Manual actuation of AFW may be accomplished through pushbutton actuation of each AFAS channel or by use of individual pump and valve controls. Each automatic AFAS actuation channel starts the AFW pumps in their starting sequence (if P-8A fails to start, a P-8C start signal is generated, and if P-8C also fails to start, a P-8B start signal is generated) and opens the associated flow control valves.

APPLICABLE SAFETY ANALYSES

Each of the analyzed accidents can be detected by one or more ESF Functions. One of the ESF Functions is the primary actuation signal for that accident. An ESF Function may be the primary actuation signal for more than one type of accident. An ESF Function may also be a secondary, or backup, actuation signal for one or more other accidents. Functions such as Manual Initiation, not specifically credited in the accident analysis, serve as backups to Functions and are part of the NRC staff approved licensing basis for the plant.

The manual initiation is not required by the accident analysis. The ESF logic must function in all situations where the ESF function is required (as discussed in the Bases for LCO 3.3.3).

Each ESF Function and its associated safety analyses are discussed in the Applicable Safety Analyses section of the Bases for LCO 3.3.3, ESF Instrumentation.

The ESF satisfies Criterion 3 of 10 CFR 50.36(c)(2).

BASES

LCO

The LCO requires that all components necessary to provide an ESF actuation be OPERABLE.

The Bases for the LCO on ESF automatic actuation Functions are addressed in LCO 3.3.3. Those associated with the Manual Initiation or Actuation Logic are addressed below.

ESF Logic and Manual Initiation Functions are required to be OPERABLE in MODES 1, 2, and 3, or in MODES 1, 2, 3, and 4, as appropriate, when the associated automatic initiation channels addressed by LCO 3.3.3 are required.

1. <u>Safety Injection Signal (SIS)</u>

SIS is actuated by manual initiation, by a CHP signal, or by two-out-of-four Pressurizer Low Pressure channels decreasing below the setpoint. Each Manual Initiation channel consists of one pushbutton which directly starts the SIS actuation logic for the associated train. Each SIS logic train is also actuated by a contact pair on one of the CHP initiation relays for the associated CHP train.

a. <u>Manual Initiation</u>

This LCO requires two channels of SIS Manual Initiation to be OPERABLE.

b. <u>Actuation Logic</u>

This LCO requires two channels of SIS Actuation Logic to be OPERABLE. Failures in the actuation subsystems are addressed in this LCO.

c. <u>CHP Logic Trains</u>

The CHP initiation relay (5P-x) input to the SIS logic is considered part of the SIS logic. Two channels, one per SIS train, must be OPERABLE.

BASES

LCO

1. Safety Injection Signal (SIS) (continued)

   d. Automatic Bypass Removal

      This LCO requires two channels of the automatic bypass removal logic for SIS Pressurizer Low Pressure to be OPERABLE. If an SIS automatic actuation channel is bypassed, other than as allowed by Table 3.3.4-1, the channel cannot perform its required safety function and must be considered to be inoperable.

      As indicated by footnote (a), the Pressurizer Low Pressure logic train for each SIS train can be bypassed when three-out-of-four channels indicate below 1700 psia. This bypass prevents undesired actuation of SIS during a normal plant cooldown. The bypass signal is automatically removed when two-out-of-four channels exceed the setpoint, in accordance with the philosophy of removing bypasses when the enabling conditions are no longer satisfied.

      The bypass permissive is set low enough so as not to be enabled during normal plant operation, but high enough to allow bypassing prior to reaching the trip setpoint.

2. Steam Generator Low Pressure Signal (SGLP)

   a. Manual Initiation

      This LCO requires two channels of SGLP Manual Initiation to be OPERABLE. As indicated by footnote (c), there is no manual control which actuates the SGLP logic circuits. The actuated components must be individually actuated using control room manual controls.

   b. Actuation Logic

      This LCO requires two channels of SGLP Actuation Logic to be OPERABLE, one for each SG.

BASES

LCO

2. Steam Generator Low Pressure Signal (SGLP) (continued)

c. Automatic Bypass Removal

This LCO requires two channels, one for each SG, of the SGLP automatic bypass removal logic to be OPERABLE. If an SIS automatic actuation channel is bypassed, other than as allowed by Table 3.3.4-1, the channel cannot perform its required safety function and must be considered to be inoperable.

As indicated by footnote (b), the SGLP from each SG may be bypassed when three-out-of-four channels indicate below 565 psia. This bypass prevents undesired actuation during a normal plant cooldown. The bypass signal is automatically removed when two-out-of-four channels exceed the setpoint, in accordance with the philosophy of removing bypasses when the enabling conditions are no longer satisfied.

The bypass permissive is set low enough so as not to be enabled during normal plant operation, but high enough to allow bypassing prior to reaching the trip setpoint.

3. Recirculation Actuation Signal (RAS)

a. Manual Initiation

This LCO requires two channels of RAS Manual Initiation to be OPERABLE. RAS is actuated by manually actuating the circuit "Test" switches.

b. Actuation Logic

This LCO requires two channels of RAS Actuation Logic to be OPERABLE.

BASES

LCO (continued)

4. Auxiliary Feedwater Actuation Signal (AFAS)

a. Manual Initiation

This LCO requires two channels of AFAS Manual Initiation to be OPERABLE. Each train of AFAS may be manually initiated with either of two sets of controls. Only one set of manual controls is required to be OPERABLE for each AFW train. One set of controls are the pushbuttons provided to actuate each train on the C-11 panel; the other set of controls are those manual controls provided on C-01 for each AFW pump and flow control valve.

b. Actuation Logic

This LCO requires two channels of AFAS Actuation Logic to be OPERABLE.

5. Containment High Pressure Signal (CHP)

a. Manual Initiation

As indicated by footnote (c), this LCO requires the manual controls necessary to actuate those valves and components actuated by an automatic CHP to be OPERABLE.

b. Actuation Logic

This LCO requires two channels of CHP Actuation Logic to be OPERABLE.

6. Containment High Radiation Signal (CHR)

a. Manual Initiation

This LCO requires two channels of CHR Manual Initiation to be OPERABLE. Pushbuttons are available for manual actuation of each CHR logic train.

BASES

LCO

6. Containment High Radiation Signal (CHR) (continued)

b. Actuation Logic

This LCO requires two channels of CHR Actuation Logic to be OPERABLE.

APPLICABILITY

ESF Functions are required to be OPERABLE in MODES 1, 2, and 3 or MODES 1, 2, 3, and 4 as specified in Table 3.3.4-1. In MODES 1, 2, and 3, there is sufficient energy in the primary and secondary systems to warrant automatic ESF System responses to:

- Close the MSIVs to preclude a positive reactivity addition and containment overpressure;
- Actuate AFW to preclude the loss of the steam generators as a heat sink (in the event the normal feedwater system is not available);
- Actuate ESF systems to prevent or limit the release of fission product radioactivity to the environment by isolating containment and limiting the containment pressure from exceeding the containment design pressure during a design basis LOCA or MSLB; and
- Actuate ESF systems to ensure sufficient borated inventory to permit adequate core cooling and reactivity control during a design basis LOCA or MSLB accident.

The CHP and CHR Functions are also required to be OPERABLE in MODE 4 to limit leakage of radioactive material from containment and limit operator exposure during and following a DBA.

The SGLP Function is not required to be OPERABLE in MODES 2 and 3, if all MSIVs are closed and deactivated and all MFRVs and MFRV bypass valves are either closed and deactivated or isolated by closed manual valves, since the SGLP Function is not required to perform any safety function under these conditions.

BASES

APPLICABILITY (continued)

In MODES 5 and 6, automatic actuation of ESF Functions is not required, because adequate time is available for plant operators to evaluate plant conditions and respond by manually operating the ESF components if required. In these MODES, ESF initiated systems are either reconfigured or disabled for shutdown cooling operation. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components.

ACTIONS

When the number of inoperable channels in a trip Function exceeds those specified in any related Condition associated with the same trip Function, then the plant is outside the safety analysis. Therefore, LCO 3.0.3 should be immediately entered, if applicable in the current MODE of operation.

A Note has been added to the ACTIONS to clarify the application of the Completion Time rules. The Conditions of this Specification may be entered independently for each Function in Table 3.3.4-1 in the LCO. Completion Times for the inoperable channel of a Function will be tracked separately.

A.1

Condition A applies to one Manual Initiation, Bypass Removal, or Actuation Logic channel inoperable. The channel must be restored to OPERABLE status to restore redundancy of the ESF Function. The 48 hour Completion Time is commensurate with the importance of avoiding the vulnerability of a single failure in the only remaining OPERABLE channel.

B.1 and B.2

If two Manual Initiation, Bypass Removal, or Actuation Logic channels are inoperable for Functions 1, 2, 3, or 4, or if the Required Action and associated Completion Time of Condition A cannot be met for Function 1, 2, 3, or 4, the reactor must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

C.1 and C.2

Condition C is entered when one or more Functions have two Manual Initiation or Actuation Logic channels inoperable for Functions 5 or 6, or when the Required Action and associated Completion Time of Condition A are not met for Functions 5 or 6. If Required Action A.1 cannot be met within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1

A functional test of each SIS actuation channel must be performed each 92 days. This test is to be performed using the installed control room test switches and test circuits for both "with standby power" and "without standby power". When testing the "with standby power" circuits, proper operation of the "SIS-X" relays must be verified; when testing the "without standby power" circuits, proper operation of the "DBA sequencer" and the associated logic circuit must be verified. The test circuits are designed to block those SIS functions, such as injection of concentrated boric acid, which would interfere with plant operation.

The Frequency of 92 days is based on plant operating experience.

SR 3.3.4.2

A CHANNEL FUNCTIONAL TEST of each AFAS Actuation Logic Channel is performed every 92 days to ensure the channel will perform its intended function when needed. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.4.2 (continued)

Instrumentation channel tests are addressed in LCO 3.3.3.

SR 3.3.4.2 addresses Actuation Logic tests of the AFAS using the installed test circuits.

The Frequency of 92 days for SR 3.3.4.2 is in agreement with the conclusions of the reliability analysis presented in topical report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation" (Ref. 2).

SR 3.3.4.3

A CHANNEL FUNCTIONAL TEST is performed on the manual ESF initiation channels, Actuation Logic channels, and bypass removal channels for specified ESF Functions. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

This Surveillance verifies that the required channels will perform their intended functions when needed.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at a Frequency of once every 18 months.

REFERENCES

1. FSAR, Chapter 7
2. CEN-327, June 2, 1986, including Supplement 1, March 3, 1989

B 3.3 INSTRUMENTATION

B 3.3.5 Diesel Generator (DG) - Undervoltage Start (UV Start)

BASES

BACKGROUND

The DGs provide a source of emergency power when offsite power is either unavailable or insufficiently stable to allow safe plant operation. Undervoltage protection will generate a UV Start in the event a Loss of Voltage or Degraded Voltage condition occurs. There are two UV Start Functions for each 2.4 kV vital bus.

Undervoltage protection and load shedding features for safety-related buses at the 2,400 V and lower voltage levels are designed in accordance with 10 CFR 50, Appendix A, General Design Criterion 17 (Ref. 1) and the following features:

1. Two levels of automatic undervoltage protection from loss or degradation of offsite power sources are provided. The first level (loss of voltage) provides normal loss of voltage protection. The second level of protection (degraded voltage) has voltage and time delay set points selected for automatic trip of the offsite sources to protect safety-related equipment from sustained degraded voltage conditions at all bus voltage levels. Coincidence logic is provided to preclude spurious trips.

2. The undervoltage protection system automatically prevents load shedding of the safety-related buses when the emergency generators are supplying power to the safeguards loads.

3. Control circuits for shedding of Class 1E and non-Class 1E loads during a Loss of Coolant Accident (LOCA) themselves are Class 1E or are separated electrically from the Class 1E portions.

BASES

BACKGROUND (continued)

Description

Each 2,400 V Bus (1C and 1D) is equipped with two levels of undervoltage protection relays (Ref. 2). The first level (Loss of Voltage Function) relays 127-1 and 127-2 are set at approximately 77% of rated voltage with an inverse time relay. One of these relays measures voltage on each of the three phases. They protect against sudden loss of voltage as sensed on the corresponding bus using a three-out-of-three coincidence logic. The actuation of the associated auxiliary relays will trip the associated bus incoming circuit breakers, start its associated DG, initiate bus load shedding, and activate annunciators in the control room. The DG circuit breaker is closed automatically upon establishment of satisfactory voltage and frequency by the use of associated voltage sensing relay 127D-1 or 127D-2.

The second level of undervoltage protection (Degraded Voltage Function) relays 127-7 and 127-8 are set at approximately 93% of rated voltage, with one relay monitoring each of the three phases. These relays protect against sustained degraded voltage conditions on the corresponding bus using a three-out-of-three coincidence logic. These relays have a built-in 0.65 second time delay, after which the associated DG receives a start signal and annunciators in the control room are actuated. If a bus undervoltage exists after an additional six seconds, the associated bus incoming circuit breakers will be tripped and a bus load shed will be initiated.

Trip Setpoints

The trip setpoints are based on the analytical limits presented in References 3 and 4, and justified in Reference 5. The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, and instrument drift, setpoints specified in SR 3.3.5.2 are conservatively adjusted with respect to the analytical limits. A detailed analysis of the degraded voltage protection is provided in References 3 and 4.

The specified setpoints will ensure that the consequences of accidents will be acceptable, providing the plant is operated from within the LCOs at the onset of the accident and the equipment functions as designed.

BASES

APPLICABLE SAFETY ANALYSES

The DG - UV Start is required for Engineered Safety Features (ESF) systems to function in any accident with a loss of offsite power. Its design basis is that of the ESF Systems.

Accident analyses credit the loading of the DG based on a loss of offsite power during a LOCA. The diesel loading has been included in the delay time associated with each safety system component requiring DG supplied power following a loss of offsite power. This delay time includes contributions from the DG start, DG loading, and Safety Injection System component actuation.

The required channels of UV Start, in conjunction with the ESF systems powered from the DGs, provide plant protection in the event of any of the analyzed accidents discussed in Reference 6, in which a loss of offsite power is assumed. UV Start channels are required to meet the redundancy and testability requirements of GDC 21 in 10 CFR 50, Appendix A (Ref. 1).

The delay times assumed in the safety analysis for the ESF equipment include the 10 second DG start delay and the appropriate sequencing delay, if applicable. The response times for ESFAS actuated equipment include the appropriate DG loading and sequencing delay.

The DG - UV Start channels satisfy Criterion 3 of 10 CFR 50.36(c)(2).

LCO

The LCO for the DG - UV Start requires that three channels per bus of each UV Start instrumentation Function be OPERABLE when the associated DG is required to be OPERABLE. The UV Start supports safety systems associated with ESF actuation.

The Bases for the trip setpoints are as follows:

The voltage trip setpoint is set low enough such that spurious trips of the offsite source due to operation of the undervoltage relays are not expected for any combination of plant loads and normal grid voltages.

BASES

LCO (continued)

This setpoint at the 2,400 V bus and reflected down to the 480 V buses has been verified through an analysis to be greater than the minimum allowable motor voltage (90% of nominal voltage). Motors are the most limiting equipment in the system. MCC contactor pickup and drop-out voltage is also adequate at the setpoint values. The analysis ensures that the distribution system is capable of starting and operating all safety-related equipment within the equipment voltage rating at the allowed source voltages. The power distribution system model used in the analysis has been verified by actual testing (Refs. 5 and 7).

The time delays involved will not cause any thermal damage as the setpoints are within voltage ranges for sustained operation. They are long enough to preclude trip of the offsite source caused by the starting of large motors and yet do not exceed the time limits of ESF actuation assumed in FSAR Chapter 14 (Ref. 6) and validated by Reference 8.

Calibration of the undervoltage relays verify that the time delay is sufficient to avoid spurious trips.

APPLICABILITY

The DG - UV Start actuation Function is required to be OPERABLE whenever the associated DG is required to be OPERABLE per LCO 3.8.1, "AC Sources - Operating," or LCO 3.8.2, "AC Sources - Shutdown," so that it can perform its function on a loss of power or degraded power to the vital bus.

ACTIONS

A DG - UV Start channel is inoperable when it does not satisfy the OPERABILITY criteria for the channel's Function.

In the event a channel's trip setpoint is found nonconservative with respect to the specified setpoint, or the channel is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition entered. The required channels are specified on a per DG basis.

BASES

ACTIONS (continued)

A.1

Condition A applies if one or more of the three phase UV sensors or relay logic is inoperable for one or more Functions (Degraded Voltage or Loss of Voltage) per DG bus.

The affected DG must be declared inoperable and the appropriate Condition(s) entered. Because of the three-out-of-three logic in both the Loss of Voltage and Degraded Voltage Functions, the appropriate means of addressing channel failure is declaring the DG inoperable, and effecting repair in a manner consistent with other DG failures.

Required Action A.1 ensures that Required Actions for the affected DG inoperabilities are initiated. Depending upon plant MODE, the actions specified in LCO 3.8.1 or LCO 3.8.2, as applicable, are required immediately.

SURVEILLANCE REQUIREMENTS

SR 3.3.5.1

A CHANNEL FUNCTIONAL TEST is performed on each UV Start logic channel every 18 months to ensure that the logic channel will perform its intended function when needed. The Undervoltage sensing relays are tested by SR 3.3.5.2. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

The Frequency of 18 months is based on the plant conditions necessary to perform the test.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.2

A CHANNEL CALIBRATION performed each 18 months verifies the accuracy of each component within the instrument channel. This includes calibration of the undervoltage relays and demonstrates that the equipment falls within the specified operating characteristics defined by the manufacturer.

The Surveillance 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 between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with the setpoint analysis.

The Frequency of 18 months is based on the plant conditions necessary to perform the test.

REFERENCES

1. 10 CFR 50, Appendix A GDCs 17 and 21
2. FSAR, Section 8.6
3. CPCo Analysis EA-ELEC-VOLT-033
4. CPCo Analysis EA-ELEC-VOLT-034
5. CPCo Analysis EA-ELEC-VOLT-17
6. FSAR, Chapter 14
7. CPCo Analysis EA-ELEC-VOLT-13
8. CPCo Analysis A-NL-92-111

B 3.3 INSTRUMENTATION

B 3.3.6 Refueling Containment High Radiation (CHR) Instrumentation

BASES

BACKGROUND

This LCO addresses Refueling CHR actuation. When the Refueling CHR Monitors are enabled by their keylock switches, a CHR actuation may be automatically initiated by a signal from either of the Refueling CHR monitors or manually by actuation of either of the control room "CHR Manual Initiate" pushbuttons (pushing either Manual Initiate pushbutton will actuate both trains of CHR). A CHR signal initiates the following actions:

a. Control Room HVAC Emergency Mode;

b. Containment Isolation Valve Closure; and

c. Block automatic starting of Engineered Safeguards pump room sump pumps.

The Refueling CHR signal provides automatic containment isolation valve closure during refueling operations, using two radiation monitors located in the refueling area of the containment (elevation 649 ft). The monitors are part of the plant area monitoring system and employ one-out-of-two logic for isolation. During normal operation these monitors are disconnected from the CHR relays and will not initiate a CHR signal. A switch is provided to connect the Refueling CHR monitors into the CHR actuation circuit, so that CHR actuation can be initiated by these monitors during refueling .

BASES

BACKGROUND (continued)

Each monitor actuates one train of CHR logic when containment radiation exceeds the setpoint. Two separate keylock switches, one per train, enable the Refueling CHR input to the CHR logic when switched to the "Refueling" position. Each Refueling CHR channel, associated keylock switch, and initiation circuit input to the CHR logic thus forms a one-out-of-one logic input to its associated CHR actuation logic train. The Refueling CHR isolation instrumentation is separate from the CHR instrumentation addressed in LCO 3.3.3, "ESF Instrumentation." However, the Refueling CHR Instrumentation does operate the same CHR actuation relays as the two-out-of-four CHR logic addressed in LCO 3.3.4. This LCO is not included in LCOs 3.3.3 and 3.3.4 because of the differences in APPLICABILITY and the single channel nature of the Refueling CHR input. The Refueling CHR signal performs the automatic containment isolation valve closure Function during refueling operations required by LCO 3.9.3, "Containment Penetrations."

The Refueling CHR Instrumentation provides protection from release of radioactive gases and particulates from the containment in the event a fuel assembly should be severely damaged during handling.

The Refueling CHR Instrumentation will detect any abnormal radiation levels in the containment refueling area and will initiate purge valve closure to limit the release of radioactivity to the environment. The containment purge supply and exhaust valves are closed on a CHR signal when a high radiation level in containment is detected.

The Refueling CHR Instrumentation includes two independent, redundant actuation subsystems, as described above. Reference 1 describes the Refueling CHR circuitry.

Trip Setpoint

No required setpoint is specified because these instruments are not assumed to function by any of the safety analyses. Typically, the instruments are set at about 25 mR/hr above expected background for planned operations (including movement of the reactor vessel head or internals).

BASES

APPLICABLE SAFETY ANALYSES	The Refueling CHR Instrumentation isolates containment in the event that area radiation exceeds an established level following a fuel handling accident. This ensures the radioactive materials are not released directly to the environment and significantly reduces the offsite doses from those calculated by the safety analyses, which do not credit containment isolation (Ref. 2). Either way, i.e., with or without containment isolation, the offsite doses remain within the guidelines of 10 CFR 100. The Refueling CHR Instrumentation is not required by the fuel handling accident analyses to maintain offsite doses within the guidelines of 10 CFR 100, but containment isolation would provide a significant reduction of the resulting offsite doses. Therefore, the Refueling CHR Instrumentation satisfies the requirements of Criterion 4 of 10 CFR 50.36(c)(2).
LCO	The LCO for the Refueling CHR Instrumentation requires that two channels of refueling CHR instrumentation and two channels of CHR manual initiation be OPERABLE, including the logic components necessary to initiate Refueling CHR Isolation. The CHR setpoint is chosen to be high enough to avoid inadvertent actuation in the event of normal background radiation fluctuations during fuel handling and movement of the reactor internals, but low enough to alarm and isolate the containment in the event of a Design Basis fuel handling accident.
APPLICABILITY	In MODE 5 or 6, the Refueling CHR isolation of containment isolation valves is not normally required to be OPERABLE. However, during CORE ALTERATIONS or during movement of irradiated fuel within containment, there is the possibility of a fuel handling accident requiring containment isolation on high radiation in containment. Accordingly, the Refueling CHR Instrumentation must be OPERABLE during CORE ALTERATIONS and when moving any irradiated fuel in containment. In MODES 1, 2, 3 and 4, both the Containment High Pressure (CHP) and CHR signals provide containment isolation as discussed in the Bases for LCO 3.3.3 and LCO 3.3.4.

BASES

ACTIONS

A.1, A.2.1, and A.2.2

Condition A applies to the failure of one Refueling CHR monitor channel, one CHR Manual Initiate channel, or one of each. The Required Action allows either initiation of a CHR signal by placing the inoperable channel in trip (which accomplishes the safety function of the inoperable channel), or suspension of CORE ALTERATIONS and movement of irradiated fuel assemblies within containment (which places the plant in a condition where the LCO does not apply). The Completion Time of 4 hours is acceptable because one additional channel of each Function remains operable during that period and the probability of an additional failure occurring during this period is very small.

The suspension of CORE ALTERATIONS and fuel movement shall not preclude completion of movement of a component to a safe position.

B.1 and B.2

Condition B applies when either no automatic Refueling CHR or no Manual CHR (or neither) is available. The Required Action is to immediately suspend CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. This places the plant in a condition where the LCO does not apply. The Completion Time is warranted on the basis that at least one containment isolation Function is completely lost.

The suspension of CORE ALTERATIONS and fuel movement shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.1 (continued)

Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or actual differing radiation levels at the two detector locations. CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

The Frequency, about once every shift, is based on operating experience that demonstrates the rarity of channel failure. Since the probability of two random failures in redundant channels in any 12 hour period is 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 3.3.6.2

A CHANNEL FUNCTIONAL TEST is performed on each Refueling CHR channel to ensure the entire channel will perform its intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

The Frequency of 31 days is based on plant operating experience with regard to channel OPERABILITY, which demonstrates that failure of more than one channel of a given Function in any 31 day interval is a rare event.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.3

A CHANNEL FUNCTIONAL TEST is performed on each CHR Manual Initiation channel to ensure it will perform its intended function.

The Frequency of 18 months is based on plant operating experience with regard to channel OPERABILITY, and is consistent with the testing of other manually actuated functions.

SR 3.3.6.4

A CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The Surveillance 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 between successive calibrations to ensure that the channel remains operational between successive tests.

No required setpoint is specified because these instruments are not assumed to function by any of the safety analyses.

The Frequency is based upon the assumption of an 18 month calibration interval in the setpoint determination.

REFERENCES

1. FSAR, Section 7.3
2. FSAR, Section 14.19

B 3.3 INSTRUMENTATION

B 3.3.7 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the Post Accident Monitoring (PAM) instrumentation is to display plant variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety Functions for Design Basis Events.

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

The availability of PAM instrumentation is important so that responses to corrective actions can be observed and the need for, and magnitude of, further actions can be determined. The required instruments are identified in FSAR Appendix 7C (Ref. 1) and address the recommendations of Regulatory Guide 1.97 (Ref. 2), as required by Supplement 1 to NUREG-0737, "TMI Action Items" (Ref. 3).

Type A variables are included in this LCO because they provide the primary information required to permit the control room operator to take specific manually controlled actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs).

Category I variables are the key variables deemed risk significant because they are needed to:

- Determine whether other 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 for an estimate of the magnitude of any impending threat.

BASES

BACKGROUND (continued)

These key variables are identified in the plant specific Regulatory Guide 1.97 analyses (Ref. 1). This analysis identified the plant specific Type A and Category 1 variables and provided justification for deviating from the NRC proposed list of Category I variables.

The specific instrument Functions listed in Table 3.3.7-1 are discussed in the LCO Bases.

APPLICABLE SAFETY ANALYSES

The PAM instrumentation ensures the OPERABILITY of Regulatory Guide 1.97 Type A variables, so that the control room operating staff can:

- Perform the diagnosis specified in the emergency operating procedures. These variables are restricted to preplanned actions for the primary success path of DBAs; and
- Take the specified, preplanned, manually controlled actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions.

The PAM instrumentation also ensures OPERABILITY of Category I, non-Type A variables. This ensures the control room operating staff can:

- 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 as well as to obtain an estimate of the magnitude of any impending threat.

Category I, non-Type A PAM instruments are retained in the Specification because they are intended to assist operators in minimizing the consequences of accidents. Therefore, these Category I variables are important in reducing public risk.

PAM instrumentation that satisfies the definition of Type A in Regulatory Guide 1.97 meets Criterion 3 of 10 CFR 50.36(c)(2).

BASES

LCO

LCO 3.3.7 requires at least two OPERABLE channels for all Functions except Containment Isolation Valve Position Indication. This is to ensure no single failure prevents the operators from being presented with the information necessary to determine the status of the plant and to bring the plant to, and maintain it in, a safe condition following that accident.

Furthermore, provision of at least two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information.

For Containment Isolation Valve Position indication, the important information is the status of the containment penetrations. The LCO requires one position indication channel for each containment isolation valve listed in FSAR Appendix 7C (Ref. 1).

Listed below are discussions of the specified instrument Functions listed in Table 3.3.7-1. Component identifiers of the sensors, indicators, power supplies, displays, and recorders in each instrument loop are found in Reference 1.

1, 2. <u>Primary Coolant System (PCS) Hot and Cold Leg Temperature (wide range)</u>

PCS wide range Hot and Cold Leg Temperatures are Type B, Category 1 variables provided for verification of core cooling and long term surveillance.

Reactor outlet temperature inputs to the PAM are provided by two wide range resistance elements and associated transmitters (one in each loop). The channels provide indication over a range of 50°F to 700°F.

3. <u>Wide Range Neutron Flux</u>

Wide Range Neutron Flux indication is a Type B, Category 1 variable, and is provided to verify reactor shutdown.

4. <u>Containment Floor Water Level (wide range)</u>

Wide range Containment Floor Water Level is a Type B, Category 1 variable, and is provided for verification and long term surveillance of PCS integrity.

BASES

LCO (continued)

5. Subcooled Margin Monitor

The Subcooled Margin Monitor (SMM) is a Type A, Category 1 variable used to identify conditions which require tripping of the primary coolant pumps and throttling of safety injection flows. Each SMM channel uses a number of PCS pressure and temperature inputs to determine the degree of PCS subcooling or superheat.

6. Pressurizer Level (Wide Range)

Pressurizer Level is a Type A, Category 1 variable, and is used 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 plant conditions necessary to establish natural circulation in the PCS and to verify that the plant is maintained in a safe shutdown condition.

7. Containment Hydrogen Monitors

Containment Hydrogen Monitors are provided to detect high hydrogen concentration conditions (a Type A, Category 1 variable) that represent a potential for containment breach and are used to determine when to place the hydrogen recombiners in operation. This variable is also important in verifying the adequacy of mitigating actions.

8. Condensate Storage Tank (CST) Level

CST Level is a Type D, Category 1 variable, and is provided to ensure water supply for AFW. The CST provides the safety grade water supply for the AFW System. Inventory is monitored by a 0 to 100% level indication. CST Level is displayed on a control room indicator. In addition, a control room annunciator alarms on low level.

The CST is the initial source of water for the AFW System. However, as the CST is depleted, manual operator action is necessary to replenish the CST.

BASES

LCO (continued)

9. Primary Coolant System Pressure (wide range)

PCS wide range pressure is a Type A, Category 1 variable provided for verification of core cooling and PCS integrity long term surveillance.

Wide range PCS loop pressure is measured by pressure transmitters with a span of 0 psia to 3000 psig. Redundant monitoring capability is provided by two channels of instrumentation. Control room indications are provided on C12 and C02.

10. Containment Pressure (wide range)

Wide range Containment Pressure is a Type C, Category 1 variable, and is provided for verification of PCS and containment OPERABILITY. It is also an input to decisions for initiating containment spray.

11, 12. Steam Generator Water Level (wide range)

Wide range Steam Generator Water Level is a Type A, Category 1 variable, and is provided to monitor operation of decay heat removal via the steam generators. The steam generator level instrumentation covers a span extending from the tube sheet to the steam separators, with an indicated range of -140% to +150%. Redundant monitoring capability is provided by two channels of instrumentation for each SG.

Operator action for maintenance of heat removal is based on the control room indication of Steam Generator Water Level. The indication is used during a SG tube rupture to determine which SG has the ruptured tube. It is also used to determine when to initiate once through cooling on low water level.

13, 14. SG Pressure

Steam Generator Pressure is a Type A, Category 1 variable used in accident identification, including Loss of Coolant, and Steam Line Break. Redundant monitoring capability is provided by two channels of instrumentation for each SG.

BASES

LCO (continued)

15. Containment Isolation Valve Position

Containment Isolation Valve (CIV) Position is a Type B, Category 1 variable, and is provided for verification of containment OPERABILITY.

CIV position is provided for verification of containment integrity. In the case of CIV position, the important information is the isolation status of the containment penetration. The LCO requires one channel of valve position indication in the control room to be OPERABLE for each CIV listed in FSAR Appendix 7C (Ref. 1). This is sufficient to redundantly verify the isolation status of each associated penetration via indicated status of the CIVs, and by knowledge of a passive (check) valve or a closed system boundary.

If a penetration flow path is isolated, position indication for the CIV(s) in the associated penetration flow path is not needed to determine status. Therefore, as indicated in Note (a) the position indication for valves in an isolated penetration flow path is not required to be OPERABLE.

16, 17, 18, 19. Core Exit Temperature

Core Exit Temperature is a Type C, Category 1 variable, and is provided for verification and long term surveillance of core cooling.

Each Required Core Exit Thermocouple (CET) channel consists of a single environmentally qualified thermocouple.

The design of the Incore Instrumentation System includes a Type K (chromel alumel) thermocouple within each of the incore instrument detector assemblies.

The junction of each thermocouple is located above the core exit, inside the incore detector assembly guide tube, that supports and shields the incore instrument detector assembly string from flow forces in the outlet plenum region. These core exit thermocouples monitor the temperature of the reactor coolant as it exits the fuel assemblies.

The core exit thermocouples have a usable temperature range from 32°F to 2300°F, although accuracy is reduced at temperatures above 1800°F.

BASES

LCO (continued)

20. Reactor Vessel Water Level

Reactor Vessel Water Level is monitored by the Reactor Vessel Level Monitoring System (RVLMS) and is a Type B, Category 1 variable provided for verification and long term surveillance of core cooling.

The RVLMS provides a direct measurement of the collapsed liquid level above the fuel alignment plate. The collapsed level represents the amount of liquid mass that is in the reactor vessel above the core. Measurement of the collapsed water level is selected because it is a direct indication of the water inventory. The collapsed level is obtained over the same temperature and pressure range as the saturation measurements, thereby encompassing all operating and accident conditions where it must function. Also, it functions during the recovery interval. Therefore, it is designed to survive the high steam temperature that may occur during the preceding core recovery interval.

The level range extends from the top of the vessel down to the top of the fuel alignment plate. A total of eight Heated Junction Thermocouple (HJTC) pairs are employed in each of the two RVLMS channels. Each pair consists of a heated junction TC and an unheated junction TC. The differential temperature at each HJTC pair provides discrete indication of uncovery at the HJTC pair location. This indication is displayed using LEDs in the control room. 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.

A RVLMS channel consists of eight sensors in a probe. A channel is OPERABLE if four or more sensors, two or more of the upper four and two or more of the lower four, are OPERABLE.

21. Containment Area Radiation (high range)

High range Containment Area Radiation is a Type E, Category 1 variable, and is provided to monitor for the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans.

BASES

APPLICABILITY

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

ACTIONS

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

Note 2 has been added in the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.7-1. The Completion Time(s) of the inoperable channel(s) of a Function will be tracked separately for each Function, starting from the time the Condition was entered for that Function.

A.1

When one or more Functions have one required channel that is inoperable, the required inoperable channel must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience and takes into account the remaining OPERABLE channel, the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

BASES

ACTIONS (continued)

B.1

This Required Action specifies initiation of actions in accordance with Specification 5.6.6, which requires a written report to be submitted to the Nuclear Regulatory Commission. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative Required Actions. This Required Action is appropriate in lieu of a shutdown requirement, given the likelihood of plant conditions that would require information provided by this instrumentation. Also, alternative Required Actions are identified before a loss of functional capability condition occurs.

C.1

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. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrumentation operation and the availability of alternate means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur.

Condition C is modified by a Note which indicates it is not applicable to hydrogen monitor channels.

D.1

Condition D applies when two hydrogen monitor channels are inoperable. Required Action D.1 requires restoring one hydrogen monitor channel to OPERABLE status within 72 hours. The 72 hour Completion Time is reasonable based on other core damage assessment capabilities available to provide information for operator decisions. Also, it is unlikely that a LOCA (which would cause core damage) would occur during this time.

BASES

ACTIONS (continued)

E.1

This Required Action directs entry into the appropriate Condition referenced in Table 3.3.7-1. The applicable Condition referenced in the Table is Function dependent. Each time Required Action C.1 or D.1 is not met, and the associated Completion Time has expired, Condition E is entered for that channel and provides for transfer to the appropriate subsequent Condition.

F.1 and F.2

If the Required Action and associated Completion Time of Condition C or D are not met, and Table 3.3.7-1 directs entry into Condition F, the plant must be brought to a MODE in which the requirements of this LCO do not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 30 hours.

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

G.1

Alternate means of monitoring Reactor Vessel Water Level and Containment Area Radiation have been developed and tested. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. If these alternate means are used, the Required Action is not to shut down the plant, but rather to follow the directions of Specification 5.6.6. 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.

BASES

SURVEILLANCE REQUIREMENTS

A Note at the beginning of the Surveillance Requirements specifies that the following SRs apply to each PAM instrumentation Function in Table 3.3.7-1.

SR 3.3.7.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. 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 verified to be reading at the bottom of the range and not failed downscale.

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

The Frequency of 31 days is based upon plant 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 CHANNEL CHECK supplements less formal, but more frequent, checks of channel during normal operational use of the displays associated with this LCO's required channels.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.2

A CHANNEL CALIBRATION is performed every 18 months or approximately every refueling. CHANNEL CALIBRATION is typically a complete check of the instrument channel including the sensor. Therefore, this SR is modified by a Note which states that it is not necessary to calibrate neutron detectors because of the difficulty of simulating a meaningful signal. Wide range and source range nuclear instrument channels are not calibrated to indicate the actual power level or the flux in the detector location. The circuitry is adjusted so that wide range and source range readings may be used to determine the approximate reactor flux level for comparative purposes. The Surveillance verifies the channel responds to the measured parameter within the necessary range and accuracy.

For the core exit thermocouples, a CHANNEL CALIBRATION is performed by substituting a known voltage for the thermocouple.

The Frequency is based upon operating experience and consistency with the typical industry refueling cycle and is justified by an 18 month calibration interval for the determination of the magnitude of equipment drift.

REFERENCES

1. FSAR, Appendix 7C, "Regulatory Guide 1.97 Instrumentation"

2. Regulatory Guide 1.97

3. NUREG-0737, Supplement 1