ITS REVIEW PACKAGE CONTENTS

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3.4

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PACKAGE 3.4

REACTOR COOLANT SYSTEM (RCS)

PART A

INTRODUCTION

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

LICENSE AMENDMENT REQUEST DATED December 11, 2000
Conversion to Improved Standard Technical Specifications

3.4
PART A

Introduction to the Discussion of the proposed Changes to the Current Technical Specifications, Justification of Differences from the Improved Standard Technical Specifications, and the supporting No Significant Hazards Determination

Pursuant to 10 CFR Part 50, Sections 50.59 and 50.90, the holders of Operating Licenses DPR-42 and DPR-60 hereby propose changes to the Facility Operating Licenses and Appendix A, Technical Specifications, as follows and as presented in the accompanying Parts B through G of this Package.

BACKGROUND

Over the past several years the nuclear industry and the Nuclear Regulatory Commission (NRC) have jointly developed Improved Standard Technical Specifications (ISTS). The NRC has encouraged licensees to implement these improved technical specifications as a means for improving plant safety through the more operator-oriented technical specifications, improved and expanded bases, reduced action statement induced plant transients, and more efficient use of NRC and industry resources.

This License Amendment Request (LAR) is submitted to conform the Prairie Island Nuclear Generating Plant (PINGP) Current Technical Specifications (CTS) to NUREG-1431, Improved Standard Technical Specifications, Westinghouse plants, Revision 1 issued April 1995 (ISTS). The resulting new Technical Specifications (TS) for Prairie Island (PI) are the PI Improved Technical Specifications (ITS) which incorporates the PI plant specific information.

NUREG-1431 is based on a hypothetical four loop Westinghouse plant. Since PI is similar in design and vintage to the R.E. Ginna Nuclear Power Plant which has already completed conversion to improved technical specifications, this amendment request relies on the Ginna ITS.

This LAR is also supported by Parts B through G. Part B contains a "clean" copy of the proposed PI ITS and Bases. Part C contains a mark-up of the PI CTS. Part D is the Description of Changes (DOC) to the PI CTS. Part E is a mark-up of the ISTS and Bases which shows the deviations from the standard incorporated to meet PI plant specific requirements. Part F gives the Justification for Deviations (JFD) from the ISTS and Part G provides the No Significant Hazards Determinations (NSHD) for changes to the PI CTS. To facilitate review of this LAR, cross-reference numbers from changes and deviations to the corresponding DOC, JFD and NSHD are provided. The methodology for mark-up and cross-references are described in the next section.

MARK-UP METHODOLOGY

The TS conversion package includes mark-ups of the CTS, the ISTS and the ISTS Bases in accordance with this guidance. Mark-up may be electronic or by hand as indicated.

Current Technical Specifications

The mark-up of the CTS is provided to show where current requirements are placed in the ITS, to show the major changes resulting from the conversion process, and to allow reviewers to evaluate significant differences between the CTS and ITS.

This ITS conversion LAR has been prepared in 14 packages following the Chapter/Section outline of the ITS as follows: 1.0, 2.0, 3.0, 3.1 . . . 3.9, 4.0 and 5.0. Accordingly, each package contains all the elements of Parts A through G as described above. The CTS Bases are not included in the CTS mark-up packages since the Bases have been rewritten in their entirety.

The current Specifications addressed by the associated ITS Chapter/Section are cross-referenced in the left margin to the new ITS location by Specification number and type (G-General, SL-Safety Limit, LCO-Limiting Condition for Operation or SR-Surveillance Requirements). Those portions of each CTS page which are not addressed in the associated ITS Chapter/Section are shadowed (electronic) or clouded and crossed out (by hand) and in the right margin is the comment, “Addressed Elsewhere”.

The CTS are marked-up to incorporate the substance of NUREG-1431 Revision 1. It is not the intent to mark every nuance required to make the format change from CTS to ITS.

In general, only technical changes have been identified. However, some non-technical changes have also been included when the changes cannot easily be determined to be non-technical by a reviewer, or if an explanation is required to demonstrate that the change is non-technical.

Some apparent changes result from the different conventions and philosophies used in the ITS. Generally these apparent changes will not be marked-up in the CTS if there is no resulting change in plant operating requirements.

Changes are identified by a change number in the right margin which map the changed specification requirement to Part D, Discussion of Changes, and Part G, No Significant Hazards Determination (NSHD) and indicate the NSHD category. The change number form is R3.4-02 where the first two numbers, 3.4 in this example, refer to ITS Chapter/Section number 3.4, and the second number, 02 in this example, is a sequentially assigned number for changes within that Chapter/Section, starting with 01. The prefix letter(s) indicates the classification of the change impact. For CTS changes this is also the NSHD category.

The change impact categories defined below conveniently group the type of changes for consideration of the effect of the change on the current plant license in Part D and are also useful for efficient discussion in Part G the "No Significant Hazards Determination" (NSHD) section. If the same change is made in Part E, then the change impact category will also show up in the change number in Part F. These categories are:

A - Administrative changes, editorial in nature that do not involve technical issues. These include reformatting, renaming (terminology changes), renumbering, and rewording of requirements.

L - Less restrictive requirements included in the PI ITS in order to conform to the guidance of NUREG-1431. Generally these are technical changes to existing TS which may include items such as extending Completion Times or reducing Surveillance Frequencies (extended time interval between surveillances). The less restrictive requirements necessitate individual justification. Each is provided with its specific NSHD.

LR - Less restrictive Removal of details and information from otherwise retained specifications which are removed from the CTS and placed in the Bases, Technical Requirements Manual (TRM), Updated Safety Analysis Report (USAR) or other licensee controlled documents. These changes include details of system design and function, procedural details or methods of conducting surveillances, or alarm or indication-only instrumentation.

M - More restrictive requirements included in the PI ITS in order to provide a complete set of Specifications conforming to the guidance of NUREG-1431. Changes in this category may be completely new requirements or they may be technical changes made to current requirements in the CTS.

R - Relocation of Current Specifications to other controlled documents or deletion of current Specifications which duplicate existing regulatory requirements.

Current requirements in the LCOs or SRs that do not meet the 10 CFR 50.36 selection criteria and may be relocated to the Bases, USAR, Core Operating Limits Report (COLR), Operational Quality Assurance Plan (OQAP), plant procedures or other licensee controlled documents. Relocating requirements to these licensee controlled documents does not eliminate the requirement, but rather, places them under more appropriate regulatory controls, such as 10CFR 50.54 (a)(3) and 10 CFR 50.59, to manage their implementation and future changes. Maintenance of these requirements in the TS commands resources which are not commensurate with their importance to safety and distract resources from more important requirements. Relocation of these items will enable more efficient maintenance of requirements under existing regulations and reduce the need to request TS changes for issues which do not affect public safety.

Deletion of Specifications which duplicate regulations eliminates the need to change Technical Specifications when changes in regulations occur. By law, licensees shall meet applicable requirements contained in the Code of Federal Regulations, or have NRC approved exemptions; therefore, restatement in the Technical Specifications is unnecessary.

The methodology for marking-up these changes is as follows:

As discussed above, administrative changes may not be marked-up in detail. Portions of the specifications which are no longer included are identified by use of the electronic strike-out feature (or crossed out by hand). Information being added is inserted into the specification in the appropriate location and is identified by use of shading features (or handwritten/insert pages).

Improved Standard Technical Specifications (NUREG-1431, Rev. 1)

The ISTS mark-up is to identify changes from the ISTS required to create a plant specific ITS by incorporating plant specific values in bracketed fields and identifying other changes with cross-reference to the Part F Justification For Differences.

All deviations from the ISTS are cross-referenced to the Part F justification for differences by a change number in the right margin. The change number form is CL3.4-05 where the prefix letter(s), CL in this example, indicate the classification of the reason for the difference, the first two numbers, 3.4 in this example, refer to the ITS Chapter/Section number 3.4, and the second number, 05 in this example, is a sequentially assigned number for deviations within that Chapter/Section, starting with a number which is larger than the last number from the Part C CTS mark-up. In some instances where a change has been made to the CTS and ISTS, the Part D change number is given since the justification for difference is the same as the discussion of change. The following categories are used as prefixes to indicate the general reason for each difference:

CL - Current Licensing basis. Issues that have been previously licensed for PI and have been retained in the ITS. This includes Specifications dictated by plant design features or the design basis. Since no plant modifications have been or will be made to accommodate conversion to ITS, the plant design basis features shall be incorporated into the PI ITS.

PA - Plant, Administrative. Plant specific wording preference or minor editorial improvements made to facilitate operator understanding.

TA - Traveler, Approved. Deviations made to incorporate an industry traveler which has been approved by the NRC.

TP - Traveler, Proposed. Deviation made to incorporate a proposed industry traveler which as of the time of submittal has not been approved by the NRC.

X - Other, Deviation from the ISTS for any other reason than those given above.

Material which is deleted from the ISTS is identified by use of the WordPerfect strike-out feature (or crossed out by hand). Information being added to the ISTS to generate the PI ITS due to any of the deviations discussed above is identified by use of WordPerfect red-line features (or handwritten/insert pages).

Bracketed Information

Many parameters, conditions, notes, surveillances, and portions of sections are bracketed in the ISTS recognizing that plant specific values are likely to vary from the "generic" values provided in the standard.

If the bracketed value applies to PI, then the "generic" information is retained without any special indication and the brackets are marked using the WordPerfect strike-out feature. In some instances, bracketed material is not discussed. If bracketed material is discussed, a change number is provided which includes the appropriate prefix as described above. When bracketed "generic" material is not incorporated, the bracketed material and brackets are marked with the WordPerfect strike-out feature (or crossed out by hand), the plant specific information is substituted for the bracketed information and a change number is provided which includes the appropriate prefix. Information added is indicated by the WordPerfect red-line (shading) feature (or handwritten/insert pages).

Optional Sections

Due to differing Westinghouse plant designs and methodologies, some ISTS section numbers include a letter suffix indicating that only one of these sections is applicable to any specific plant. The appropriate section is indicated in the Table of Contents, the suffix letter is deleted, and justification, if required, is included in the appropriate Chapter/Section package.

Bases, Improved Standard Technical Specifications (NUREG-1431, Rev. 1)

The ISTS Bases have been marked-up to support the plant specific PI ITS and allow reviewers to identify changes from NUREG-1431. To the extent possible, the words of NUREG-1431, Rev. 1 are retained to maximize standardization. Where the existing words in the NUREG are incorrect or misleading with respect to Prairie Island, they have been revised. In addition, descriptions have been added to cover plant specific portions of the specifications. Change numbers have been provided for the ISTS Bases with the same format as the ISTS Specification mark-up. In some instances, the same change number is used to describe the change.

Material which is deleted from the ISTS Bases is identified by use of the strike-out feature of WordPerfect (or crossed out by hand). Information being added to the ISTS Bases to generate the PI ITS is identified by use of the red-line (shading) feature of WordPerfect (or handwritten/insert pages).

Bracketed Material

Many parameters and portions of Bases are bracketed in the ISTS recognizing that plant specific values and discussions are likely to vary from the "generic" information provided in the standard.

If the bracketed information applies to PI, then the "generic" information is retained without any special indication and the brackets are marked using the WordPerfect strike-out feature. No change number or justification is provided for use of bracketed material, unless special circumstances warrant discussion.

When bracketed "generic" Bases material is not incorporated, the bracketed material and brackets are marked with the WordPerfect strike-out feature (or crossed out by hand) and the plant specific information substituted for the bracketed information is indicated by the WordPerfect red-line (shading) feature (or handwritten/insert pages). A change number with the same format as those used for the ISTS Specification mark-up is provided.

ACRONYMS

Many acronyms are used throughout this submittal. The intent of the final ITS (Part B) is that in general acronyms be written in full prior to the first use. Commonly used acronyms may not be written in full. Other parts of this package may not always write in full each acronym prior to first use; therefore, a list of acronyms is attached to assist in the review of this package.

Attachment to Part A

LIST OF ACRONYMS

AB	Auxiliary Building
ABSVS	Auxiliary Building Special Ventilation System
AFD	Axial Flux Difference
AFW	Auxiliary Feedwater System
ALARA	As Low As Reasonably Achievable
ALT	Actuation Logic Test
ASA	Applicable Safety Analyses
ASME	American Society of Mechanical Engineers
AOO	Anticipated Operational Occurrences
AOT	Allowed Outage Time
BAST	Boric Acid Storage Tank
BIT	Boron Injection Tank
BOC	Beginning of Cycle
CC	Component Cooling
COT	CHANNEL OPERATIONAL TEST
CAOC	Constant Axial Offset Control
CET	Core Exit Thermocouple
CL	Cooling Water
CLB	Current Licensing Basis
COLR	Core Operating Limits Reports
CRDM	Control Rod Drive Mechanism
CRSVS	Control Room Special Ventilation System
CS	Containment Spray
CST	Condensate Storage Tanks
CTS	Current Technical Specification(s)
DBA	Design Basis Accident
DDCL	Diesel Driven Cooling Water
DG	Diesel Generator
DNB	Departure from Nucleate Boiling
DNBR	Departure from nucleate boiling ratio
ECCS	Emergency Core Cooling System

EDG	Emergency Diesel Generators
EFPD	Effective Full Power Days
EOC	End of Cycle
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Features Actuation System
FWLB	Feedwater Line Break
GDC	General Design Criteria
GITS	Ginna Improved Technical Specifications
HELB	High Energy Line Break
HZP	Hot Zero Power
IPE	Individual Plant Evaluation
ISTS	Improved Standard Technical Specifications
ITC	Isothermal Temperature Coefficient
ITS	Improved Technical Specifications
LA	License Amendment
LAR	License Amendment Request
LBLOCA	Large Break LOCA
LCO	Limiting Conditions for Operation
LHR	Linear Heat Rate
LOCA	Loss of Coolant Accident
LTOP	Low Temperature Overpressure Protection
MFIV	Main Feedwater Isolation Valve
MFRV	Main Feedwater Regulation Valve
MFW	Main Feedwater
MOSCA	MODE or Other Specified Condition of Applicability
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valves
MSLB	Main Steam Line Break
MSLI	Main Steam Line Isolation
MSSV	Main Steam Safety Valves
MTC	Moderator Temperature Coefficient
NIS	Nuclear Instrumentation System
NMC	Nuclear Management Company
NPSH	Net Positive Suction Head

NRCV	Non-Return Check Valve
NUREG-1431	The ISTS for Westinghouse plants
OPPS	OverPressure Protection System
PCT	Peak Cladding Temperature
PI	Prairie Island
PITS	Prairie Island Technical Specifications
PIV	Pressure Isolation Valve
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
PSV	Pressurizer Safety Valve
PTLR	Pressure and Temperature Limits Report
QTPR	Quadrant Power Tilt Ratio
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RHR	Residual Heat Removal System
RPI	Rod Position Indication
RPS	Reactor Protection System
RTB	Reactor Trip Breaker
RTBB	Reactor Trip Bypass Breaker
RTP	Rated Thermal Power
RTS	Reactor Trip System
RWST	Refueling Water Storage Tank
SBLOCA	Small Break Loss of Coolant Accident
SBVS	Shield Building Ventilation System
SCWS	Safeguards Chilled Water System
SDM	Shut Down Margin
SFDP	Safety Function Determination Program
SFP	Spent Fuel Pool
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SL	Safety Limit

SLB	Steam Line Break
SR	Surveillance Requirements
SSC	Structures, Systems and Components
TADOT	Trip Actuating Device Operational Test
TDAFW	Turbine Driven Auxiliary Feedwater
TRM	Technical Requirements Manual
TS	Technical Specifications
TSSC	Technical Specification Selection Criteria
TSTF	Term used for a NUREG change (traveler)
VCT	Volume Control Tank
VFTP	Ventilation Filter Test Program
UHS	Ultimate Heat Sink
USAR	Updated Safety Analysis Report
WCAP	Westinghouse technical report

PACKAGE 3.4

REACTOR COOLANT SYSTEM (RCS)

PART B

PROPOSED PRAIRIE ISLAND IMPROVED TECHNICAL SPECIFICATIONS AND BASES

List of Pages

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNITS 1 AND 2

Improved Technical Specifications Conversion Submittal

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow - Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ≥ the limit specified in the COLR;

b. RCS average temperature ≤ the limit specified in the COLR; and

c. RCS total flow rate ≥ the value specified in the COLR.

APPLICABILITY: MODE 1.

-----------------------------------NOTE-----------------------------------
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is greater than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average temperature is less than or equal to the limit specified in the COLR.	12 hours
SR 3.4.1.3	-------------------------NOTE------------------------- Required to be performed within 7 days after ≥ 90% RTP. -- Verify RCS total flow rate is within the limit specified in the COLR.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 Each RCS loop average temperature (T_{avg}) shall be ≥ 540°F.

APPLICABILITY: MODE 1,
MODE 2 with $k_{eff} \geq 1.0$.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} in one or more RCS loops not within limit.	A.1 Be in MODE 2 with $k_{eff} < 1.0$.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.2.1	Verify RCS T_{avg} in each loop ≥ 540°F.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -----------NOTE----------- Required Action A.2 shall be completed whenever this Condition is entered. ------------------------------ Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5 with RCS pressure < 500 psig.	6 hours 36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. -----------NOTE----------- Required Action C.2 shall be completed whenever this Condition is entered. ------------------------------ Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.	C.1 Initiate action to restore parameter(s) to within limits.	Immediately
	AND	
	C.2 Determine RCS is acceptable for continued operation.	Prior to entering MODE 4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	----------------------NOTE---------------------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ---	
	Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR.	30 minutes

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops - MODES 1 and 2

LCO 3.4.4 Two RCS loops shall be OPERABLE and in operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of LCO not met.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify each RCS loop is in operation.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops - MODE 3

LCO 3.4.5 Two RCS loops shall be OPERABLE, and either:

a. Two RCS loops shall be in operation when the Rod Control System is capable of rod withdrawal; or

b. One RCS loop shall be in operation when the Rod Control System is not capable of rod withdrawal.

-------------------------------------NOTE-------------------------------------

Both reactor coolant pumps may not be in operation for ≤ 12 hours to perform preplanned work activities provided:

a. No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature.

APPLICABILITY: MODE 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RCS loop inoperable.	A.1 Restore inoperable RCS loop to OPERABLE status.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	12 hours
C. One RCS loop not in operation with Rod Control System capable of rod withdrawal.	C.1 Restore required RCS loop to operation.	1 hour
	OR	
	C.2 Place the Rod Control System in a condition incapable of rod withdrawal.	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two RCS loops inoperable. OR No RCS loop in operation.	D.1 Place the Rod Control System in a condition incapable of rod withdrawal.	Immediately
	AND	
	D.2 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	AND	
	D.3 Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required RCS loops are in operation.	12 hours
SR 3.4.5.2	Verify steam generator secondary side water levels are ≥ 60% (Wide Range) for both RCS loops.	12 hours
SR 3.4.5.3	-------------------------NOTE------------------------ Not required to be performed until 24 hours after a required pump is not in operation. --- Verify correct breaker alignment and indicated power are available to each required pump.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops - MODE 4

LCO 3.4.6 Two loops consisting of any combination of RCS loops and residual heat removal (RHR) loops shall be OPERABLE, and one loop shall be in operation.

------------------------------------NOTES------------------------------------

1. All reactor coolant pumps (RCPs) and RHR pumps may not be in operation for ≤ 1 hour per 8 hour period to perform tests provided:

   a. No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1; and

   b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. No RCP shall be started with any RCS cold leg temperature ≤ the Over Pressure Protection System (OPPS) enable temperature specified in the PTLR unless:

   a. The secondary side water temperature of each steam generator (SG) is ≤ 50°F above each of the RCS cold leg temperatures; or

   b. There is a steam or gas bubble in the pressurizer.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RCS loop inoperable. <u>AND</u> Two RHR loops inoperable.	A.1 Initiate action to restore a second loop to OPERABLE status.	Immediately
B. One RHR loop inoperable. <u>AND</u> Two RCS loops inoperable.	----------------NOTE---------------- Required Action B.1 is not applicable if all RCS and RHR loops are inoperable and Condition C is entered. -- B.1 Be in MODE 5.	 24 hours
C. All RCS and RHR loops inoperable. <u>OR</u> No RCS or RHR loop in operation.	C.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1. <u>AND</u> C.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.6.1	Verify one RHR or RCS loop is in operation.	12 hours
SR 3.4.6.2	Verify SG secondary side water level is ≥ 60% (Wide Range) for each required RCS loop.	12 hours
SR 3.4.6.3	-------------------------NOTE------------------------- Not required to be performed until 24 hours after a required pump is not in operation. ---	
	Verify correct breaker alignment and indicated power are available to each required pump.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops - MODE 5, Loops Filled

LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

a. One additional RHR loop shall be OPERABLE; or

b. The secondary side water level of at least one steam generator (SG) shall be ≥ 60% (Wide Range).

------------------------------------NOTES------------------------------------

1. The RHR pump of the loop in operation may not be in operation for ≤ 1 hour per 8 hour period to perform tests provided:

   a. No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1; and

   b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. One required RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

3. No reactor coolant pump shall be started with one or more RCS cold leg temperatures ≤ the Over Pressure Protection System (OPPS) enable temperature specified in the PTLR unless:

   a. The secondary side water temperature of each SG is ≤ 50°F above each of the RCS cold leg temperatures; or

   b. There is a steam or gas bubble in the pressurizer.

4. Both RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

APPLICABILITY: MODE 5 with RCS loops filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop inoperable. AND Both SGs secondary side water levels not within limits.	A.1 Initiate action to restore a second RHR loop to OPERABLE status. OR A.2 Initiate action to restore required SG secondary side water levels to within limits.	Immediately Immediately
B. Both RHR loops inoperable. OR No RHR loop in operation.	B.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1. AND B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify one RHR loop is in operation.	12 hours
SR 3.4.7.2	Verify SG secondary side water level is ≥ 60% (Wide Range) in the required SG.	12 hours
SR 3.4.7.3	-------------------------NOTE------------------------- Not required to be performed until 24 hours after a required pump is not in operation. --- Verify correct breaker alignment and indicated power are available to each required RHR pump.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops - MODE 5, Loops Not Filled

LCO 3.4.8 Two residual heat removal (RHR) loops shall be OPERABLE and one RHR loop shall be in operation.

-----------------------------------NOTES-----------------------------------

1. All RHR pumps may not be in operation for ≤ 15 minutes provided:

   a. No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1;

   b. The core outlet temperature is maintained > 10°F below saturation temperature; and

   c. No draining operations to further reduce the RCS water volume are permitted.

2. One RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

APPLICABILITY: MODE 5 with RCS loops not filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR loop inoperable.	A.1 Initiate action to restore RHR loop to OPERABLE status.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Both RHR loops inoperable. OR No RHR loop in operation.	B.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1. AND ----------------NOTE---------------- A Safety Injection pump may be run as required to maintain adequate core cooling and RCS inventory. --	Immediately
	B.2 Initiate action to restore one RHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify one RHR loop is in operation.	12 hours
SR 3.4.8.2	---------------------------NOTE--------------------------- Not required to be performed until 24 hours after a required pump is not in operation. -- Verify correct breaker alignment and indicated power are available to each required RHR pump.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 90%; and

b. Two groups of pressurizer heaters OPERABLE and capable of being powered from an emergency power supply.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit.	A.1 Be in MODE 3.	6 hours
	AND	
	A.2 Fully insert all rods.	6 hours
	AND	
	A.3 Place Rod Control system in a condition incapable of rod withdrawal.	6 hours
	AND	
	A.4 Be in MODE 4.	12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One group of pressurizer heaters inoperable.	B.1 Restore group of pressurizer heaters to OPERABLE status.	72 hours
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water level is ≤ 90%.	12 hours
SR 3.4.9.2	Verify required pressurizer heaters are capable of being powered from an emergency power supply.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Two pressurizer safety valves shall be OPERABLE with lift settings ≥ 2410 psig and ≤ 2560 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > the Over Pressure Protection System (OPPS) enable temperature specified in the PTLR.

-----------------------------------NOTE-----------------------------------
The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 36 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met. OR Both pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 with any RCS cold leg temperature ≤ the OPPS enable temperature specified in the PTLR.	6 hours 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within 2460 to 2510 psig.	In accordance with the Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

LCO 3.4.11 Each PORV and associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

--NOTES--

1. Separate Condition entry is allowed for each PORV and each block valve.

2. LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both PORVs inoperable due to excessive seat leakage.	A.1 Close and maintain power to associated block valve(s).	1 hour

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One PORV inoperable for reasons other than Condition A.	B.1 Close associated block valve.	1 hour
	AND	
	B.2 Remove power from associated block valve.	1 hour
	AND	
	B.3 Restore PORV to OPERABLE status.	72 hours
C. One block valve inoperable.	----------------NOTE---------------- Required Actions C.1 and C.2 do not apply when block valve is inoperable solely as a result of complying with Required Actions B.2 or E.2 --	
	C.1 Place associated PORV in manual control.	1 hour
	AND	
	C.2 Restore block valve to OPERABLE status.	72 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3.	6 hours
	AND	
	D.2 Be in MODE 4.	12 hours
E. Both PORVs inoperable for reasons other than Condition A.	E.1 Close associated block valves.	1 hour
	AND	
	E.2 Remove power from associated block valves.	1 hour
	AND	
	E.3 Be in MODE 3.	6 hours
	AND	
	E.4 Be in MODE 4.	12 hours
F. Both block valves inoperable.	-----------------NOTE----------------- Required Action F.1 does not apply when block valve is inoperable solely as a result of complying with Required Actions B.2 or E.2 --	
	F.1 Restore one block valve to OPERABLE status.	2 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Required Action and associated Completion Time of Condition F not met.	G.1 Be in MODE 3. <u>AND</u> G.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	-----------------------------NOTES----------------------------- 1. Not required to be performed with block valve closed in accordance with the Required Actions of this LCO. 2. Only required to be performed in MODES 1 and 2. --- Perform a complete cycle of each block valve.	92 days
SR 3.4.11.2	-----------------------------NOTE----------------------------- Only required to be performed in MODES 1 and 2. --- Perform a complete cycle of each PORV.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) > Safety Injection (SI) Pump Disable Temperature

LCO 3.4.12 LTOP shall be provided with:

a. a maximum of one SI pump capable of injecting into the RCS;

b. the emergency core cooling system (ECCS) accumulators isolated;

c. an OPERABLE Over Pressure Protection System (OPPS); and

d. two OPERABLE pressurizer power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR.

-------------------------------------NOTES-------------------------------------

1. Both SI pumps may be run for ≤ 1 hour while conducting SI system testing providing there is a steam or gas bubble in the pressurizer and at least one isolation valve between the SI pump and the RCS is shut.

2. ECCS accumulator may be unisolated when accumulator pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

APPLICABILITY: MODE 4 when any RCS cold leg temperature is ≤ the OPPS enable temperature specified in the PTLR and > the SI pump disable temperature specified in the PTLR.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Two SI pumps capable of injecting into the RCS.	A.1 Initiate action to assure a maximum of one SI pump is capable of injecting into the RCS.	Immediately
B. An ECCS accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	B.1 Isolate affected ECCS accumulator.	1 hour
C. Required Action and associated Completion Time of Condition B not met.	C.1 Increase RCS cold leg temperature to > the OPPS enable temperature specified in the PTLR.	12 hours
	OR	
	C.2 Depressurize affected ECCS accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	12 hours
D. One required PORV inoperable.	D.1 Restore required PORV to OPERABLE status.	7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two PORVs inoperable. <u>OR</u> Required Action and associated Completion Time of Condition A, C, or D not met. <u>OR</u> OPPS inoperable.	E.1 Be in MODE 5. <u>AND</u> E.2 Depressurize RCS and establish RCS vent of ≥ 3 square inches.	8 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify a maximum of one SI pump is capable of injecting into the RCS.	12 hours
SR 3.4.12.2	----------NOTE---------- Only required to be performed when ECCS accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. -------------------- Verify each ECCS accumulator is isolated.	Once within 12 hours and every 12 hours thereafter
SR 3.4.12.3	Verify PORV block valve is open for each required PORV.	72 hours
SR 3.4.12.4	----------NOTE---------- Not required to be performed until 12 hours after decreasing RCS cold leg temperature to ≤ the OPPS enable temperature specified in the PTLR. -------------------- Perform a COT on OPPS.	31 days
SR 3.4.12.5	Perform CHANNEL CALIBRATION for each OPPS actuation channel.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 Low Temperature Overpressure Protection (LTOP) ≤ Safety Injection Pump (SI) Pump Disable Temperature

LCO 3.4.13 LTOP shall be provided with: 1) no SI Pumps capable of injecting into the RCS; 2) the emergency core cooling system (ECCS) accumulators isolated; and 3) one of the following pressure relief capabilities:

a. An Over Pressure Protection System (OPPS) shall be OPERABLE with two pressurizer power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR; or

b. The RCS depressurized and an RCS vent of ≥ 3 square inches.

------------------------------------NOTES------------------------------------

1. Both safety injection (SI) pumps may be run for ≤ 1 hour while conducting SI system testing provided there is a steam or gas bubble in the pressurizer and at least one isolation valve between the SI pump and the RCS is shut.

2. During reduced inventory conditions an SI pump may be run as required to maintain adequate core cooling and RCS inventory.

3. ECCS accumulator may be unisolated when accumulator pressure is less than the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.

APPLICABILITY: MODE 4 when any RCS cold leg temperature is ≤ the SI Pump disable temperature specified in the PTLR,
MODE 5 when the steam generator (SG) primary system manway and pressurizer manway are closed and secured in position,
MODE 6 when the reactor vessel head is on and the SG primary system manway and pressurizer manways are closed and secured in position.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both SI pump(s) capable of injecting into the RCS.	A.1 Initiate action to assure no SI pump is capable of injecting into the RCS.	Immediately
B. An ECCS accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	B.1 Isolate affected ECCS accumulator.	1 hour
C. Required Action and associated Completion Time of Condition B not met.	C.1 Increase RCS cold leg temperature to > the OPPS enable temperature specified in the PTLR.	12 hours
	<u>OR</u>	
	C.2 Depressurize affected ECCS accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. ----------NOTE--------- Only applicable in LCO 3.4.13.a. --------------------------- One required PORV inoperable.	D.1	Restore required PORV to OPERABLE status.	24 hours
E. Two required PORVs inoperable for LCO 3.4.13.a. <u>OR</u> Required Action and associated Completion Time of Condition A, C, or D not met. <u>OR</u> OPPS inoperable.	E.1	Depressurize RCS and establish RCS vent of ≥ 3 square inches.	8 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	Verify no SI pumps are capable of injecting into the RCS.	12 hours
SR 3.4.13.2	---------------------------NOTE--------------------------- Only required to be performed when ECCS accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. --- Verify each ECCS accumulator is isolated.	Once within 12 hours and every 12 hours thereafter
SR 3.4.13.3	Verify required RCS vent ≥ 3 square inches open.	12 hours for unlocked open vent valve(s) <u>AND</u> 31 days for other vent path(s)
SR 3.4.13.4	Verify PORV block valve is open for each required PORV.	72 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.13.5	---------------------------NOTE------------------------- 1. Not required to be performed until 12 hours after decreasing RCS cold leg temperature to ≤ the OPPS enable temperature specified in the PTLR. 2. Only required to be performed when complying with LCO 3.4.13.a. -- Perform a COT on OPPS.	31 days
SR 3.4.13.6	Perform CHANNEL CALIBRATION for OPPS actuation channel.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 RCS Operational LEAKAGE

LCO 3.4.14 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS unidentified LEAKAGE not within limit.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2.1 Identify LEAKAGE. OR B.2.2 Be in MODE 5.	6 hours 54 hours 84 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. RCS identified LEAKAGE not within limit for reasons other than pressure boundary LEAKAGE.	C.1 Be in MODE 3. <u>AND</u> C.2.1 Reduce LEAKAGE to within limits. <u>OR</u> C.2.2 Be in MODE 5.	6 hours 14 hours 44 hours
D. Pressure boundary LEAKAGE exists. <u>OR</u> SG LEAKAGE not within limit.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.14.1	---NOTE--- Not required to be performed until 12 hours after establishment of steady state operation. --- Verify RCS operational leakage within limits by performance of RCS water inventory balance.	 72 hours
SR 3.4.14.2	Verify steam generator tube integrity is in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.15 Leakage from each RCS PIV shall be within limit.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

--NOTES--

1. Separate Condition entry is allowed for each flow path.

2. Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more flow paths with leakage from one or more RCS PIVs not within limit.	----------------NOTE---------------- Each valve used to satisfy Required Action A.1 must have been verified to meet SR 3.4.15.1 and be in the high pressure portion of the system. ------------------------------------	
	A.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve.	4 hours
	<u>AND</u>	
	A.2 Restore RCS PIV to within limits.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	-------------------------NOTES------------------------- Not required to be performed in MODES 3 and 4. -- Verify leakage from each RCS PIV is equivalent to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure ≥ 2215 psig and ≤ 2255 psig.	24 months AND Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months AND Prior to returning the valve to service after maintenance, repair, or replacement work is performed

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Leakage Detection Instrumentation

LCO 3.4.16 The following RCS leakage detection instrumentation shall be OPERABLE:

a. One containment sump A monitor (pump run time instrumentation); and

b. One containment radionuclide monitor.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----NOTE-----
LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required containment sump monitor inoperable.	A.1 -----NOTE----- Not required until 12 hours after establishment of steady state operation. ----------	
	Perform SR 3.4.14.1.	Once per 24 hours
	<u>AND</u>	

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Restore required containment sump monitor to OPERABLE status.	30 days
B. Required containment radionuclide monitor inoperable.	B.1.1 Analyze grab samples of the containment atmosphere. OR B.1.2 ------------NOTE------------ Not required until 12 hours after establishment of steady state operation. ------------------------------- Perform SR 3.4.14.1.	Once per 24 hours Once per 24 hours
	AND B.2 Restore required containment radionuclide monitor to OPERABLE status.	30 days
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. All required monitors inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Perform CHANNEL CHECK of the required containment radionuclide monitor.	12 hours
SR 3.4.16.2	Perform COT of the required containment radionuclide monitor.	92 days
SR 3.4.16.3	Perform CHANNEL CALIBRATION of the required containment sump monitor.	24 months
SR 3.4.16.4	Perform CHANNEL CALIBRATION of the required containment radionuclide monitor.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 RCS Specific Activity

LCO 3.4.17 The specific activity of the reactor coolant shall be within the following limits:

a. Dose Equivalent I-131 specific activity ≤ 1.0 μCi/gm; and

b. Gross specific activity ≤ $100/\bar{E}$ μCi/gm.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 specific activity not within limit .	------------------Note------------------ LCO 3.0.4 is not applicable. ---	
	A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.17-1.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Gross specific activity of the reactor coolant not within limit.	B.1 Be in MODE 3 with $T_{avg} < 500°F$.	6 hours
C. Required Action and associated Completion Time of Condition A not met. OR DOSE EQUIVALENT I-131 specific activity in the unacceptable region of Figure 3.4.17-1.	C.1 Be in MODE 3 with $T_{avg} < 500°F$.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify reactor coolant gross specific activity $\leq 100/\bar{E}\ \mu Ci/gm$.	7 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.17.2	---------------------------NOTE-------------------------- Only required to be performed in MODE 1. -- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity $\leq$ 1.0 μCi/gm.	14 days <u>AND</u> Between 2 and 6 hours after a THERMAL POWER change of $\geq$ 15% RTP within a 1 hour period
SR 3.4.17.3	---------------------------NOTE-------------------------- Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for $\geq$ 48 hours. -- Determine $\bar{E}$ from a reactor coolant sample.	184 days

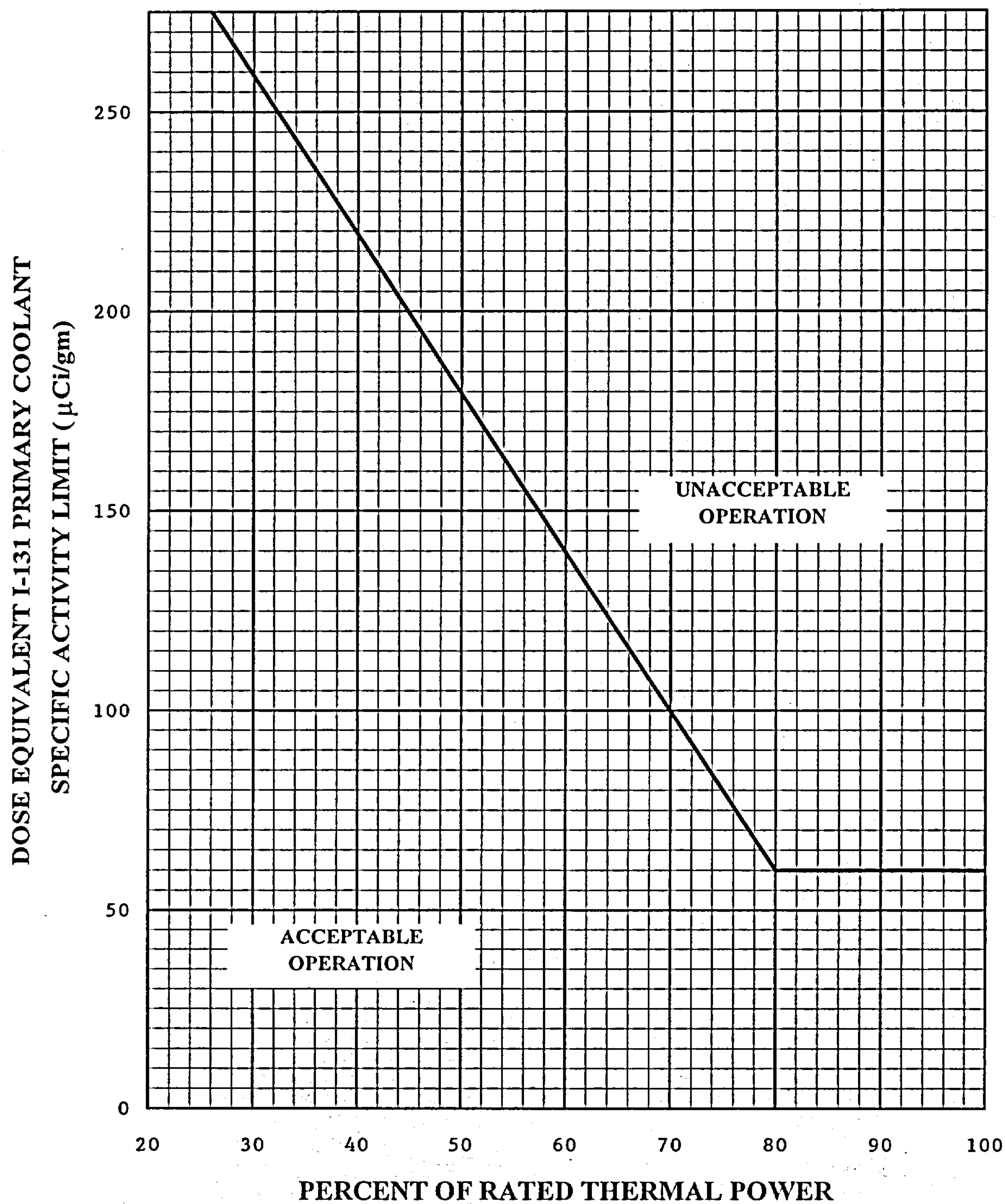

Figure 3.4.17-1 (page 1 of 1)
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity
Limit Versus Percent of RATED THERMAL POWER

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.18 RCS Loops - Test Exceptions

LCO 3.4.18 The requirements of LCO 3.4.4, "RCS Loops - MODES 1 and 2," may be suspended, with THERMAL POWER < P-7.

APPLICABILITY: MODES 1 and 2 during startup and PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER ≥ P-7.	A.1 Open reactor trip breakers.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.18.1	Verify THERMAL POWER is < P-7.	1 hour
SR 3.4.18.2	Perform a COT for each power range neutron flux - low and intermediate range neutron flux channel and P-7.	Prior to initiation of startup and PHYSICS TESTS

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow - Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

The RCS pressure limit is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits.

The RCS coolant average temperature limit is consistent with full power operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNB limits.

The RCS flow rate normally remains constant during an operational fuel cycle with both pumps running. The minimum RCS flow limit specified in the COLR corresponds to that assumed for DNB analyses. Flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNBR criteria. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit and RCS average temperature limit specified in the COLR are based on transient analyses assumptions, with allowance for steady state fluctuation, deadband and measurement errors. The measured RCS flow rate is decreased approximately 2.3% for conservatism, when being compared to the limit specified in the COLR.

The RCS DNB parameters satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO specifies limits on the monitored process variables- pressurizer pressure, RCS average temperature, and RCS total flow rate - to ensure the core operates within the limits assumed in the safety analyses. These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

The numerical values for pressure, temperature, and flow rate specified in the COLR are given for the measurement location and have been adjusted for instrument error.

BASES (continued)

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES, the power level is low enough that DNB is not a concern.

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp > 5% RTP per minute or a THERMAL POWER step > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Since increasing power transients are initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations. Decreasing power transients are in the direction which provides increased DNBR margin.

Another set of limits on DNB related parameters is provided in SL 2.1.1, "Reactor Core SLs." Those limits are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action.

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, as required by Required Action B.1, to restore DNB margin and eliminate the potential for violation of the accident analysis bounds.

BASES

ACTIONS

A.1 (continued)

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience.

B.1

If Required Action A.1 is not met within the associated 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 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.3

Measurement of RCS total flow rate once every 24 months allows the installed RCS flow instrumentation to be calibrated and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate. This verification may be performed via a precision calorimetric heat balance or other means.

The Frequency of 24 months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance.

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is required to be performed within 7 days after reaching 90% RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of 90% RTP to obtain accurate RCS flow data.

REFERENCES

1. USAR, Section 14.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

This LCO is based upon meeting several major considerations before the reactor can be made critical and while the reactor is critical.

The first consideration is isothermal temperature coefficient (ITC), LCO 3.1.3, "Isothermal Temperature Coefficient (ITC)." In the transient and accident analyses, the ITC is assumed to be in a range from slightly positive to negative and the operating temperature is assumed to be within the nominal operating envelope while the reactor is critical. The LCO on minimum temperature for criticality helps ensure the plant is operated consistent with these assumptions.

The second consideration is the protective instrumentation. Because certain protective instrumentation (e.g., excore neutron detectors) can be affected by moderator temperature, a temperature value within the nominal operating envelope is chosen to ensure proper indication and response while the reactor is critical.

The third consideration is the pressurizer operating characteristics. The transient and accident analyses assume that the pressurizer is within its normal startup and operating range (i.e., saturated conditions and steam bubble present). It is also assumed that the RCS temperature is within its normal expected range for startup and power operation. Since the density of the RCS water, and hence the response of the pressurizer to transients, depends upon the initial temperature of the moderator, a minimum value for moderator temperature within the nominal operating envelope is chosen for critically.

The fourth consideration is that the reactor vessel is above its minimum nil ductility reference temperature when the reactor is critical.

BASES (continued)

APPLICABLE SAFETY ANALYSES

Although the RCS minimum temperature for criticality is not itself an initial condition assumed in Design Basis Accidents (DBAs), the closely aligned temperature for hot zero power (HZP) is a process variable that is an initial condition of DBAs, such as the rod cluster control assembly (RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed at zero power that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

All low power safety analyses assume initial RCS loop temperatures are within the nominal operating envelope around the HZP temperature of 547°F (Ref. 1). The minimum temperature for criticality limitation provides a small band, 7°F, for critical operation below HZP. This band allows critical operation below HZP during plant startup and does not adversely affect any safety analyses since the ITC is not significantly affected by the small temperature difference between HZP and the minimum temperature for criticality.

The RCS minimum temperature for criticality satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Compliance with the LCO ensures that the reactor will not be made or maintained critical ($k_{eff} \geq 1.0$) at a temperature less than a small band below the HZP temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

BASES (continued)

APPLICABILITY

In MODE 1 and MODE 2 with $k_{eff} \geq 1.0$, LCO 3.4.2 is applicable since the reactor can only be critical ($k_{eff} \geq 1.0$) in these MODES.

The special test exception of LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 2," permits PHYSICS TESTS to be performed at ≤ 5% RTP with RCS loop average temperatures slightly lower than normally allowed so that fundamental nuclear characteristics of the core can be verified. In order for nuclear characteristics to be accurately measured, it may be necessary to operate outside the normal restrictions of this LCO. For example, to measure the ITC at beginning of cycle, it is necessary to allow RCS loop average temperatures to fall below $T_{no\ load}$, which may cause RCS loop average temperatures to fall below the temperature limit of this LCO.

ACTIONS

A.1

If the parameters that are outside the limit cannot be restored, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 2 with $k_{eff} < 1.0$ within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time is reasonable, based on operating experience, to reach MODE 2 with $k_{eff} < 1.0$ in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

RCS loop average temperature is required to be verified at or above 540°F every 12 hours. The SR to verify RCS loop average temperatures every 12 hours takes into account indications and alarms that are continuously available to the operator in the control

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1 (continued)

room and are consistent with other routine Surveillances which are typically performed once per shift. In addition, operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.

REFERENCES

1. USAR, Section 14.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The PTLR contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature, based on Reference 1.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G, requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials and requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G.

BASES

BACKGROUND (continued)

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NDT} of the vessel material has been established by periodically removing and evaluating irradiated reactor vessel material specimens. The operating P/T limit curves have been adjusted based on the evaluation findings and the recommendations of the program prescribed in Reference 2.

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit curve includes the Reference 2 requirement that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must

BASES

BACKGROUND (continued)	be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E, provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.
APPLICABLE SAFETY ANALYSES	The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition. RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The two elements of this LCO are: a. The limit curves for heatup, cooldown, and ISLH testing; and b. Limits on the rate of change of temperature. The LCO limits apply to all components of the RCS, except the pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure. The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

BASES

LCO (continued)

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follow:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G. Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow - Departure from Nucleate Boiling (DNB) Limits"; LCO 3.4.2, "RCS Minimum Temperature for Criticality"; and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

BASES (continued)

ACTIONS

A.1 and A.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E, may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note requiring that Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

BASES

ACTIONS (continued)

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. In reduced pressure conditions, which requires reduced temperature, the possibility of propagation of undetected flaws is decreased.

If the required restoration activity cannot be accomplished within 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature.

If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Action B.1 and Required Action B.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions.

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 with RCS pressure < 500 psig within 36 hours.

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

C.1 and C.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than when in MODE 1, 2, 3, or 4, so that

BASES

ACTIONS

C.1 and C.2 (continued)

the RCPB is returned to a condition that has been verified by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, or inspection of the components.

ASME Code, Section XI, Appendix E, may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

Condition C is modified by a Note requiring that Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to

BASES

SURVEILLANCE REQUIREMENTS	SR 3.4.3.1 (continued) monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time. Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied. This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.
REFERENCES	1. WCAP-14040-NP-A, January 1996. 2. USAR, Section 4.7.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops - MODES 1 and 2

BASES

BACKGROUND

The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the RCS include:

a. Moderating the neutron energy level to the thermal state, to increase the probability of fission;

b. Improving the neutron economy by acting as a reflector;

c. Carrying the soluble neutron poison, boric acid; and

d. Providing a second barrier against fission product release to the environment.

The reactor coolant is circulated through two loops connected in parallel to the reactor vessel, each containing a SG, a reactor coolant pump (RCP), and appropriate flow and temperature instrumentation for both control and protection. The reactor vessel contains the clad fuel. The SGs provide the heat sink to the isolated secondary coolant. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage. This forced circulation of the reactor coolant ensures mixing of the coolant for proper boration and chemistry control.

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the design bases accident initial conditions including RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant

BASES

APPLICABLE SAFETY ANALYSES (continued)

forced flow rate, which is represented by the number of RCS loops in service.

Both transient and steady state analyses include the effect of flow on the departure from nucleate boiling ratio (DNBR). The transient and accident analyses for the plant have been performed assuming both RCS loops are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are most important to RCP operation are the two pump coastdown, single pump locked rotor, misaligned rod, and rod withdrawal events (Ref. 1).

The plant is designed to operate with both RCS loops in operation to maintain DNBR during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant.

RCS Loops - MODES 1 and 2 satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, two pumps are required at power.

An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program.

APPLICABILITY

In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the

BASES

APPLICABILITY (continued)

assumptions of the accident analyses remain valid, both RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops-MODE 3";
LCO 3.4.6, "RCS Loops-MODE 4";
LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level" (MODE 6).

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification every 12 hours that each RCS loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal while maintaining the margin to DNB. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.

REFERENCES

1. USAR, Section 14.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops - MODE 3

BASES

BACKGROUND

In MODE 3, the primary function of the RCS is removal of decay heat and transfer of this heat, via the steam generator (SG), to the secondary plant. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through two RCS loops, connected in parallel to the reactor vessel, each containing a SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The reactor vessel contains the clad fuel. The SGs provide the heat sink. The RCPs circulate the water through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage.

In MODE 3, RCPs are normally used to provide forced circulation for heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP running is sufficient to remove core decay heat in response to transients or operational events. However, two RCS loops are required to be OPERABLE to ensure redundant capability for decay heat removal.

The MODE 3 decay heat removal requirements are low enough that natural circulation is sufficient to remove core decay heat when the potential for operational events is minimized (Ref. 1).

APPLICABLE SAFETY ANALYSES

Whenever the reactor trip breakers (RTBs) are in the closed position and the control rod drive mechanisms (CRDMs) are energized, an inadvertent rod withdrawal from subcritical, resulting

BASES

APPLICABLE ANALYSES SAFETY (continued)	in a power excursion, is possible. Such a transient could be caused by a malfunction of the rod control system. Therefore, in MODE 3 with the Rod Control System capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires two RCS loops to be OPERABLE and in operation to ensure that the accident analyses input assumptions are met. Failure to provide decay heat removal by forced circulation, when control rods may be withdrawn, may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. RCS Loops - MODE 3 satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The purpose of this LCO is to require that both RCS loops be OPERABLE. In MODE 3 with the Rod Control System capable of rod withdrawal, both RCS loops must be in operation. Two RCS loops are required to be in operation in MODE 3 with the Rod Control System capable of rod withdrawal due to the postulation of a power excursion because of an inadvertent control rod withdrawal. The required number of RCS loops in operation ensures that the transient analysis acceptance criteria will be met. When the Rod Control System is not capable of rod withdrawal, only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. An additional RCS loop is required to be OPERABLE to ensure redundant capability for decay heat removal.

BASES

LCO (continued)

The Note permits both RCPs to not be in operation for ≤ 12 hours to perform preplanned work activities.

One purpose of the Note is to allowance performance of tests that are designed to validate various accident analyses values. One of these tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test was performed during the initial startup testing program, and would normally only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve must be revalidated by conducting the test again. Another test performed during the startup testing program was the validation of rod drop times, both with and without flow. Any future no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits stopping the pumps in order to perform this test and validate the assumed analysis values.

Another purpose of the Note is to allow stopping of both RCP's for a sufficient time to perform station electrical lineup changes without transition to MODE 4. Transition to MODE 4 would put the plant through unnecessary cooldown and heatup transients. The 12 hour time period specified is adequate to perform the necessary load shedding, switching and load restoration activities and restart an RCP without requiring transition to MODE 4.

Utilization of the Note is permitted provided the following conditions are met:

BASES

LCO (continued)

a. No operations are permitted that would dilute the RCS boron concentration with coolant with boron concentration less than required to meet SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited to preclude the need for a boration, due to the time required to achieve a uniform distribution when in natural circulation (Ref. 1); and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

An OPERABLE RCS loop consists of one OPERABLE RCP and one OPERABLE SG which has the minimum water level specified in SR 3.4.5.2. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with the Rod Control System capable of rod withdrawal. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to Mode 3 with the Rod Control System not capable of rod withdrawal.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops-MODES 1 and 2";
LCO 3.4.6, "RCS Loops-MODE 4";
LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled";

BASES

APPLICABILITY (continued)

LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level"(MODE 6).

ACTIONS

A.1

If one RCS loop is inoperable, redundancy for heat removal is lost. The Required Action is restoration of the RCS loop to OPERABLE status within the Completion Time of 72 hours. This time allowance is a justified period to be without the redundant, nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core and because of the low probability of a failure in the remaining loop occurring during this period.

B.1

If restoration is not possible within 72 hours, the unit must be brought to MODE 4. In MODE 4, the unit may be placed on the Residual Heat Removal System. The additional Completion Time of 12 hours is compatible with required operations to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

If one RCS loop is not in operation, and the Rod Control System is capable of rod withdrawal, the Required Action is either to restore the RCS loop to operation or place the Rod Control System in a condition incapable of rod withdrawal (e.g., to de-energize all CRDMs by opening the RTBs or de-energizing the motor generator

BASES

ACTIONS

C.1 and C.2 (continued)

(MG) sets). When the Rod Control System is capable of rod withdrawal, it is postulated that a power excursion could occur in the event of an inadvertent control rod withdrawal. This mandates having the heat transfer capacity of two RCS loops in operation. If only one loop is in operation, the Rod Control System must be rendered incapable of rod withdrawal. The Completion Times of 1 hour to restore the required RCS loop to operation or defeat the Rod Control System is adequate to perform these operations in an orderly manner without exposing the unit to risk for an undue time period.

D.1, D.2, and D.3

If both RCS loops are inoperable or no RCS loop is in operation, except during conditions permitted by the Note in the LCO section, the Rod Control System must be placed in a condition incapable of rod withdrawal (e.g., all CRDM's de-energized by opening the RTBs or de-energizing the MG sets). All operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended, and action to restore one of the RCS loops to OPERABLE status and operation must be initiated. Suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core; however, coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification every 12 hours that the required loops are in operation. Verification may include flow rate, temperature, or pump status monitoring, which helps ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.

SR 3.4.5.2

SR 3.4.5.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side water level is ≥ 60% wide range or equivalent narrow range level for both RCS loops. If the SG secondary side is < 60% wide range equivalent water level, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

SR 3.4.5.3

Verification that each required RCP is OPERABLE ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to each required RCP. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a pump is not in operation.

BASES (continued)

REFERENCES

1. License Amendment Request Dated November 19, 1999. (Approved by License Amendment 152/143, July 14, 2000.)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to either the steam generator (SG) secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through two RCS loops connected in parallel to the reactor vessel, each containing a SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The RCPs or RHR pumps circulate the coolant through the reactor vessel and SGs or the RHR heat exchangers at a sufficient rate to ensure proper heat transfer and boric acid mixing.

In MODE 4, either RCPs or RHR pumps can be used to provide forced circulation. The intent of this LCO is to provide forced flow from at least one RCS loop or one RHR loop for decay heat removal and transport. The flow provided by one RCS loop or RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that two paths be available to provide redundancy for decay heat removal.

APPLICABLE SAFETY ANALYSES

In MODE 4, RCS circulation increases the time available for mitigation of an accidental boron dilution event. The RCS and RHR loops provide this circulation.

RCS Loops - MODE 4 satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

The purpose of this LCO is to require that at least two loops be OPERABLE in MODE 4 and that one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS loops and RHR loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits all RCPs or RHR pumps to not be in operation for ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are designed to validate various accident analyses values. One of the LCO tests performed during the startup testing program was validation of rod drop times during cold conditions, both with and without flow. If changes are made to the RCS that would cause a change in flow characteristics of the RCS, the input values must be revalidated by conducting the test again. Any future no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits stopping the pumps in order to perform this test and validate the assumed analysis values. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met along with any other conditions imposed by startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant with boron concentration less than required to meet SDM of LCO 3.1.1, therefore maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM in maintained is prohibited to preclude the need for a boration, due to the time required to achieve a uniform distribution when in natural circulation (Ref. 1); and

BASES

LCO (continued)

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 requires a steam or gas bubble in the pressurizer or that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature ≤ the OPPS enable temperature specified in the PTLR. A steam or gas bubble ensures that the pressurizer will accommodate the swell resulting from an RCP start. Either of these restraints prevents a low temperature overpressure event due to a thermal transient when an RCP is started.

An OPERABLE RCS loop consists of an OPERABLE RCP and an OPERABLE SG which has the minimum water level specified in SR 3.4.6.2.

Similarly for the RHR System, an OPERABLE RHR loop consists of an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

BASES (continued)

ACTIONS

A.1

If one RCS loop is inoperable and two RHR loops are inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second RCS or RHR loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal. Entry to a reduced MODE (MODE 5 or 6) requires RHR availability for long term decay heat removal. Remaining in MODE 4, with RCS loop operation, is conservative.

B.1

If one RHR loop is OPERABLE and in operation and there are no RCS loops OPERABLE, an inoperable RCS or RHR loop must be restored to OPERABLE status to provide a redundant means for decay heat removal.

If the parameters that are outside the limits cannot be restored, the unit must be brought to MODE 5 within 24 hours. Bringing the unit to MODE 5 is a conservative action with regard to decay heat removal. With only one RHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining RHR loop, it would be safer to initiate that loss from MODE 5 ($\leq$ 200°F) rather than MODE 4 (200 to 350°F). The Completion Time of 24 hours is a reasonable time, based on operating experience, to reach MODE 5 from MODE 4 in an orderly manner and without challenging plant systems.

The Note directs operations to Condition C if all RCS loops and RHR loops are inoperable, to ensure recognition of the correct Required Action.

BASES

ACTIONS (continued)

C.1 and C.2

If no loop is OPERABLE or in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RCS or RHR loop to OPERABLE status and operation must be initiated. The margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core; however, coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR requires verification every 12 hours that one RCS or RHR loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which helps ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS and RHR loop performance.

SR 3.4.6.2

SR 3.4.6.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side water

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.6.2 (continued)

level is ≥ 60% wide range or equivalent narrow range level. If the SG secondary side is < 60% wide range equivalent water level, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

SR 3.4.6.3

Verification that each required pump is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a pump is not in operation.

REFERENCES

1. License Amendment Request Dated November 19, 1999. (Approved by License Amendment 152/143, July 14, 2000.)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat either to the steam generator (SG) secondary side coolant via natural circulation (Ref. 1) or the component cooling water via the residual heat removal (RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs via natural circulation are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The RCS must be intact to support natural circulation. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with RCS loops filled, the reactor coolant is circulated by means of two RHR loops connected to the RCS, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for control and indication. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification.

The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR loop for decay heat removal and transport. The flow provided by one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an RHR loop that must be OPERABLE and in operation. The second path can be another

BASES

BACKGROUND (continued)	OPERABLE RHR loop or maintaining a SG with secondary side water level above 60% wide range to provide an alternate method for decay heat removal via natural circulation.
APPLICABLE SAFETY ANALYSES	In MODE 5, RCS circulation increases the time available for mitigation of an accidental boron dilution event. The RHR loops provide this circulation. RCS Loops - MODE 5 (Loops Filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).
LCO	The purpose of this LCO is to require that at least one RHR loop be OPERABLE and in operation with an additional RHR loop OPERABLE or a SG with secondary side water level ≥ 60% wide range. One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to provide redundancy. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is a SG. Should the operating RHR loop fail, the SG could be used to remove decay heat via natural circulation. Note 1 permits all RHR pumps to not be in operation ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests designed to validate various accident analyses values. One of the tests performed during the startup testing program was validation of rod drop times during cold conditions, both with and without flow. If changes are made to the RCS that would cause a change in flow characteristics of the RCS, the input values must be revalidated by conducting the test again. Any future no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits stopping the pumps in order to perform this test and validate the assumed analysis values. The 1 hour time period is adequate to perform the test, and operating

BASES

LCO (continued)

experience has shown that boron stratification is not likely during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration with coolant with boron concentration less than required to meet SDM of LCO 3.1.1, therefore maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure SDM is maintained is prohibited to preclude the need for a boration, due to the time required to achieve a uniform distribution when in natural circulation (Ref 2); and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 3 requires a steam or gas bubble in the pressurizer or that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature ≤ the OPPS enable temperature specified in the PTLR. A steam or gas bubble ensures that the pressurizer will accommodate the swell resulting from an RCP start. Either of these restraints prevents a low temperature overpressure event due to a thermal transient when an RCP is started.

BASES

LCO (continued)

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. An OPERABLE SG can perform as a heat sink via natural circulation when it has the minimum water level specified in SR 3.4.7.2.

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of a SG is required to be ≥ 60% wide range.

Operation in other MODES is covered by:

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

ACTIONS

A.1 and A.2

If one RHR loop is inoperable and the SGs have secondary side water levels < 60% wide range, redundancy for heat removal is lost.

BASES

ACTIONS

A.1 and A.2 (continued)

Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore a SG secondary side water level. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no RHR loop is in operation, except during conditions permitted by Note 1, or if no loop is OPERABLE, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. Suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core; however, coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that the required loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which helps ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.7.2

Verifying that at least one SG is OPERABLE by ensuring its secondary side water level is ≥ 60% wide range or equivalent narrow range level ensures an alternate decay heat removal method via natural circulation in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

SR 3.4.7.3

Verification that each required RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required RHR pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. If secondary side water level is ≥ 60% wide range in at least one SG, this Surveillance is not needed. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a pump is not in operation.

REFERENCES

1. NRC Information Notice 95-35, "Degraded Ability of Steam Generators to Remove Decay Heat by Natural Circulation".

2. License Amendment Request Dated November 19, 1999. (Approved by License Amendment 152/143, July 14, 2000.)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops - MODE 5, Loops Not Filled

BASES

BACKGROUND

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

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

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation increases the time available for mitigation of an accidental boron dilution event. The RHR loops provide this circulation.

RCS loops - MODE 5 (Loops Not Filled) satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

The purpose of this LCO is to require that at least two RHR loops be OPERABLE and one of these loops be in operation to transfer heat from the reactor coolant at a controlled rate. Heat cannot be removed via the RHR System unless forced flow is used. A minimum of one operating RHR pump meets the LCO requirement for one loop in operation. An additional RHR loop is required to be OPERABLE to provide redundancy.

BASES

LCO (continued)

Note 1 permits all RHR pumps to not be in operation for ≤ 15 minutes. The circumstances for stopping both RHR pumps are to be limited to situations when the outage time is short and core outlet temperature is maintained > 10°F below saturation temperature. The Note prohibits boron dilution with coolant at boron concentrations less than required to assure SDM is maintained or draining operations when RHR forced flow is stopped.

Note 2 allows one RHR loop to be inoperable for a period of ≤ 2 hours, provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when these tests are safe and possible.

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

APPLICABILITY

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

Operation in other MODES is covered by:

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

BASES (continued)

ACTIONS

A.1

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

B.1 and B.2

If no RHR loops are OPERABLE or in operation, except during conditions permitted by Note 1, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action must be initiated immediately to restore an RHR loop to OPERABLE status and operation. The margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core; however, coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must continue until one loop is restored to OPERABLE status and operation.

The Note in Required Action B.2 allows the use of one safety injection pump to provide heat removal in the event of a loss of RHR system cooling during reduced RCS inventory conditions.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This SR requires verification every 12 hours that one loop is in operation. Verification may include flow rate, temperature, or pump status monitoring, which helps ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.4.8.2

Verification that each required pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a pump is not in operation.

REFERENCES

None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation, and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters, and their emergency power supplies. Pressurizer safety valves and pressurizer power operated relief valves are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," respectively.

The intent of the LCO is to ensure that a steam bubble exists in the pressurizer prior to power operation to minimize the consequences of potential overpressure transients. The presence of a steam bubble is consistent with analytical assumptions. Relatively small amounts of noncondensible gases are typically present in the RCS and can inhibit the condensation heat transfer between the pressurizer spray and the steam, and diminish the spray effectiveness for pressure control. These small amounts of noncondensible gases can be ignored if the steam bubble is present.

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove core decay heat by either forced or natural circulation of reactor coolant. Inability to control the system pressure and maintain subcooling under conditions of natural

BASES

BACKGROUND (continued)	circulation flow in the primary system could lead to a loss of single phase natural circulation and decreased capability to remove core decay heat. One group of pressurizer heaters is adequate to maintain natural circulation conditions during a loss of offsite power. Two groups are required to be available to ensure redundant capability.
APPLICABLE SAFETY ANALYSES	In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. Safety analyses performed for lower MODES are not limiting with respect to pressurizer parameters. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption the analyses neglect the small fraction of noncondensible gases normally present. Safety analyses presented in the USAR (Ref. 1) do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure. The maximum pressurizer water level limit, which ensures that a steam bubble exists in the pressurizer, satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii). Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737, is the reason for providing an LCO.
LCO	The LCO requirement for the pressurizer to be OPERABLE with ≤ 90% level ensures that a steam bubble exists. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The level limit is specified to agree with the high pressurizer level trip allowable value. The LCO has been established to ensure the capability to establish and maintain

BASES

LCO (continued)

pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

The LCO requires two groups of OPERABLE pressurizer heaters, capable of being powered from an emergency power supply. These are Groups A and B. One group of pressurizer heaters with a capacity ≥ 100 kW is adequate to maintain natural circulation conditions during a loss of offsite power (Ref. 2). Two groups are required to be OPERABLE to ensure redundant capability.

APPLICABILITY

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

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

ACTIONS

A.1, A.2, A.3, and A.4

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal

BASES

ACTIONS

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

upper limit, even with the plant at steady state conditions. Normally the plant will trip in this event since the upper limit of this LCO is the same as the allowable value for Pressurizer High Water Level-Reactor Trip.

If the pressurizer water level is not within the limit, action must be taken to bring the unit to a MODE in which the LCO does not apply. To achieve this status, within 6 hours the unit must be brought to MODE 3, with all rods fully inserted and incapable of withdrawal. Additionally the unit must be brought to MODE 4 within 12 hours. This takes the unit out of the applicable MODES.

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.

B.1

If one group of pressurizer heaters is inoperable, restoration is required within 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand caused by loss of offsite power would be unlikely in this period. Pressure control may be maintained during this time using normal station powered heaters.

C.1 and C.2

If one group of pressurizer heaters is inoperable and cannot be restored in the allowed Completion Time of Required Action B.1, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3

BASES

ACTIONS

C.1 and C.2 (continued)

within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The Frequency of 12 hours has been shown by operating practice to be sufficient to regularly assess level for any deviation. Alarms are also available for early detection of abnormal level indications.

SR 3.4.9.2

This SR is not applicable for the Group A heaters since this group is permanently powered by a Class 1E power supply.

This Surveillance demonstrates that the Group B heaters can be manually transferred from the normal to the emergency power supply and energized. The Frequency of 24 months is based on a typical fuel cycle and is consistent with similar verifications of emergency power supplies.

REFERENCES

1. USAR, Section 14.
2. USAR, Section 4.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop type, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The required relief capacity for each valve, 325,000 lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODE 4, with one or more RCS cold leg temperatures ≤ the OPPS enable temperature specified in the PTLR, and MODE 5 and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) > Safety Injection (SI) Pump Disable Temperature", and LCO 3.4.13, "Low Temperature Overpressure Protection (LTOP) ≤ Safety Injection (SI) Pump Disable Temperature."

BASES

BACKGROUND (continued)	The as left upper and lower pressure limits are based on the ± 1% tolerance requirement (Ref. 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established. The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the American Society of Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.
APPLICABLE SAFETY ANALYSES	All accident and safety analyses in the USAR (Ref. 2) that require safety valve actuation assume operation of both pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analysis (Ref. 3) is also based on operation of both safety valves. Transients that could result in overpressurization if not properly terminated include: a. Uncontrolled rod withdrawal from power; b. Loss of reactor coolant flow; c. Loss of external electrical load; d. Loss of normal feedwater; e. Loss of all AC power to station auxiliaries; and f. Locked rotor.

BASES

APPLICABLE SAFETY ANALYSES (continued)	Detailed analyses of the above transients are contained in Reference 2. Compliance with this LCO is consistent with the design bases and accident analyses assumptions. Pressurizer safety valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	The two pressurizer safety valves are set to open at the RCS design pressure (2485 psig), within a ± 3% tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits following testing are based on the ± 1 % tolerance requirements (Ref. 1) for lifting pressures above 1000 psig. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analyses being required prior to resumption of reactor operation.
APPLICABILITY	In MODES 1, 2, and 3, and portions of MODE 4 above the OPPS enable temperature, OPERABILITY of both valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included, although the listed accidents may not require the safety valves for protection. The LCO is not applicable in MODE 4 when any RCS cold leg temperatures are ≤ the OPPS enable temperature specified in the

BASES

APPLICABILITY (continued)

PTLR or in MODE 5 because LTOP is provided. Overpressure protection is not required in MODE 6 with reactor vessel head detensioned.

The note allows entry into MODES 3 and 4 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 36 hour exception is based on 18 hour outage time for each of the two valves. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining RCS overpressure protection. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if both pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperatures ≤ the OPPS enable temperature specified in the PTLR within 24 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

BASES

ACTIONS

B.1 and B.2 (continued)

and without challenging plant systems. With any RCS cold leg temperatures at or below the OPPS enable temperature specified in the PTLR, overpressure protection is provided by the LTOP function. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by both pressurizer safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code, which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is ± 3% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III, with the 1968 Winter Addendum.
2. USAR, Section 14.
3. WCAP-7769, Rev. 1, June 1972.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power Operated Relief Valves (PORVs)

BASES

BACKGROUND

The pressurizer is equipped with two types of devices for pressure relief: pressurizer safety valves and PORVs. The PORVs are air operated valves that are controlled to open at a specific set pressure when the pressurizer pressure increases and close when the pressurizer pressure decreases. The PORVs may also be manually operated from the control room.

Block valves, which are normally open, are located between the pressurizer and the PORVs. The block valves are used to isolate the PORVs in case of excessive leakage or a stuck open PORV. Block valve closure is accomplished manually using controls in the control room. A stuck open PORV is, in effect, a small break loss of coolant accident (LOCA). As such, block valve closure terminates the RCS depressurization and coolant inventory loss.

The PORVs and their associated block valves may be used by plant operators to depressurize the RCS to recover from certain transients if normal pressurizer spray is not available. Additionally, the series arrangement of the PORVs and their block valves permits performance of surveillances on the valves during power operation.

The PORVs may also be used for feed and bleed core cooling in the case of multiple equipment failure events that are not within the design basis, such as a total loss of feedwater.

The PORVs, their block valves, and their controls are powered from the vital buses that normally receive power from offsite power sources, but are also capable of being powered from emergency power sources in the event of a loss of offsite power. The PORVs and their associated block valves are powered from two separate safety trains.

BASES

BACKGROUND (continued)	The two PORVs each have a relief capacity of 179,000 lb/hr at 2335 psig. The functional design of the PORVs is based on maintaining pressure below the Pressurizer High Pressure Reactor Trip setpoint following a step reduction of 47.5% of full load with steam dump. In addition, the PORVs minimize challenges to the pressurizer safety valves and also may be used for low temperature overpressure protection (LTOP). See LCO 3.4.12 and LCO 3.4.13 for LTOP requirements.

APPLICABLE SAFETY ANALYSES

Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal pressurizer spray is not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator.

The PORVs are also modeled in safety analyses for events that result in increasing RCS pressure for which departure from nucleate boiling ratio (DNBR) criteria are critical (Ref. 1). By assuming PORV actuation, the primary pressure remains below the high pressurizer pressure trip setpoint; thus, the DNBR calculation is more conservative. As such, this actuation is not required to mitigate these events, and PORV automatic operation is, therefore, not an assumed safety function.

Pressurizer PORVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation to mitigate the effects associated with an SGTR.

By maintaining two PORVs and their associated block valves OPERABLE, the single failure criterion is satisfied. An OPERABLE block valve may be either open, or closed and energized with the capability to be opened, since the required safety function is accomplished by manual operation. Although typically open to allow PORV operation, the block valves may be OPERABLE when closed to isolate the flow path of an inoperable PORV that is capable of being manually cycled (i.e., as in the case of excessive PORV leakage). Similarly, isolation of an OPERABLE PORV does not render that PORV or block valve inoperable provided the relief function remains available with manual action.

An OPERABLE PORV is required to be capable of manually opening and closing and not experiencing excessive seat leakage. Excessive seat leakage, although not associated with a specific acceptance criteria, exists when conditions dictate closure of the block valve to limit leakage.

Satisfying the LCO helps minimize challenges to fission product barriers.

APPLICABILITY

In MODES 1, 2, and 3, the PORVs are required to be OPERABLE for manual actuation to mitigate a SGTR. The PORV and its block valve are also required to be OPERABLE in MODES 1, 2, and 3 to maintain the integrity of the reactor coolant pressure boundary. This requires the ability to manually control the block valve to isolate a PORV with excessive seat leakage or a stuck-open PORV.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3 when RCS pressure is high and there is potential for a SGTR. The LCO is not applicable in

BASES

APPLICABILITY (continued)

MODES 4 and 5, and MODE 6 with the reactor vessel head in place, when both pressure and core energy are decreased and a SGTR can not occur. LCO 3.4.12 AND LCO 3.4.13 address the PORV requirements in these MODES.

ACTIONS

Note 1 has been added to clarify that each pressurizer PORV and each block valve are treated as separate entities, each with separate Completion Times (i.e., the Completion Time is on a component basis) for Conditions A, B and C. The exception for LCO 3.0.4, Note 2, permits entry into MODES 1, 2, and 3 to perform cycling of the PORVs or block valves to verify their OPERABLE status, in the event that testing was not satisfactorily performed in lower MODES.

A.1

PORVs may be inoperable due to excessive seat leakage. In this condition, either the PORVs must be restored or the flow path isolated within 1 hour. The associated block valve is required to be closed, but power must be maintained to the associated block valve, since removal of power would render the block valve inoperable. This permits operation of the plant until the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the problem condition.

Quick access to the PORV for pressure control can be made when power remains on the closed block valve. The Completion Time of 1 hour is based on plant operating experience that has shown that minor problems can be corrected or closure accomplished in this time period.

B.1, B.2 and B.3

If one PORV is inoperable for reasons other than condition A, it must be either restored, or isolated by closing the associated block

BASES

ACTIONS

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

valve and removing the power to the associated block valve. The Completion Times of 1 hour are reasonable, based on the small potential for challenges to the PORVs during this time period, and provide the operator adequate time to correct the situation. If the inoperable valve cannot be restored to OPERABLE status, it must be isolated within the specified time. Because there is at least one PORV that remains OPERABLE, an additional 72 hours is provided to restore the inoperable PORV to OPERABLE status. If the PORV cannot be restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

C.1 and C.2

If one block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in manual control. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation. Because at least one PORV remains OPERABLE, the operator is permitted a Completion Time of 72 hours to restore the inoperable block valve to OPERABLE status. The time allowed to restore the block valve is based upon the Completion Time for restoring an inoperable PORV in Condition B, since the PORVs may not be capable of mitigating an event if the inoperable block valve is not full open. If the block valve is restored within the Completion Time of 72 hours, the PORV may be restored to automatic operation. If it cannot be

BASES

ACTIONS

C.1 and C.2 (continued)

restored within this additional time, the plant must be brought to a MODE in which the LCO does not apply, as required by Condition D.

The Required Actions C.1 and C.2 are modified by a Note stating that the Required Actions do not apply if the sole reason for the block valve being declared inoperable is as a result of power being removed to comply with other Required Actions. In this event, the Required Actions for inoperable PORV(s) (which require the block valve power to be removed once it is closed) are adequate to address the condition. While it may be desirable to also place the PORV(s) in manual control, this may not be possible for all causes of Condition B or E entry with PORV(s) inoperable and not capable of being manually cycled (e.g., as a result of failed control power fuse(s) or control switch malfunction(s)).

D.1 and D.2

If the Required Action of Condition A, B, or C is not met, then the plant must be brought to a MODE in which the LCO does not apply.

To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12 and LCO 3.4.13.

BASES

ACTIONS (continued)

E.1, E.2, E.3 and E.4

If both PORVs are inoperable for reasons other than Condition A, it is necessary to either restore at least one valve within the Completion Time of 1 hour or isolate the flow path by closing and removing the power to the associated block valves. The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time and provides the operator time to correct the situation. If no PORVs are restored within the Completion Time, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12 and LCO 3.4.13.

F.1

If both block valves are inoperable, it is necessary to restore at least one block valve within 2 hours. The Completion Time is reasonable, based on the small potential for challenges to the system during this time and provides the operator time to correct the situation.

The Required Action F.1 is modified by a Note stating that the Required Action does not apply if the sole reason for the block valve being declared inoperable is as a result of power being removed to comply with other Required Actions. In this event, the Required Actions for inoperable PORV(s) (which require the block valve power to be removed once it is closed) are adequate to address the condition. While it may be desirable to also place the PORV(s) in manual control, this may not be possible for all causes of Condition B or E entry with PORV(s) inoperable and not capable of being manually cycled (e.g., as a result of failed control power fuse(s) or control switch malfunction(s)).

BASES

ACTIONS (continued)

G.1 and G.2

If the Required Actions of Condition F are not met, then the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODES 4 and 5, automatic PORV OPERABILITY may be required. See LCO 3.4.12 and LCO 3.4.13.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that the valve(s) can be opened and closed if needed. The basis for the Frequency of 92 days is the ASME Code, Section XI.

This SR is modified by two Notes. Note 1 modifies this SR by stating that it is not required to be performed with the block valve closed in accordance with the Required Action of Condition B or E. Opening the block valve in this condition increases the risk of an unisolable leak from the RCS since the PORV is already inoperable.

Note 2 modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions, prior to entering MODE 1 or 2.

SR 3.4.11.2

SR 3.4.11.2 requires a complete cycle of each PORV. Operating a PORV through one complete cycle ensures that the PORV can be

BASES

SURVEILLANCE REQUIREMENTS	<u>SR 3.4.11.2</u> (continued) manually actuated for mitigation of an SGTR. The Frequency of 24 months is based on a typical refueling cycle and industry accepted practice. The Note modifies this SR to allow entry into and operation in MODE 3 prior to performing the SR. This allows the test to be performed in MODE 3 under operating temperature and pressure conditions prior to entering MODE 1 or 2.
REFERENCES	1. USAR, Section 14.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) > Safety Injection (SI) Pump Disable Temperature

BASES

BACKGROUND

The LTOP function limits RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. The Over Pressure Protection System (OPPS) and the pressurizer power operated relief valves (PORVs) provide the LTOP function (Ref. 2). The PTLR provides the maximum allowable OPPS actuation setpoints for the PORVs and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES. The LTOP MODES are the MODES as defined in the Applicability statement of LCO 3.4.12 and LCO 3.4.13.

The pressurizer safety valves and PORVs at their normal setpoints do not provide overpressure protection for certain low temperature operational transients. Inadvertent pressurization of the RCS at temperatures below the OPPS enable temperature specified in the PTLR could result in exceeding the ASME Appendix G (Ref. 3) brittle fracture P/T limits. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

This LCO provides RCS overpressure protection by restricting coolant input capability and ensuring adequate pressure relief capacity. In MODE 4, above the safety injection (SI) pump disable temperature, limiting coolant input capability requires one (SI) pump incapable of injection into the RCS and isolating the emergency core cooling system (ECCS) accumulators. In MODE 4, above the SI

BASES

BACKGROUND (continued)

pump disable temperature, one PORV is the overpressure protection device that acts to terminate an increasing pressure event.

Limiting coolant input capability reduces the ability to provide core coolant addition. The LCO does not require the makeup control system deactivated or the SI actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the charging system can provide adequate flow. If conditions require the use of more than one SI pump for makeup in the event of loss of inventory, then pumps can be made available through manual actions.

In MODE 4, above the SI pump disable temperature, pressure relief consists of two PORVs with reduced lift settings. Two PORVs are required for redundancy. One PORV has adequate relieving capability to prevent overpressurization for the required coolant input capability.

As designed for the LTOP function, each PORV is signaled to open by OPPS if the RCS pressure approaches the lift setpoint provided when OPPS is enabled. The OPPS monitors both RCS temperature and RCS pressure and indicates when a condition not acceptable in the PTLR limits is approached. The wide range RCS temperature setpoints indicate conditions requiring enabling OPPS.

The PTLR presents the OPPS setpoints for LTOP.

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 2) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, and in MODE 4 with RCS cold leg temperature exceeding the OPPS enable temperature specified in the PTLR, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At about the OPPS enable temperature specified in the PTLR and below, overpressure prevention falls to two OPERABLE PORVs or to a depressurized

BASES

APPLICABLE SAFETY ANALYSES (continued)

RCS and a sufficiently sized RCS vent. Each of these means has a limited overpressure relief capability. LCO 3.4.13, "LTOP ≤ SI Pump Disable Temperature," provides the requirements for overpressure prevention at the lower temperatures.

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, LTOP must be re-evaluated to ensure its functional requirements can still be met using the PORV method.

The PTLR contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 2 analyses to determine the impact of the change on the LTOP acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients. The bounding mass input transient is inadvertent safety injection with injection from one SI pump and three charging pumps, and letdown isolated. The bounding heat input transient is reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

The following limitations are required during the Applicability of this specification to ensure that mass and heat input transients in excess of analysis assumptions do not occur:

a. Rendering one SI pump incapable of injection;

b. Deactivating the ECCS accumulator discharge isolation valves in their closed positions; and

c. Disallowing start of an RCP if secondary temperature is more than 50°F above primary temperature in any one loop. LCO 3.4.6, "RCS Loops - MODE 4," provides this protection.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The Reference 2 analyses demonstrate that one PORV can maintain RCS pressure below limits when only one SI pump and all charging pumps are actuated. Thus, the LCO allows only one SI pump OPERABLE during the Applicability of this specification.

Since one PORV cannot handle the pressure transient resulting from ECCS accumulator injection, when RCS temperature is low, the LCO also requires accumulator isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

The isolated ECCS accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions.

Fracture mechanics analyses established the temperature of LTOP Applicability at the OPPS enable temperature specified in the PTLR.

The consequences of a small break loss of coolant accident (LOCA) in LTOP MODE 4 above the SI Pump disable temperature conform to 10 CFR 50.46 and 10 CFR 50, Appendix K, requirements by having a maximum of one SI pump OPERABLE and SI actuation enabled.

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limit shown in the PTLR. The OPPS setpoints are derived by analyses that model the performance of the system, assuming the limiting LTOP transient of one SI pump and all charging pumps injecting into the RCS. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The OPPS setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met.

The OPPS setpoints in the PTLR will be updated when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are

BASES

APPLICABLE SAFETY ANALYSES (continued)

periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

The LTOP function satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

This LCO requires that LTOP be provided, by limiting coolant input capability and by OPERABLE pressure relief capability. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires that a maximum of one SI pump be capable of injecting into the RCS, and all ECCS accumulator discharge isolation valves be closed and deenergized (when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR).

The LCO is modified by two Notes. Note 1 allows operation of both SI pumps for ≤ 1 hour for conducting SI system testing providing there is a steam or gas bubble in the pressurizer and at least one isolation valve between the SI pump and the RCS is shut. The purpose of this note is to permit the conduct of the integrated SI test and other SI system tests and operations that may be performed in MODE 4. In this case, pressurizer level is maintained at less than 50% and a positive means of isolation is provided between the SI pumps and the RCS to prevent fluid injection to the RCS. This isolation is accomplished by either a closed manual valve or motor operated valve with the power removed. This combination of conditions under strict administrative control assure that overpressurization cannot occur.

BASES

LCO (continued)

Note 2 states that ECCS accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR (less allowance for instrument uncertainty). This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

To provide low temperature overpressure mitigation through pressure relief, the LCO requires an OPERABLE OPPS with two pressurizer PORVs. A PORV is OPERABLE for LTOP when its block valve is open, its low pressure lift setpoint has been selected (OPPS enabled), and the backup air supply is charged.

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ the OPPS enable temperature specified in the PTLR and > the SI Pump disable temperature specified in the PTLR. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above the OPPS enable temperature specified in the PTLR.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above the OPPS enable temperature specified in the PTLR. LCO 3.4.13 provides the LTOP requirements in MODE 4 ≤ SI pump disable temperature and in MODES 5 and 6.

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

BASES (continued)

ACTIONS

A.1

With two SI pumps capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

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

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

If isolation is needed and cannot be accomplished in 1 hour, Required Action C.1 and Required Action C.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > the OPPS enable temperature specified in the PTLR, an accumulator pressure of 800 psig cannot exceed the LTOP analysis limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit from the PTLR also gives this protection.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

D.1

In MODE 4 when any RCS cold leg temperature is ≤ the OPPS enable temperature specified in the PTLR, with one required PORV

BASES

ACTIONS

D.1 (continued)

inoperable, the PORV must be restored to OPERABLE status within a Completion Time of 7 days. Two PORVs are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

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

E.1

MODE 5 must be entered, the RCS must be depressurized and a vent must be established within 12 hours when:

a. Both PORVs are inoperable; or

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

c. The OPPS is inoperable.

The vent must be sized ≥ 3 square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. The vent opening is based on the cross sectional flow area of a PORV. A PORV maintained in the open position satisfies the vent requirement. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

BASES

ACTIONS

E.1 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1 and SR 3.4.12.2

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, one SI pump is verified incapable of injecting into the RCS and the ECCS accumulator discharge isolation valves are verified closed and deenergized.

The SI pump is rendered incapable of injecting into the RCS by employing at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the pump control switch being placed in pullout with a blocking device installed over the control switch that would prevent an unplanned pump start.

The ECCS accumulator motor operated isolation valves can be verified closed and deenergized by use of control board indication. SR 3.4.12.2 is modified by a Note specifying that this verification is only required when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. If accumulator pressure is less than this limit, no verification is required since the accumulator cannot pressurize the RCS to or above the OPPS setpoint.

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.3

The PORV block valve must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve may be remotely verified open in the main control room.

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

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

SR 3.4.12.4

Performance of a COT is required every 31 days on OPPS to verify and, as necessary, adjust the PORV lift setpoints. 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 OPERATIONAL 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 COT will verify the setpoints are within the PTLR allowed maximum limits in the PTLR. PORV actuation during this testing could depressurize the RCS and is not required.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.12.4 (continued)

A Note has been added indicating that this SR is required to be performed 12 hours after decreasing RCS cold leg temperature to ≤ the OPPS enable temperature specified in the PTLR. The COT may not have been performed before entry into the LTOP MODES. The 12 hour initial time considers the unlikehood of a low temperature overpressure event during this time.

SR 3.4.12.5

Performance of a CHANNEL CALIBRATION on OPPS is required every 24 months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

REFERENCES

1. 10 CFR 50, Appendix G.

2. USAR, Section 4.4.

3. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix G, with ASME Code Case N-514.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 Low Temperature Overpressure Protection (LTOP) ≤ Safety Injection Pump (SI) Pump Disable Temperature

BASES

BACKGROUND

The LTOP function limits RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. The Over Pressure Protection System (OPPS) provides the actuation setpoints for the pressurizer power operated relief valves (PORVs) for the LTOP function (Ref. 2). The PTLR provides the maximum allowable OPPS actuation setpoints and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES. The LTOP MODES are the MODES as defined in the Applicability statement of LCO 3.4.12 and LCO 3.4.13.

The pressurizer safety valves and PORVs at their normal setpoints do not provide overpressure protection for certain low temperature operational transients. Inadvertent pressurization of the RCS at temperatures below the OPPS enable temperature specified in the PTLR could result in exceeding the ASME Appendix G (Ref. 3) brittle fracture P/T limits. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

This LCO provides RCS overpressure protection by restricting coolant input capability and ensuring adequate pressure relief capacity. In MODE 4, at or below the safety injection (SI) pump disable temperature, limiting coolant input capability requires both SI pumps incapable of injection into the RCS and isolating the emergency core cooling system (ECCS) accumulators. The pressure

BASES

BACKGROUND (continued)

relief capacity requires either two redundant RCS relief valves or a depressurized RCS and an RCS vent of sufficient size. One PORV or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

Limiting coolant input capability reduces the ability to provide core coolant addition. The LCO does not require the makeup control system deactivated or the safety injection SI actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the charging system can provide adequate flow. If conditions require the use of an SI pump for makeup in the event of loss of inventory, the pump can be made available through manual actions.

The LTOP pressure relief consists of two PORVs with reduced lift settings or a depressurized RCS and an RCS vent of sufficient size. Two PORVs are required for redundancy. One PORV has adequate relieving capability to prevent overpressurization for the required coolant input capability.

OPPS and PORV Requirements

As designed for the LTOP function, each PORV is signaled to open by OPPS if the RCS pressure approaches the lift setpoint provided when OPPS is enabled. The OPPS monitors both RCS temperature and RCS pressure and indicates when a condition not acceptable in the PTLR limits is approached. The wide range RCS temperature setpoints indicate conditions requiring enabling OPPS. The PTLR presents the OPPS setpoints for LTOP.

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure

BASES

BACKGROUND

RCS Vent Requirements (continued)

in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient, and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 2) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, and in MODE 4 with RCS cold leg temperature exceeding the OPPS enable temperature specified in the PTLR, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At about the OPPS enable temperature specified in the PTLR and below, overpressure prevention falls to two OPERABLE PORVs or to a depressurized RCS and a sufficiently sized RCS vent. Each of these means has a limited overpressure relief capability. LCO 3.4.12, "LTOP > SI Pump Disable Temperature," provides the requirements for overpressure prevention at temperatures above the SIP disable temperature.

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, LTOP must be re-evaluated to ensure its functional requirements can still be met using the PORV method or the depressurized and vented RCS condition.

The PTLR contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 2 analyses to determine the impact of the change on the LTOP acceptance limits.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients. The bounding mass input transient is inadvertent safety injection with injection from one SI pump and three charging pumps, and letdown isolated. The bounding heat input transient is reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

The following limitations are required during the Applicability of this specification to ensure that mass and heat input transients in excess of analysis assumptions do not occur:

a. Rendering both SI pumps incapable of injection;

b. Deactivating the ECCS accumulator discharge isolation valves in their closed positions; and

c. Disallowing start of an RCP if secondary temperature is more than 50°F above primary temperature in any one loop. LCO 3.4.6, "RCS Loops - MODE 4," provides this protection.

The Reference 2 analyses demonstrate that either one PORV or the depressurized RCS and RCS vent can maintain RCS pressure below limits when all charging pumps are actuated. Neither one PORV nor the RCS vent can handle the pressure transient resulting from inadvertent SI pump or ECCS accumulator injection when the RCS is below the SI Pump disable temperature. Thus, the LCO requires both SI pumps to be disabled below the temperature specified in the PTLR.

The LCO also requires accumulator isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. The isolated ECCS accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Fracture mechanics analyses established the temperature of LTOP Applicability at the OPPS enable temperature specified in the PTLR. The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limit shown in the PTLR. The OPPS setpoints are derived by analyses that model the performance of the system, assuming the limiting LTOP transient of all charging pumps injecting into the RCS. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The OPPS setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met.

The OPPS setpoints in the PTLR will be updated when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

With the RCS depressurized, analyses show a vent size equivalent to the cross sectional flow area of a PORV is capable of mitigating the allowed LTOP overpressure transient. The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, both SI pumps disabled and all charging pumps OPERABLE when the RCS is below the SI Pump disable temperature, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

The RCS vent is passive and is not subject to active failure.

The LTOP function satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

This LCO requires that LTOP be provided, by limiting coolant input capability and by OPERABLE pressure relief capability. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO requires both SI pumps be incapable of injecting into the RCS, and all ECCS accumulator discharge isolation valves be closed and deenergized (when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR).

The LCO is modified by three Notes. Note 1 allows operation of both SI pumps for ≤ 1 hour for conducting SI system testing providing there is a steam or gas bubble in the pressurizer and at least one isolation valve between the SI pump and the RCS is shut. The purpose of this note is to permit the conduct of the integrated SI test and other SI system tests and operations that may be performed in MODES 4, 5 or 6. In this case, pressurizer level is maintained at less than 50% and a positive means of isolation is provided between the SI pumps and the RCS to prevent fluid injection to the RCS. This isolation is accomplished by either a closed manual valve or motor operated valve with the power removed. This combination of conditions under strict administrative control assure that overpressurization cannot occur.

Note 2 allows operation of an SI pump during reduced inventory conditions as required to maintain adequate core cooling and RCS inventory. The purpose of this note is to allow use of an SI pump in the event of a loss of other injection capability (e.g., loss of Residual Heat Removal System cooling while in reduced inventory conditions). The operation of an SI pump under such conditions would be controlled by an approved emergency operating procedure.

BASES

LCO (continued)

Note 3 states that ECCS accumulator isolation is only required when accumulator pressure is more than or at the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR (less allowance for instrument uncertainty). This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

a. An OPERABLE OPPS with two PORVs; or

   A PORV is OPERABLE for LTOP when its block valve is open, its low pressure lift setpoint has been selected (OPPS enabled), and the backup air supply is charged.

b. A depressurized RCS and an RCS vent.

   An RCS vent is OPERABLE when open with an area of ≥ 3.0 square inches. Because the RCS vent opening specification is based on the flow capacity of a PORV, a PORV maintained in the open position may be utilized to meet the RCS vent requirement.

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ the SI Pump disable temperature specified in the PTLR, in MODE 5, and in MODE 6 when the reactor vessel head is on and the SG primary system manways and pressurizer manway are closed and secured. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above the OPPS enable temperature specified in the PTLR. When the reactor vessel head is off, overpressurization cannot occur.

BASES

APPLICABILITY (continued)

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above the OPPS enable temperature specified in the PTLR. LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) ≤ Safety Injection Pump (SI) Pump Disable Temperature," provides the requirements for MODE 4 below the OPPS enable temperature and above the SI Pump disable temperature.

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

ACTIONS

A.1

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

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

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

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

If isolation is needed and cannot be accomplished in 1 hour, Required Action C.1 and Required Action C.2 provide two options, either of which must be performed in the next 12 hours. By

BASES

ACTIONS

B.1, C.1, and C.2 (continued)

increasing the RCS temperature to > the OPPS enable temperature specified in the PTLR, an accumulator pressure of 800 psig cannot exceed the LTOP analysis limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit from the PTLR also gives this protection.

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

D.1

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature. Thus, with one PORV inoperable in MODE 4 when any RCS cold leg temperature is ≤ the SI Pump disable temperature specified in the PTLR, MODE 5 or in MODE 6 with the head on, the Completion Time to restore two valves to OPERABLE status is 24 hours. A Note clarifies that Condition D is only applicable when the OPPS and PORVs are being used to satisfy the pressure relief requirements of LCO 3.4.13.a.

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

E.1

The RCS must be depressurized and a vent must be established within 8 hours when:

BASES

ACTIONS

E.1 (continued)

a. Both required PORVs are inoperable; or

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

c. The OPPS is inoperable.

The vent must be sized ≥ 3 square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. The vent opening is based on the cross sectional flow area of a PORV. A PORV maintained in the open position satisfies the vent requirement. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

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

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 and SR 3.4.13.2

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

The SI pumps are rendered incapable of injecting into the RCS by employing at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 and SR 3.4.13.2 (continued)

pump control switch being placed in pullout with a blocking device installed over the control switch that would prevent an unplanned pump start.

The ECCS accumulator motor operated isolation valves can be verified closed and deenergized by use of control board indication. SR 3.4.13.2 is modified by a Note specifying that this verification is only required when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. If accumulator pressure is less than this limit, no verification is required since the accumulator cannot pressurize the RCS to or above the OPPS setpoint.

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

SR 3.4.13.3

The RCS vent of ≥ 3 square inches is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that is not locked, sealed, or secured in the open position.

b. Once every 31 days for other vent path(s) (e.g., a vent valve that is locked, sealed, or secured in position). A removed pressurizer safety valve or open manway also fits this category.

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.13.4

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

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

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

SR 3.4.13.5

Performance of a COT is required every 31 days on OPPS to verify and, as necessary, adjust the PORV lift setpoints. 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 OPERATIONAL 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 COT will verify the setpoints are within the PTLR allowed maximum limits in the PTLR. PORV actuation during this testing could depressurize the RCS and is not required.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.5 (continued)

Note 1 has been added indicating that this SR is not required to be performed until 12 hours after decreasing RCS cold leg temperature to ≤ the OPPS enable temperature specified in the PTLR. The COT may not have been performed before entry into the LTOP MODES. The 12 hour initial time considers the unlikehood of a low temperature overpressure event during this time.

Note 2 has been added to specify that this SR is only required to be performed when OPPS and PORVs are providing the LTOP function per LCO 3.4.13.a.

SR 3.4.13.6

Performance of a CHANNEL CALIBRATION on OPPS is required every 24 months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

REFERENCES

1. 10 CFR 50, Appendix G.
2. USAR, Section 4.4.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix G, with ASME Code Case N-514.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. RCS component joints are made by welding, bolting, rolling, or pressure loading. Valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

AEC GDC Criterion 16 (Ref. 1), requires means for monitoring the reactor coolant pressure boundary to detect LEAKAGE. LCO 3.4.16 "RCS Leakage Detection Instrumentation," describes requirements for leakage detection instrumentation.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

BASES (continued)

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1 gpm primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The USAR (Ref. 2) analysis for SGTR assumes the plant has been operating with a 5 gpm primary to secondary leak rate for a period of time sufficient to establish radionuclide equilibrium in the secondary loop. Following the tube rupture, the initial primary to secondary LEAKAGE is relatively inconsequential when compared to the mass transfer through the ruptured tube.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes 1 gpm (at 70°F) primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits).

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the reactor coolant pressure boundary (RCPB). LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

Seal welds are provided at the threaded joints of all reactor vessel head penetrations (spare penetrations, full-length Control Rod Drive Mechanisms, and thermocouple columns). Although these seals are part of the RCPB as defined in 10CFR50 Section 50.2, minor leakage past the seal weld is not a fault in the RCPB or a structural integrity concern. Pressure retaining components are differentiated from leakage barriers in the ASME Boiler and Pressure Vessel Code. In all cases, the joint strength is provided by the threads of the closure joint.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere

BASES

LCO

c. Identified LEAKAGE (continued)

with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified leakage must be evaluated to assure that continued operation is safe. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Any One Steam Generator (SG)

The 150 gallons per day (gpd) limit on one SG is based on implementation of the Steam Generator Voltage Based Alternate Repair Criteria and is more restrictive than standard operating leakage limits to provide additional margin to accommodate a crack which might grow at greater than the expected rate or unexpectedly extend outside the thickness of the tube support plate.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.4.15, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

BASES (continued)

ACTIONS

A.1

Unidentified LEAKAGE in excess of the LCO limits must be identified or reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

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

If unidentified LEAKAGE cannot be identified or cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals, gaskets, and pressurizer safety valves seats is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours. If the LEAKAGE source cannot be identified within 54 hours, then the reactor must be placed in MODE 5 within 84 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

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. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

C.1, C2.1, and C2.2

If RCS identified LEAKAGE, other than pressure boundary leakage, is not within limits, then the reactor must be placed in MODE 3 within 6 hours. In this condition, 14 hours are allowed to reduce the identified leakage to within limits. If the identified LEAKAGE is not

BASES

ACTIONS

C.1, C2.1, and C2.2 (continued)

within limits within this time, the reactor must be placed in MODE 5 within 44 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions in an orderly manner without challenging plant systems.

D.1 and D.2

If RCS pressure boundary LEAKAGE exists or if SG LEAKAGE (150 gpd limit) is not within limits, the reactor must be placed in MODE 3 within 6 hours and in MODE 5 within 36 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions in an orderly manner without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

The RCS water inventory balance must be met with the reactor at steady state operating condition (stable temperature, power level, equilibrium xenon, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). Therefore, a Note is added allowing that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by monitoring containment atmosphere radioactivity. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.16, "RCS Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.14.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam

BASES

SURVEILLANCE REQUIREMENTS	<u>SR 3.4.14.2</u> (continued) Generator Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.
REFERENCES	1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criterion 16, issued for comment July 10, 1967, as referenced in USAR, Section 1.2. 2. USAR, Section 14.5.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

RCS PIVs separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

The PIV leakage limit applies to each individual valve. Leakage through both series PIVs in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.14, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 3.4.14.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressurization of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident, that could degrade the ability for low pressure injection.

The basis for this LCO is the 1975 NRC "Reactor Safety Study" (Ref. 1) that identified potential intersystem LOCAs as a significant

BASES

BACKGROUND (continued)	contributor to the risk of core melt. A subsequent study (Ref. 2) evaluated various PIV configurations to determine the probability of intersystem LOCAs. PIVs are provided to isolate the RCS from low pressure systems susceptible to intersystem LOCAs. The PIVs are listed in the LCO section of these Bases. Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.
APPLICABLE SAFETY ANALYSES	Reference 1 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the Residual Heat Removal (RHR) System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. The low pressure portion of the RHR System is designed for 600 psig. An overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent increased risk of core melt. Reference 2 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA. A plant specific review against the NRC criteria for intersystem LOCAs was performed to identify the most risk significant configurations (Ref. 3). Valves identified in this study are listed in the LCO discussion in these Bases. RCS PIV leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

BASES (continued)

LCO

RCS PIV OPERABILITY protects the low pressure systems attached to the RCS from potential failure due to overpressurization. This protection (that is, RCS PIV OPERABILITY) is provided by the leak tight PIVs.

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken. This LCO only applies to the following PIVs which were determined to be in the most risk significant configurations (Ref. 4):

a. Residual Heat Removal (RHR) System, RHR to loop B accumulator injection line:

Unit 1	SI-6-2
Unit 2	2SI-6-2

b. Safety Injection (SI) System, low pressure SI to upper plenum:

Unit 1	SI-9-3, SI-9-4, SI-9-5, SI-9-6
Unit 2	2SI-9-3, 2SI-9-4, 2SI-9-5, 2SI-9-6

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.

Reference 4 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal

BASES

LCO (continued)

pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

ACTIONS

The Actions are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

<u>A.1 and A.2</u>

The flow path must be isolated by two valves. Required Action A.1 is modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be within the high pressure portion of the system.

Required Action A.1 requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected

BASES

ACTIONS

A.1 and A.2 (continued)

system if leakage cannot be reduced. The 4 hour Completion Time allows the actions and restricts the operation with leaking isolation valves.

Required Action A.2 specifies that the leaking PIV be restored within limits.

The 72 hour Completion Time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This timeframe considers the time required to complete this Action and the low probability of a second valve failing during this period.

B.1 and B.2

If leakage cannot be reduced, the system isolated, or the other Required Actions accomplished, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. 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 3.4.15.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 is required to verify that leakage is below the specified limit and to identify each leaking valve.

The leakage limit of 0.5 gpm per inch of nominal valve diameter up

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1 (continued)

to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition of at least 150 psid.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed at the following times:

a. Every 24 months, a typical refueling cycle;

b. Prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months; and

c. Prior to returning a PIV to service after maintenance, repair, or replacement work has been performed.

The 24 month Frequency is consistent with 10 CFR 50.55a(g) as contained in the Inservice Testing Program, is within the frequency allowed by Reference 4, and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1 (continued)

In addition, testing must be performed once after the valve has been worked on (maintenance, repair, or replacement) to ensure tight reseating.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures. A differential pressure of at least 150 psid is sufficient to ensure the valves are seated.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

REFERENCES

1. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
2. NUREG-0677, May 1980.
3. Letter from Robert A. Clark, NRC, to L. O. Mayer, NSP, Subject: "Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves," dated April 20, 1981.
4. American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section XI.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

AEC GDC 16 (Ref. 1) requires that means be provided for monitoring reactor coolant pressure boundary (RCPB) to detect RCS LEAKAGE. Reference 2 describes methods used for RCS leakage detection.

Leakage detection systems must have the capability to detect significant RCPB degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can be readily detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The containment sump A pump run time instrumentation may be used to detect increases in unidentified LEAKAGE.

The reactor coolant contains radioactivity that, when released to the containment, can be detected by radiation monitoring instrumentation. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivities of 10^{-9} μCi/cc radioactivity for particulate monitoring and of 10^{-6} μCi/cc radioactivity for gaseous monitoring are practical for these leakage detection systems. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS LEAKAGE.

BASES

BACKGROUND (continued)	An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Humidity measurements can be used as a less sensitive indicator of potential RCS LEAKAGE (Ref. 2). Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed changes in containment sump A pump run time. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required by this LCO. Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS leakage into the containment. The relevance of temperature and pressure measurements are affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.
APPLICABLE SAFETY ANALYSES	The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system response times and sensitivities are described in the USAR (Refs. 2 and 3). Multiple instrument locations are utilized, if needed, to ensure that the transport delay time of the leakage from its source to an instrument location yields an acceptable overall response time.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS LEAKAGE into containment is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE provides quantitative information to the operators, allowing them to take corrective action should a leakage occur detrimental to the safety of the unit and the public.

Containment radionuclide monitoring used for RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii). Containment sump A monitoring used for RCS leakage detection instrumentation satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

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

The LCO is satisfied when monitors of diverse measurement means are available to provide indication of RCS leakage. Thus, the containment sump A monitor (pump run time instrumentation), in combination with a containment radionuclide monitor, provides an acceptable minimum.

One of the containment atmosphere (gaseous and particulate) radiation monitoring channels, R-11 or R-12, normally provides the required monitoring.

BASES (continued)

APPLICABILITY

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

In MODE 5 or 6, the temperature is to be ≤ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

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

A.1 and A.2

With the required containment sump monitor inoperable, no other form of sampling can provide the equivalent information; however, the containment radionuclide monitor will provide indications of changes in leakage. Together with the radionuclide monitor, the periodic surveillance for RCS water inventory balance, SR 3.4.14.1, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage. A Note is added allowing that SR 3.4.14.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, equilibrium xenon, pressurizer and makeup tank levels, makeup and letdown, and reactor coolant pump seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

BASES

ACTIONS

A.1 and A.2 (continued)

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

B.1.1, B.1.2, and B.2

When the required containment radionuclide monitoring instrumentation channel is inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.14.1, must be performed to provide alternate periodic information.

With a sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of the required containment radionuclide monitor. The 24 hour interval provides periodic information that is adequate to detect leakage. The 30 day Completion Time recognizes at least one other form of leakage detection is available.

C.1 and C.2

If a Required Action of Condition A or B cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating

BASES

ACTIONS

C.1 and C.2 (continued)

experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1

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

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires the performance of a CHANNEL CHECK of the required containment radionuclide monitor. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.16.2

SR 3.4.16.2 requires the performance of a COT on the required containment radionuclide monitor. The test ensures that the monitor can perform its function in the desired manner. 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 OPERATIONAL TEST of a relay. This is acceptable because all of the 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 test verifies the alarm

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.16.2 (continued)

setpoint and relative accuracy of the instrument string. The Frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

SR 3.4.16.3 and SR 3.4.16.4

These SRs require the performance of a CHANNEL CALIBRATION for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 24 months is a typical refueling cycle and considers channel reliability.

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction Permits," Criterion 16, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

2. USAR, Section 6.5.

3. USAR, Section 7.5.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 RCS Specific Activity

BASES

BACKGROUND

The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.

The parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following a SGTR accident with an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are

BASES

APPLICABLE SAFETY ANALYSES (continued)	conservative in that specific site parameters of the Prairie Island site, such as site boundary location and meteorological conditions, were not considered in this evaluation (Ref. 2). The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices. RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The specific iodine activity is limited to 1.0 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the Design Basis Accident (DBA) will be a small fraction of the allowed thyroid dose. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose. The SGTR accident analysis (Ref. 3) shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits.
APPLICABILITY	In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values.

BASES

APPLICABILITY (continued)

For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 specific activity greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the limits of Figure 3.4.17-1 are not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is done to continue to provide a trend.

The DOSE EQUIVALENT I-131 specific activity must be restored to within limits within 48 hours. The Completion Time of 48 hours is required, if the limit violation resulted from normal iodine spiking.

Permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 μCi/gm DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4.17-1, accommodates the possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 μCi/gm DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4.17-1 should be minimized since the activity levels allowed by the figure increase the dose at the site boundary following a postulated steam generator tube rupture.

A Note to the ACTIONS excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is

BASES

ACTIONS

A.1 and A.2 (continued)

limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1

With the gross specific activity in excess of the allowed limit, the reactor must be placed in a MODE in which the requirement does not apply. The change within 6 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves and prevents venting the SG to the environment in a SGTR event. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner without challenging plant systems.

C.1

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 specific activity is in the unacceptable region of Figure 3.4.17-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner without challenging plant systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

SR 3.4.17.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with T_{avg} at least 500°F. The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.

SR 3.4.17.2

This Surveillance is performed in MODE 1 only to ensure DOSE EQUIVALENT I-131 specific activity remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the DOSE EQUIVALENT I-131 specific activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.17.3

A radiochemical analysis for $\bar{E}$ determination is required every 184 days (6 months) with the plant operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for $\bar{E}$ is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.

This SR has been modified by a Note that indicates sampling is required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. 10 CFR 100.11, 1973.

2. Letter from Dominic C. DiIanni, NRC, to L. O. Mayer, NSP, dated December 4, 1981.

3. USAR, Section 14.5.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.18 RCS Loops - Test Exceptions

BASES

BACKGROUND

The primary purpose of this test exception is to provide an exception to LCO 3.4.4, "RCS Loops - MODES 1 and 2," to permit reactor criticality under no flow conditions during certain PHYSICS TESTS (natural circulation demonstration, station blackout, and loss of offsite power) to be performed while at low THERMAL POWER levels. Section XI of 10 CFR 50, Appendix B, 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 the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the power plant as specified in AEC GDC Criterion 1 (Ref. 1).

The key objectives of a test program are to provide assurance that the facility has been adequately designed to validate the analytical models used in the design and analysis, to verify the assumptions used to predict plant response, to provide assurance that installation of equipment at the unit has been accomplished in accordance with the design, and to verify that the operating and emergency procedures are adequate. Testing is performed prior to initial criticality, during startup, and following low power operations.

APPLICABLE SAFETY ANALYSES

Operating the plant without forced convection flow is not bounded by any safety analyses. However, operating experience has demonstrated this exception to be safe under the present applicability.

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36(c)(2)(ii) apply. Test Exception LCOs provide flexibility to perform certain

BASES

APPLICABLE SAFETY ANALYSES (continued)	operations by appropriately modifying requirements of other LCOs. A discussion of the criteria for the other LCOs is provided in their respective Bases.
LCO	This LCO provides an exemption to the requirements of LCO 3.4.4. The LCO is provided to allow for the performance of PHYSICS TESTS in MODE 2 (after a refueling), where the core cooling requirements are significantly different than after the core has been operating. Without the LCO, plant operations would be held bound to the normal operating LCOs for reactor coolant loops and circulation (MODES 1 and 2), and the appropriate tests could not be performed. In MODE 2, where core power level is considerably lower and the associated PHYSICS TESTS must be performed, operation is allowed under no flow conditions provided THERMAL POWER is ≤ P-7 and the reactor trip setpoints of the OPERABLE power level channels are set ≤ the allowable value of Table 3.3.1-1, Function 2.b. This ensures, if some problem caused the plant to enter MODE 1 and start increasing plant power, the Reactor Trip System (RTS) would automatically shut it down before power became too high, and thereby prevent violation of fuel design limits. The exemption is allowed even though there are no bounding safety analyses. However, these tests are performed under close supervision during the test program and provide valuable information on the plant's capability to cool down without offsite power available to the reactor coolant pumps.
APPLICABILITY	This LCO is applicable when performing low power PHYSICS TESTS without any forced convection flow. This testing is performed to establish that heat input from nuclear heat does not

BASES

APPLICABILITY (continued)

exceed the natural circulation heat removal capabilities. Therefore, no safety or fuel design limits will be violated as a result of the associated tests.

ACTIONS

A.1

When THERMAL POWER is ≥ the P-7 interlock setpoint, the only acceptable action is to ensure the reactor trip breakers (RTBs) are opened immediately in accordance with Required Action A.1 to prevent operation of the fuel beyond its design limits. Opening the RTBs will shut down the reactor and prevent operation of the fuel outside of its design limits.

SURVEILLANCE REQUIREMENTS

SR 3.4.18.1

Verification that the power level is < the P-7 interlock setpoint will ensure that the fuel design criteria are not violated during the performance of the PHYSICS TESTS. The Frequency of once per hour is adequate to ensure that the power level does not exceed the limit. Plant operations are conducted slowly during the performance of PHYSICS TESTS and monitoring the power level once per hour is sufficient to ensure that the power level does not exceed the limit.

SR 3.4.18.2

The power range and intermediate range neutron channels and the P-7 interlock setpoint must be verified to be OPERABLE and adjusted to the proper value. A COT is performed prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. 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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.18.2 (continued)

This clarifies what is an acceptable CHANNEL OPERATIONAL TEST of a relay. This is acceptable because all of the 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.

REFERENCES

1. AEC "General Design Criteria for Nuclear Power Plant Construction," Criterion 1, issued for comment July 10, 1967, as referenced in USAR Section 1.2.

PACKAGE 3.4

REACTOR COOLANT SYSTEM (RCS)

PART C

MARKUP OF PRAIRIE ISLAND CURRENT TECHNICAL SPECIFICATIONS

List of Pages

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

Addressed Elsewhere

~~3.10.I. Monitor Inoperability Requirements~~

~~1. If the rod bank insertion limit monitor is inoperable, or if the rod position deviation monitor is inoperable, individual rod positions shall be logged once per shift, after a load change greater than 10 percent of RATED THERMAL POWER, and after 30 inches or more of rod motion.~~

~~2. If both the rod position deviation monitor and one or both of the quadrant power tilt monitors are inoperable for 2 hours or more, the nuclear overpower trip shall be reset to 93% of RATED THERMAL POWER in addition to the increased surveillance requirements.~~

~~3. If one or both of the quadrant power tilt monitors is inoperable, individual upper and lower excore detector calibrated outputs and the calculated power tilt shall be logged every two hours after a load change greater than 10% of RATED THERMAL POWER.~~

~~J.~~ ~~RCS Pressure, Temperature and Flow~~ DNB ~~Limits~~ ~~Parameters~~

LCO3.4.1

The following DNB related parameters limits shall be maintained during POWER OPERATION:

LR3.4-01

a. Reactor Coolant System Tavg ≤ limit specified in the COLR ~~564°F~~

b. Pressurizer Pressure ≥ limit specified in the COLR ~~2220 psia~~*

c. Reactor Coolant Flow ≥the value specified in the CORE OPERATING LIMITS REPORT

LCO3.4.1 Cond A

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours

LCO3.4.1 Cond B

or reduce THERMAL POWER to MODE 2 ~~less than 5% of RATED THERMAL POWER~~ within the next 64 hours.

A3.4-08

L3.4-05

SR3.4.1.1
SR3.4.1.2
SR3.4.1.3

Compliance with a. and b. is demonstrated by verifying that each of the parameters is within its limits at least once each 12 hours. Compliance with c. is demonstrated by verifying that the parameter is within its limit after each refueling cycle.

LCO3.4.1 NOTE

L3.4-02

*Limit not applicable during either a THERMAL POWER ramp ~~increase~~ in excess of (5%) RATED THERMAL POWER per minute or a THERMAL POWER step ~~increase~~ in excess of (10%) RATED THERMAL POWER

3.4~~1~~ REACTOR COOLANT SYSTEM

~~Applicability~~

~~Applies to the operating status of the reactor coolant system when irradiated fuel is in the containment.~~ A3.4-03

~~Objective~~

~~To specify those limiting conditions for operation of the reactor coolant system which must be met to assure safe reactor operation.~~

Specification

A. Operational Components

1. Reactor Coolant Loops and Coolant Circulation

a. Reactor Critical

LCO3.4.4 (1) A reactor shall not be in MODE 2~~made or maintained critical~~ unless both reactor coolant loops (with their associated steam generator and reactor coolant pump) are in operation, except 1) during low power PHYSICS TESTS M3.4-04

LCO3.4.18 with thermal power <P-7 (if power >P-7, open Reactor Trip Breaker) M3.4-06

SR3.4.18.1
SR3.4.18.2 New SR, Verify power <P-7 and perform COT on P-7, power (low setpoint) and intermediate range neutron flux channels. M3.4-07

or 2) as specified in 3.1.A.1.a.(2) below.

A3.4-08

LCO3.4.4 (2) With less than the above required reactor coolant loops in operation, be in at least MODE 3~~HOT SHUTDOWN~~ within 6 hours.

SR3.4.4.1 New SR, Verify each RCS loop in operation M3.4-11

b. Reactor Coolant System MODE 3~~Average Temperature Above 350°F~~ M3.4-12

LCO3.4.5 (1) Reactor coolant system ~~average temperature~~ shall not be in MODE 3~~exceed 350°F~~ unless both reactor coolant loops (with their associated steam generator and reactor coolant pump) are OPERABLE and both RCS loops shall be in operation when the Rod Control System is capable of rod withdrawal, or A3.4-08 M3.4-13

~~TS 3.1-1 Overflow~~

~~with~~ at least one reactor coolant loop in operation when the Rod Control System is not capable of rod withdrawal. M3.4-13

(except as specified in 3.1.A.1.b(2) and 3.1.A.1.b.(3) below).

LCO3.4.5 Cond A Cond B

(2) A reactor coolant loop may be inoperable for 72 hours ~~provided STARTUP OPERATION is discontinued until OPERABILITY is restored.~~ If OPERABILITY is not restored within the time specified, be in MODE 4 ~~reduce reactor coolant system average temperature below 350°F~~ within the next 12 ~~6~~ hours. A3.4-14 A3.4-08 L3.4-16

(3) With both reactor coolant pumps inoperable or not in operation immediately:

LCO3.4.5 Cond D

(a) De-energize all control rod drive mechanisms,

(b) Suspend all operations involving a reduction of RCS boron concentration,

LCO3.4.5 LCO Note

(c) Establish and maintain the core outlet temperature at least 10 °F below saturation temperature, and

LCO3.4.5 Cond D

(c) Initiate action to restore one reactor coolant pump to OPERABLE status and operation.*

~~If at least one reactor coolant pump is not restored to OPERABILITY and operation within 72 hours, reduce reactor coolant system average temperature to below 350°F within the next 12 hours. While applicable, this specification supercedes 3.1.A.1.b(2).~~ A3.4-18

LCO3.4.5 NOTE

* If the RCP shutdown or inoperability was due to preplanned work activities such as testing, switching, or maintenance, immediate restoration action is not required, but if at least one reactor coolant pump is not restored to operability and operation within 12 hours,

~~reduce reactor coolant system average temperature to below 350 °F within the next 12 hours.~~ A3.4-22

De-energizing control rods is not required for preplanned work activities. L3.4-23

LCO3.4.5 Cond C

If one RCS loop is not in operation with Rod Control System capable of rod withdrawal, within one hour restore the loop to operation, or render the Rod Control System inoperable. M3.4-17

SR3.4.5.1 SR3.4.5.2 SR3.4.5.3

New SRs, Verify RCS loops operating, verify SG level >60% WR, verify breaker alignment for non-operating pump. M3.4-21

~~3.1.A.1.c.~~ Reactor Coolant System Mode 4 and Mode 5, loops filled~~Average Temperature Below 350°F (and Reactor Coolant Level Above the Reactor Vessel Flange)~~ A3.4-08

(1) Whenever the reactor coolant system ~~average temperature~~ is in MODE 4 and MODE 5, loops filled ~~below 350°F,~~ except during REFUELING, at least two methods for removing decay heat shall be OPERABLE with one in operation* (except as specified in 3.1.A.1.c.(2) below). Acceptable methods for removing decay heat are at least one reactor coolant loop~~pump and its associated steam generator~~ (in MODE 5, SG level shall be ≥60% WR); or a residual heat removal loop ~~including a pump and its associated heat exchanger~~.

LCO3.4.6
LCO3.4.7

LR3.4-24

M3.4-26

(2) With only one OPERABLE method of removing decay heat, initiate prompt action to restore two OPERABLE methods of removing decay heat.

LCO3.4.6
LCO3.4.7
Cond A

LR3.4-24

If the remaining operable method is an RHR loop, be in MODE 5~~COLD SHUTDOWN~~ within 24 hours.

LCO3.4.6
Cond B

A3.4-08

(3) With no OPERABLE methods of removing decay heat, suspend all operations involving a reduction in boron concentration of the reactor coolant system and initiate prompt action to restore one OPERABLE method of removing decay heat.

LCO3.4.6
Cond C
LCO3.4.7
Cond B

(4) A reactor coolant pump may be started at RCS temperature less than the Over Pressure Protection System Enable Temperature specified in the PTLR, only if either of the following conditions is met:

LCO3.4.6
NOTE 2
LCO3.4.7
NOTE 3

There is a steam or gas bubble in the pressurizer, or

The (steam generator minus RCS) temperature difference for the steam generator in that loop is less than 50°F.

LCO3.4.7
NOTE 4

Both RHR loops may be removed from operation for planned heatup to MODE 4 when at least one RCS loop is in operation.

A3.4-28

SR3.4.6.1
SR3.4.6.2
SR3.4.6.3

New SRs, Verify one RHR or RCS loop in operation, verify SG level > 60% WR, verify pump breaker alignment

M3.4-31

SR3.4.7.1
SR3.4.7.2
SR3.4.7.3

New SRs, Verify one RHR loop in operation, verify SG level > 60% WR, verify pump breaker alignment

M3.4-32

~~TS 3.1-2 Overflow~~

A3.4-08

d. Reactor Coolant System, Mode 5, loops not filled~~Level Below or at the Reactor Vessel Flange~~

LR3.4-24

LCO3.4.8

(1) Both residual heat removal loops, ~~each consisting of a pump and its associated heat exchanger,~~ shall be OPERABLE with one in operation* (except as specified in 3.1.A.1.d.(2) below).

LCO3.4.8 Cond A,B

(2) With one or both residual heat removal loop(s) inoperable, prompt action shall be taken to restore the inoperable residual heat removal loop(s) to an OPERABLE status~~.~~

LCO3.4.8 Cond B

and, if both RHR loops are inoperable, also suspend all operations involving reduction in RCS boron concentration. M3.4-33

LCO3.4.8 LCO3.4.13 Notes

During reduced inventory conditions, a safety injection pump may be run as required to maintain adequate core cooling and RCS inventory in the event of a loss of Residual Heat Removal System cooling.

LCO3.4.6 LCO3.4.7 NOTE 1

*All pumps may be shutdown for up to one hour per 8 hour period provided the reactor is subcritical, no operations are permitted that would cause dilution of the reactor coolant boron concentration and core outlet temperature is maintained at least 10°F below saturation temperature. M3.4-38

LCO3.4.8 NOTE 1

* All pumps may be shutdown for up to 15 minutes provided the reactor is subcritical, no operations are permitted that would cause dilution of the reactor coolant boron concentration, core outlet temperature is maintained at least 10°F below saturation temperature, and no RCS draining operations are permitted. M3.4-34

LCO3.4.7 LCO3.4.8 NOTE 2

One required RHR loop may be inoperable for ≤ 2 hours for surveillance provided that the other RHR loop is OPERABLE and in operation. L3.4-36

SR3.4.8.1 SR3.4.8.2

New SRs, Verify one RHR loop in operation, verify breaker alignment for RHR pump not operating M3.4-37

~~3.1.A.2 Reactor Coolant System Pressure Control~~

a. Pressurizer

A3.4-08

LCO3.4.9

(1) When ~~A~~ reactor is in MODE 1, 2, and 3, the pressurizer shall be OPERABLE with~~shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 350°F unless~~

pressurizer level ≤ 90% ~~there is a steam bubble in the pressurizer~~ and two heater groups ~~A and B~~ are operable and capable of being powered from an emergency power supply. M3.4-41 LR3.4-24

(except as specified in 3.1.A.2.a.2 and 3.1.A.2.a.3 below).

A3.4-08

LCO3.4.9 Cond B,C

(2) During Modes 1, 2, and 3~~STARTUP OPERATION or POWER OPERATION,~~

~~Group A or B~~ one pressurizer heater group may be inoperable for 72 hours

A3.4-14

~~provided STARTUP OPERATION is discontinued until OPERABILITY is restored.~~

If OPERABILITY is not restored within the time specified, be in at least MODE 3~~HOT SHUTDOWN~~ within the next 6 hours and MODE 4~~reduce reactor coolant system average temperature below 350°F~~ within 12~~the following 6~~ hours. A3.4-08 A3.4-39

LCO3.4.9 Cond A

(3) With the pressurizer otherwise inoperable, ~~within one hour~~ initiate the action necessary to place the unit in M3.4-42

~~HOT SHUTDOWN, and be in~~ at least MODE 3~~HOT SHUTDOWN~~ within the next 6 hours, A3.4-08

fully insert all rods within the next 6 hours, render Control Rod system incapable of rod withdrawal within the next 6 hours, M3.4-43

and be in MODE 4~~reduce reactor coolant average temperature below 350°F~~ within 12~~the following 6~~ hours. A3.4-08 A3.4-39

SR3.4.9.1 New SR, Verify steam bubble in pressurizer. M3.4-44

b. Pressurizer Safety Valves

LCO3.4.10 (1) Reactor Coolant System Modes 1, 2, 3, and 4 with all cold leg ~~average~~temperatures greater than OPPS enable temperature specified in the PTLR~~or equal to 350°F~~ A3.4-08 M3.4-45

LCO3.4.10 When A reactor is in MODE 1, 2, 3, and 4 with all ~~shall not be made or maintained critical nor shall~~ reactor coolant system cold leg ~~average~~ temperatures ~~exceed~~ OPPS enable temperature specified in the PTLR, ~~350°F unless~~ two pressurizer safety valves shall be~~are~~ OPERABLE, with lift settings of 2485 psig ±3 ~~1~~%. A3.4-08 M3.4-45 A3.4-46

LCO3.4.10 Cond A If these conditions cannot be satisfied, ~~discontinue STARTUP OPERATION and~~ within 15 minutes restore valve to OPERABLE status~~initiate the action necessary to place the unit in HOT SHUTDOWN~~, and A3.4-14

LCO3.4.10 Cond B be in at least MODE 3~~HOT SHUTDOWN~~ within the next 6 hours and ~~reduce~~ be in MODE 4 with any reactor coolant system cold leg ~~average~~ temperature below the OPPS enable temperature specified in the PTLR ~~350°F~~ within 24~~the following 6~~ hours. A3.4-08 M3.4-45

LCO3.4.10 NOTE NOTE: The lift settings are not required to be within the LCO limits during MODE 3 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 36 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup. L3.4-48

L3.4-47

~~2) Reactor Coolant System Average Temperature below 350°F~~

~~At least one pressurizer safety valve shall be OPERABLE, with a lift setting of 2485 psig ±1%, whenever the head is on the reactor vessel, except during hydrostatic tests. With no pressurizer safety valve OPERABLE, promptly place an OPERABLE residual heat removal loop into operation.~~

~~3.1.A.2.c~~ Pressurizer Power Operated Relief Valves

A3.4-08

LCO3.4.11

(1) Modes 1, 2, and 3 (Reactor Coolant System average temperature greater than or equal to 350°F)

A3.4-08

LCO3.4.11

(a) In MODES 1, 2, and 3 ~~Reactor coolant system average temperature shall not exceed 350° F*unless~~ two power operated relief valves (PORVs) and their associated block valves are OPERABLE (except as specified in 3.1.A.2.c(1)(b) below).

A3.4-14

LCO3.4.11 Cond D,G

A3.4-08

(b) ~~During STARTUP OPERATION or POWER OPERATION, any one of the following conditions of inoperability may exist for each unit.~~ If OPERABILITY is not restored within the time specified or the required action cannot be completed, be in ~~at least~~ MODE 3 ~~HOT SHUTDOWN~~ within the next 6 hours and MODE 4 ~~reduce reactor coolant system average temperature below 350°F*~~

A3.4-08

within 12 ~~the following 6~~ hours.

A3.4-39

LCO3.4.11 Cond A

A3.4-49

1. With one or both PORVs inoperable because of excessive seat leakage, within one hour ~~either restore the PORV(s) to OPERABLE status or~~ close the associated block valve(s) with power maintained to the block valve(s).

LCO3.4.11 Cond B

A3.4-49

2. With one PORV inoperable due to causes other than excessive seat leakage, within one hour ~~either restore the PORV to OPERABLE status or~~ close and remove power from the associated block valve. Restore the PORV to OPERABLE status within the following 72 hours.

LCO3.4.11 Cond E

A3.4-49

3. With both PORVs inoperable due to causes other than excessive seat leakage, within one hour ~~either restore at least one PORV to OPERABLE status or~~ close and remove power from the associated block valves and be in ~~at least~~ MODE 3 ~~HOT SHUTDOWN~~ within the next 6 hours and MODE 4 ~~reduce reactor coolant system average temperature below 350°F*~~

A3.4-08

within 12 ~~the following 6~~ hours.

A3.4-39

LCO3.4.11 Cond C

A3.4-49

4. With one block valve inoperable, within one hour ~~either restore the block valve to OPERABLE status or~~ place its associated PORV in manual control. Restore the block valve to OPERABLE status within the following 72 hours.

LCO3.4.11 Cond F

A3.4-49

5. With both block valves inoperable, within two hours ~~one hour either restore the block valves to OPERABLE status or place the PORVs in manual control.~~ Restore ~~at least~~ one block valve to OPERABLE status ~~within the next hour~~.

LCO3.4.12 (2) Reactor Coolant System average temperature greater than or equal to the temperature specified in the PTLR for disabling both safety injection pumps and below the Over Pressure Protection System Enable Temperature specified in the PTLR

LCO3.4.12 With Reactor Coolant System temperature greater than or equal to the temperature specified in the PTLR for disabling both safety injection pumps and less than the Over Pressure Protection System Enable Temperature specified in the PTLR; with a maximum of one SI pump capable of injecting into the RCS and the ECCS accumulators** isolated, both pressurizer power operated relief valves (PORVs) shall be OPERABLE (except as specified in 3.1.A.2.c.(2).(a) and 3.1.A.2.c.(2).(b) below) with the Over Pressure Protection System enabled, ~~the associated block valve open, and the associated backup air supply charged~~. M3.4-51 M3.4-52 LR3.4-53

LCO3.4.12 Cond A If both SI pumps are capable of injecting into the RCS, prompt action shall be taken to make one incapable of injecting into the RCS. M3.4-51

LCO3.4.12 Cond B,C If an ECCS accumulator is not isolated as required, isolate the affected accumulator within 1 hour, or increase the RCS cold leg temperature > LTOP enable temperature specified in the PTLR or depressurize the accumulator to less than the maximum RCS pressure for the existing cold leg temperature allowed in the PTLR. M3.4-52

LCO3.4.12 NOTE 2 **Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves in the PTLR. M3.4-52

SR3.4.12.1 SR3.4.12.2 SR3.4.12.3 New SRs, Verify no more than one SI pump capable of injecting into RCS, verify accumulator isolation valve isolated, verify PORV block valve open. M3.4-52 M3.4-54

~~3.1.A.2.c.(2)~~.

LCO3.4.12 Cond D

(a) One PORV may be inoperable for 7 days.

L3.4-50

LCO3.4.12 Cond E

If these conditions cannot be met, be in MODE 5 within 8 hours and depressurize and vent the reactor coolant system through at least a 3 square inch vent within ~~the next~~ 128 hours.

L3.4-50

LCO3.4.12 Cond E

(b) With both PORVs inoperable, be in MODE 5 within 8 hours and complete depressurization and venting of the RCS through at least a 3 square inch vent within 128 hours.

LCO3.4.13

(3) Reactor Coolant System average temperature below the temperature specified in the PTLR for disabling both safety injection pumps

LCO3.4.13

With Reactor Coolant System temperature less than the temperature specified in the PTLR for disabling both safety injection pumps, when the head is on the reactor vessel and the reactor coolant system is not vented through a 3 square inch or larger vent; with both SI pumps incapable of injecting into the RCS and the ECCS accumulators isolated, both Pressurizer power operated relief valves (PORVs) shall be OPERABLE (except as specified in 3.1.A.2.c.(3).(a) and 3.1.A.2.c.(3).(b) below) with the Over Pressure Protection System enabled, ~~the associated block valve open, and the associated backup air supply charged~~.

M3.4-51

M3.4-52

LR3.4-53

LCO3.4.13 Cond D

(a) One PORV may be inoperable for 24 hours.

LCO3.4.13 Cond E

If these conditions cannot be met, depressurize and vent the reactor coolant system through at least a 3 square inch vent within 8 hours.

LCO3.4.13 Cond E

(b) With both PORVs inoperable, complete depressurization and venting of the RCS through at least a 3 square inch vent within 8 hours.

LCO3.4.13 Cond A

If one or both SI pumps are capable of injecting into the RCS, prompt action shall be taken to make both incapable of injecting into the RCS.

M3.4-51

LCO3.4.13 Cond B

If an ECCS accumulator is not isolated as required, isolate the affected accumulator within 1 hour, or increase the RCS cold leg temperature > LTOP enable temperature specified in the PTLR or depressurize the accumulator to less than the maximum RCS pressure for the existing cold leg temperature allowed in the PTLR.

M3.4-52

SR3.4.13.1
SR3.4.13.2
SR3.4.13.3
SR3.4.13.4

New SRs: Verify no SI capable of injecting into RCS, verify accumulator isolation valve isolated, verify RCS vent > 3 square inches, verify PORV block valve open.

M3.4-52
M3.4-54

~~TS 3.1-5 Overflow~~

~~3.1.A.3 Reactor Coolant Vent System~~

~~a. A reactor shall not be made or maintained critical nor shall reactor coolant system average temperature exceed 200°F unless Reactor Coolant Vent System paths from both the reactor vessel head and pressurizer steam space are OPERABLE and closed (except as specified in 3.1.A.3.b and 3.1.A.3.c below).~~ R3.4-56

~~b. During STARTUP OPERATION and POWER OPERATION, any one of the following conditions of inoperability may exist for each unit provided STARTUP OPERATION is discontinued until OPERABILITY is restored. If any one of these conditions is not restored to an OPERABLE status within 30 days, be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours:~~

~~(1) Both of the parallel vent valves in the reactor vessel head vent path inoperable, or~~

~~(2) Both of the parallel vent valves in the pressurizer vent path inoperable, or~~

~~(3) The vent valve to the pressurizer relief tank discharge line inoperable, or~~

~~(4) The vent valve to the containment atmospheric discharge line inoperable.~~

~~c. With no Reactor Coolant Vent System path OPERABLE, restore at least one vent path to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the following 30 hours.~~

LCO3.4.2
SR3.4.2.1

New Specification 3.4.2, RCS Minimum Temperature for Criticality

M3.4-57

3.1.B. Pressure/Temperature Limits

1. Reactor Coolant System

LCO3.4.3

a. The Unit 1 and Unit 2 Reactor Coolant Systems (except the pressurizer) temperature, pressure heatup rates, and cooldown rates shall be maintained within the limits specified in the Pressure and Temperature Limits report (PTLR).

LCO3.4.3

LCO3.4.3
Cond A

b. If these conditions cannot be satisfied in Modes 1, 2, 3, and 4, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; within 72 hours determine that the Reactor Coolant System remains acceptable for continued operation or be in at least MODE 3~~HOT SHUTDOWN~~ within the next 6 hours and MODE 5~~reduce the reactor coolant system average temperature and pressure to less than 200°F~~ and 500 psig, respectively, within 36~~the following 30~~ hours.

A3.4-61

M3.4-62

A3.4-08

A3.4-39

LCO3.4.3
Cond C

If these conditions cannot be satisfied in other than Mode 1, 2, 3, or 4, promptly initiate action to restore parameters to within limits and determine if the RCS is acceptable for continued operation prior to entering MODE 4.

M3.4-63

SR3.4.3.1

New SR, Verify pressure, temperature, heatup, cooldown rates within PTLR limits

M3.4-64

~~2. Pressurizer~~

~~a. The pressurizer temperature shall be limited to:~~

~~1. A maximum heatup of 100°F in any 1-hour period.~~

~~2. A maximum cooldown of 200°F in any 1-hour period.~~

~~b. The pressurizer spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.~~

~~c. If these conditions cannot be satisfied, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT SHUTDOWN within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.~~

R3.4-66

~~3.1.B.3. Steam Generators~~

~~a. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.~~ R3.4-67

~~b. If these conditions cannot be satisfied, reduce the steam generator pressure to less than or equal to 200 psig within 30 minutes; perform an engineering evaluation to determine the effects of the overpressurization on the structural integrity of the steam generator; determine that the steam generator remains acceptable for continued operation prior to increasing its temperature above 200°F.~~

3.1.C. REACTOR COOLANT SYSTEM LEAKAGE

1. Leakage Detection

A3.4-71

LCO 3.0.4 does not apply.

A3.4-08

LCO3.4.16

In Modes 1, 2, 3 and 4~~The reactor coolant system average temperature shall not exceed 200°F unless~~ at least two means of reactor coolant system leakage detection shall be OPERABLE, one of which must depend on the detection of radionuclides in the containment. The other means is the containment sump A pump run time instrumentation. One means of leakage detection may be inoperable for 30 days provided daily RCS water balance inventories are performed and, if radionuclide detection is inoperable, daily compensatory sampling shall be performed. If these conditions cannot be met, be in MODE 3 in 6 hours and MODE 5 in 36 hours.

LCO3.4.16 Cond A Cond B Cond C

L3.4-68

LCO3.4.16 Cond D

If these conditions cannot be satisfied (all required monitors inoperable), immediately enter LCO 3.0.3 ~~within one hour initiate the action necessary to place the affected unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours~~.

A3.4-71

SR3.4.16.1 SR3.4.16.2 SR3.4.16.3 SR3.4.16.4

New SRs, Channel Check, COT and Calibration of containment radionuclide instrumentation and Calibration of sump pump run time instrumentation.

M3.4-72

2. Leakage Limitations

LCO3.4.14

The following leakage limitations are applicable whenever in Modes 1, 2, 3 and 4~~the reactor coolant system average temperature exceeds 200°F~~.

A3.4-08

A3.4-73

LCO3.4.14 Cond A

LCO3.4.14 Cond B

a. If the LEAKAGE~~leakage rate, from other than controlled leakage sources, such as the reactor coolant pump controlled leakage seals,~~ exceeds 1 gpm and the source of the leakage is not identified within 4 hours of leak detection, be in at least MODE 3~~HOT SHUTDOWN~~ within the next 6 hours. If the source of leakage is not identified within 54~~an additional 48~~ hours, be in MODE 5~~COLD SHUTDOWN~~ within 84~~the following 30~~ hours.

A3.4-08

A3.4-39

LCO3.4.14

b. If the sources of leakage are identified ~~and the results of the evaluations are that continued operation is safe~~, operation of the reactor with a total leakage, other than leakage from controlled sources, not exceeding 10 gpm shall be permitted (except as specified in 3.1.C.2.c below).

LR3.4-74

LCO3.4.14 Cond D

c. If it is determined that leakage exists through a fault which has developed in a Reactor Coolant System component body, pipe wall, vessel wall, or pipe weld, and that the fault cannot be isolated,~~within one hour initiate action to place the unit in HOT SHUTDOWN and~~ be in at least MODE 3 ~~HOT SHUTDOWN~~ within the next 6 hours and be in MODE 5 ~~COLD SHUTDOWN~~ within 36~~the next 30~~ hours and take corrective action prior to resumption of unit operation.

M3.4-42

A3.4-08

A3.4-39

LCO3.4.14
Cond C

d. If the total LEAKAGE~~leakage, other than leakage from controlled sources,~~ exceeds 10 gpm, ~~within one hour initiate action to place the unit in HOT SHUTDOWN and~~ be in at least MODE 3 ~~HOT SHUTDOWN~~ within the next 6 hours. If the condition is not corrected within 14~~an additional 8~~ hours, be in MODE 5~~COLD SHUTDOWN~~ within 44~~the following 30~~ hours ~~and remain in COLD SHUTDOWN until the condition is corrected~~.

A3.4-73

M3.4-42

A3.4-08

A3.4-39

A3.4-77

LCO3.4.14 Cond D

e. If the total reactor coolant system to secondary coolant system leakage through any one steam generator of a unit exceeds 150 gallons per day (GPD), ~~within one hour initiate action to place the unit in HOT SHUTDOWN and~~ be in at least MODE 3 ~~HOT SHUTDOWN~~ within the next 6 hours and be in MODE 5 ~~COLD SHUTDOWN~~ M3.4-42 A3.4-08

within 36 ~~the following 30~~ hours A3.4-39

SR3.4.14.2

and perform an inservice steam generator tube inspection in accordance with Steam Generator Program ~~Technical Specification 4.12~~. A3.4-78

3. <u>Pressure Isolation Valve Leakage</u>

LCO3.4.15

Leakage through the pressure isolation valves shall not exceed the maximum allowable leakage specified in Specification 4.3 when in MODES 1, 2, 3, and 4 ~~reactor coolant system average temperature exceeds 200°F~~. A3.4-08

LCO3.4.15 Cond A

LCO3.4.15 Cond B

If the maximum allowable leakage is exceeded, isolate the high pressure portion of the system in 4 hours and restore the PIV to within limits in 72 hours. Valves used to isolate the PIV flow path must meet SR 3.4.15.1 leakage limits. ~~within one hour initiate the action necessary to place the unit in HOT SHUTDOWN, and~~ If these requirements can not be met, be in at least MODE 3 ~~HOT SHUTDOWN~~ within the next 6 hours and in MODE 5 ~~COLD SHUTDOWN~~ L3.4-79 A3.4-08

within 36 ~~the following 30~~ hours. A3.4-39

~~3.1.D.~~ RCS Specific ~~MAXIMUM~~ ~~COOLANT~~ ACTIVITY

LCO3.4.17

1. The specific activity of the primary coolant (except as specified in 3.1.D.2 and 3 below) shall be limited to:

   a. Less than or equal to 1.0 microcuries per gram DOSE EQUIVALENT I-131, and

   b. Less than or equal to 100/Ē microcuries per gram of gross radioactivity.

2. If a reactor is in Modes 1 and 2 or ~~critical or the~~ in Modes 3 with reactor coolant system average temperature ~~is~~ greater than or equal to 500°F: A3.4-08

   LCO3.4.17 Cond A

   a. With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131, verify once per 4 hours to be within the acceptable region of Figure TS.3.1-3, for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure TS.3.1-3, M3.4-81

   LCO3.4.17 Cond C

   the reactor shall be in MODE 3 with ~~shutdown and~~ reactor coolant system average temperature cooled to below 500°F within 6 hours. A3.4-08

   LCO3.4.17 Cond B

   b. With the gross specific activity of the primary coolant greater than 100/Ē microcurie per gram, the reactor shall be in MODE 3 with ~~shutdown and~~ reactor coolant system average temperature cooled to below 500°F within 6 hours. A3.4-08

~~3. If a reactor is at or above COLD SHUTDOWN, with the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 or greater than 100/Ē microcuries per gram, perform the sampling and analysis requirements of item 4a of Table 4.1-2B until the specific activity of the primary coolant is restored to within its limits.~~ L3.4-82

~~Next pages are Figure TS.3.1-3 and TS.3.1-12.~~

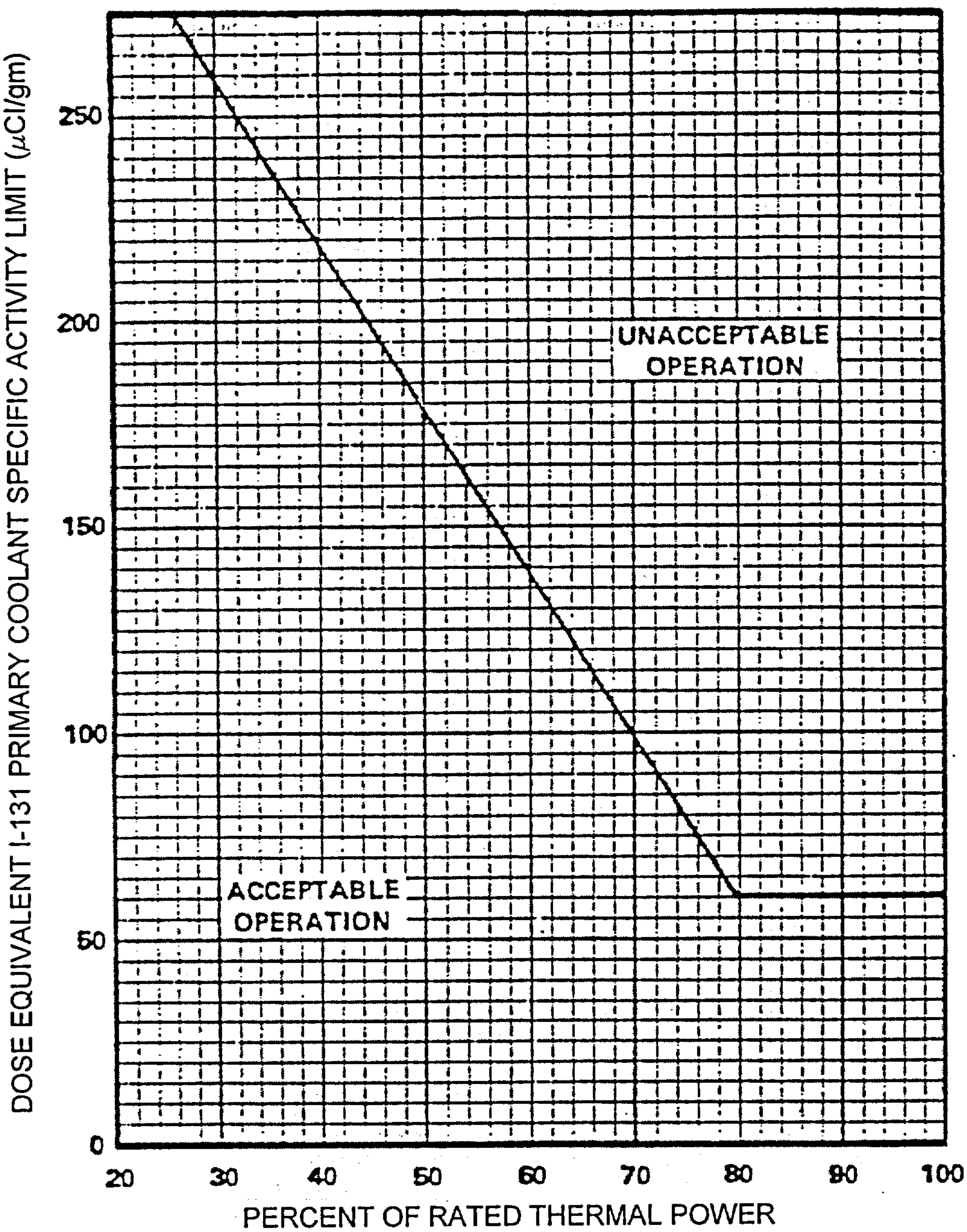

Figure ~~TS.3.1-3~~ 3.4.17-1 (Page 1 of 1)

Reactor Coolant DOSE EQUIVALENT I-131 ~~Primary Coolant~~ Specific Activity Limit Versus Percent of RATED THERMAL POWER ~~with the Primary Coolant Specific Activity > 1.0 μCi/gram Dose Equivalent I-131~~

Addressed Elsewhere

3.3.A.2.g The valve position monitor lights or alarms for motor-operated valves specified in 3.3.A.1.g above may be inoperable for 72 hours provided the valve position is verified once each shift.

LCO3.4.12

3. A maximum of one safety injection pump shall be capable of injecting into the RCS whenever in MODE 4 with RCS temperature is less than the Over Pressure Protection System Enable Temperature specified in the PTLR A3.4-83

LCO3.4.12
LCO3.4.13
NOTE 1

except that both SI pumps may be run for up to one hour while conducting the integrated SI test** when either of the following conditions is met:

(a) There is a steam or gas bubble in the pressurizer and an isolation valve between the SI pump and the RCS is shut, or

(b) The reactor vessel head is removed.

LCO3.4.13

4. No safety injection pumps*** shall be capable of injecting into the RCS whenever in MODE 4 with RCS temperature is less than the temperature specified in the PTLR for disabling both safety injection pumps (except one or both pumps may be run as specified in 3.3.A.3 and 3.1.A.1.d.(2)). A3.4-83

LCO3.4.12
LCO3.4.13

5. Both reactor coolant system accumulators shall be isolated* whenever RCS temperature is less than the Over Pressure Protection System Enable Temperature specified in the PTLR.

LCO3.4.12
NOTE 2
LCO3.4.13
NOTE 3

*This specification does not apply whenever the reactor coolant system accumulators are depressurized or the reactor vessel head is removed.

LCO3.4.12
LCO3.4.13
NOTE 1

**Other SI system tests and operations may also be conducted under these conditions.

LCO3.4.13

***This specification does not apply whenever the reactor vessel head is removed.

Addressed Elsewhere

TABLE TS 4.1-1C (Page 3 of 4)

MISCELLANEOUS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHECK	CALIBRATE	FUNCTIONAL TEST	RESPONSE TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
23. Deleted					
24. Steam Exclusion Actuation	W	Y	M	N.A.	1, 2, 3

SR3.4.12.4
SR3.4.12.5
SR3.4.13.5
SR3.4.13.6

M3.4-84

FUNCTIONAL UNIT	CHECK	CALIBRATE	FUNCTIONAL TEST	RESPONSE TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
25. Overpressure Mitigation	N.A.	R	M ~~R~~	N.A.	4(38), 5

Addressed Elsewhere

FUNCTIONAL UNIT	CHECK	CALIBRATE	FUNCTIONAL TEST	RESPONSE TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
26. Auxiliary Feedwater Pump Suction Pressure	N.A.	R	R	N.A.	1, 2, 3
27. Auxiliary Feedwater Pump Discharge Pressure	N.A.	R	R	N.A.	1, 2, 3
28. NaOH Caustic Stand Pipe Level	W	R	M	N.A.	1, 2, 3, 4
29. Hydrogen Monitors	S	Q	M	N.A.	1, 2
30. Containment Temperature Monitors	M	R	N.A.	N.A.	1, 2, 3, 4
31. Turbine Overspeed Protection Trip Channel	N.A.	R	M	N.A.	1

~~Table TS 4.1-1C (Page 3 of 4) REV 121 11/9/95~~

TABLE TS.4.1-1C (Page 4 of 4)

TABLE NOTATIONS

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	Shift
D	Daily
W	Weekly
M	Monthly
Q	Quarterly
S/U	Prior to each startup
Y	Yearly
R	Each refueling shutdown
N.A.	Not applicable

TABLE NOTATION

Addressed Elsewhere

(30) Prior to each startup following shutdown in excess of two days if not done in previous 30 days.

(31) When the reactor trip system breakers are closed and the control rod drive system is capable of rod withdrawal.

(32) Following rod motion in excess of six inches when the computer is out of service.

(33) Transfer logic to Refueling Water Storage Tank.

(34) When either main steam isolation valve is open.

(35) Includes those instruments named in the emergency procedure.

(36) Except for containment hydrogen monitors and refueling water storage tank level which are separately specified in this table.

(37) When RHR is in operation.

(38) Within 12 hours of When the reactor coolant system average temperature is less than the

SR3.4.12.4
SR3.4.13.5

M3.4-85

Over Pressure Protection System Enable Temperature specified in the PTLR.

(39) Whenever CONTAINMENT INTEGRITY is required.

~~Table TS.4.1-1C~~
~~(Page 4 of 4)~~
~~REV 135 5/4/98~~

~~Table TS.4.1-2A (Page 1 of 2)~~

Addressed Elsewhere

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

	Equipment	Test	Frequency	FSAR Sect. Reference
	1. Control Rod Assemblies	Rod Drop Times of full length rods	All rods during each refueling shutdown or following each removal of the reactor vessel head; affected rods following maintenance on or modification to the control rod drive system which could affect performance of those specified rods	7
	2. Control Rod Assemblies	Partial movement of all rods	Every Quarter	7
SR3.4.10.1	3. Pressurizer Safety Valves	Verify OPERABLE in accordance with the Inservice Testing Program (± 3%). Following testing, lift settings shall be within ±1%	Per ASME Code, Section XI Inservice Testing Program	-
Addressed Elsewhere	4. Main Steam Safety Valves	Verify each required lift setpoint in accordance with the Inservice Testing Program (± 3%). Following testing, lift settings shall be within ±1%	Per ASME Code, Section XI Inservice Testing Program	-
	5. Reactor Cavity	Water Level	Prior to moving fuel assemblies or control rods and at least once every day while the cavity is flooded.	
SR3.4.11.1	6. Pressurizer PORV Block Valves	Functional	Quarterly, unless the block valve has been closed per Specification 3.1.A.2.c.(1).(b).2 or 3.1.A.2.c.(1).(b).3.	
SR3.4.11.2	7. Pressurizer PORVs	Functional	Every 24 ~~18~~ months	

L3.4-86

~~Table TS.4.1-2A (Page 1 of 2) REV 123 5/21/96~~

~~Table TS.4.1-2A (Page 2 of 2)~~

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS

	Equipment	Test	Frequency	FSAR Sect. Reference
	8. Deleted			L3.4-89
SR3.4.14.1	9. Primary System Leakage	Evaluate	72 hours ~~Daily~~	4
	10. Deleted			
	11. Turbine stop valves, governor valves, and intercept valves. (Part of turbine overspeed protection)	Functional	Turbine stop valves, governor valves and intercept valves are to be tested at a frequency consistent with the methodology presented in WCAP-11525 "Probabilistic Evaluation of Reduction in Turbine Valve test Frequency", and in accordance with the established NRC acceptance criteria for the probability of a turbine missle ejection incident of 1.0×10^{-5} per year. In no case shall the turbine valve test interval exceed one year.	10

Addressed Elsewhere

~~Table TS.4.1-2A~~
~~(Page 2 of 2)~~
~~REV 123 5/21/96~~

TABLE TS.4.1-2B

MINIMUM FREQUENCIES FOR SAMPLING TESTS

		TEST	FREQUENCY	
SR3.4.17.1	1.	RCS Gross Activity Determination	1 ~~5~~/week	L3.4-88
SR3.4.17.2	2.	RCS Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1/14 days (when at power)	
SR3.4.17.3	3.	RCS Radiochemistry E determination	1/6 months(1) (when at power)	
LCO3.4.17	4.	RCS Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1.0 uCi/gram DOSE EQUIVALENT I-131 ~~or 100/E uCi/gram~~ (at or above 500 °F ~~cold shutdown~~), and	L3.4-91 L3.4-82
SR3.4.17.2			b) One sample between 2 and 6 hours following THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period (above hot shutdown)	
	~~5.~~	~~RCS Radiochemistry (2)~~	~~Monthly~~	LR3.4-94
	~~6.~~	~~RCS Tritium Activity~~	~~Weekly~~	LR3.4-94
	~~7.~~	~~Deleted~~		
	~~8.~~	~~RCS Boron Concentration* (3)~~	~~2/Week (4)~~	LR3.4-96
	9.	RWST Boron Concentration	Weekly	Addressed Elsewhere
	10.	Boric Acid Tanks Boron Concentration	2/Week	
	11.	Caustic Standpipe NaOH Concentration	Monthly	
	12.	Accumulator Boron Concentration	Monthly	
	13.	Spent Fuel Pit Boron Concentration	Weekly(7)(8)	

~~* Required at all times.~~

~~TABLE TS.4.1-2B~~

Addressed Elsewhere

MINIMUM FREQUENCIES FOR SAMPLING TESTS

TEST	FREQUENCY
14. Secondary Coolant Gross Beta-Gamma activity	Weekly
15. Secondary Coolant Isotopic Analysis for DOSE EQUIVALENT I-131 concentration	1/6 months (5)
16. Secondary Coolant Chemistry	
pH	5/week (6)
pH Control Additive	5/week (6)
Sodium	5/week (6)

Notes:

SR3.4.17.3 Note

1. Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

~~2. To determine activity of corrosion products having a half-life greater than 30 minutes.~~

LR3.4-94

3. During REFUELING, the boron concentration shall be verified by chemical analysis daily.

~~4. The maximum interval between analyses shall not exceed 5 days.~~ LR3.4-96

5. If activity of the samples is greater than 10% of the limit in Specification 3.4.D, the frequency shall be once per month.

6. The maximum interval between analyses shall not exceed 3 days.

Addressed Elsewhere

~~4.3~~PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

~~Applicability~~

~~Applies to the surveillance performed on the primary coolant system pressure isolation valves to verify operability.~~ A3.4-03

~~Objective~~

~~To increase the reliability of primary coolant system pressure isolation valves thereby reducing the potential of an intersystem loss of coolant accident.~~

Specification

SR3.4.15.1 Periodic leakage testing of each of the following valves shall be individually accomplished every 24 months, prior to resuming power operation after ~~each time the plant is placed in the cold shutdown condition for refueling~~, each time the plant is placed in MODE 5 ~~a cold shutdown condition~~ for 7 days or more if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair, or replacement work is performed: A3.4-99 A3.4-08

~~System~~	~~Valve Number~~ ~~Unit No. 1~~	~~Unit No. 2~~	~~Maximum Allowable Leakage (*)(**)~~
~~Low-Pressure SI to Upper Plenum~~	~~SI-9-6~~	~~2SI-9-6~~	~~≤ 5 gpm~~
	~~SI-9-4~~	~~2SI-9-4~~	~~≤ 5 gpm~~
	~~SI-9-5~~	~~2SI-9-5~~	~~≤ 5 gpm~~
	~~SI-9-3~~	~~2SI-9-3~~	~~≤ 5 gpm~~
~~RHR to Loop B Accumulator Inj Line~~	~~SI-6-2~~	~~2SI-6-2~~	~~≤ 5 gpm~~

LR3.4-97

~~To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.~~

NOTES:

* 1. Leakage rates less than or equal to 0.5 ~~one~~ gpm per nominal inch of valve size are acceptable at an RCS pressure of ≥ 2215 psig and ≤ 2255 psig.

SR3.4.15.1 L3.4-87

2. ~~Leakage rates greater than one, but less than or equal to five gpm are considered acceptable if the latest measured rate has not exceeded the previous measured rate by an amount which reduces the margin to five gpm by 50% or more. Otherwise the leakage rate is considered unacceptable.~~

~~3.~~ Leakage rates greater than five gpm are considered unacceptable.

~~** Minimum differential test pressure shall not be less than 150 psid.~~ LR3.4-97

4.6. Addressed Elsewhere

B. Station Batteries

1. Each battery shall be tested each month. Tests shall include measuring voltage of each cell to the nearest hundredth volt, and measuring the temperature and density of a pilot cell in each battery.

2. The following additional measurements shall be made every three months: the density and height of electrolyte in every cell, the amount of water added to each cell, and the temperature of each fifth cell.

3. All measurements shall be recorded and compared with previous data to detect signs of deterioration or need of equalization charge according to the manufacturer's recommendation.

4. The batteries shall be subjected to a performance test discharge during the first refueling and once every five years thereafter. Battery voltage shall be monitored as a function of time to establish that the battery performs as expected during heavy discharge and that all electrical connections are tight.

5. Integrity of Station Battery fuses shall be checked once each day when the battery charger is running.

C. Pressurizer Heater Emergency Power Supply

SR3.4.9.2 The emergency pressurizer heater supply shall be demonstrated OPERABLE at

least once every 24~~18~~ months L3.4-86

~~by transferring Backup Heater Group "B" from its normal bus to its safeguards bus and energizing the heaters~~. LR3.4-98

R3.4-56

~~4.18 REACTOR COOLANT VENT SYSTEM PATHS~~

~~Applicability~~

~~Applies to the surveillance performed on the Reactor Coolant Vent System paths to verify OPERABILITY.~~

~~Objective~~

~~To assure that the capability exists to vent noncondensible gases from the Reactor Coolant System that could inhibit natural circulation core cooling.~~

~~Specification~~

~~A. Vent Path Operability~~

~~Each Reactor Coolant Vent System path shall be demonstrated OPERABLE prior to commencing STARTUP OPERATION after each refueling by:~~

~~1. Verifying all manual isolation valves in each vent path are blocked and tagged in the open position.~~

~~2. Cycling each solenoid operated valve in the vent paths through at least one complete cycle of full travel from the control room.~~

~~B. System Flow Testing~~

~~Flow shall be verified through each Reactor Coolant Vent System path following each refueling.~~

PACKAGE 3.4

REACTOR COOLANT SYSTEM (RCS)

PART D

DISCUSSION OF CHANGES
(DOC)

to

PRAIRIE ISLAND
CURRENT TECHNICAL SPECIFICATIONS

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
UNITS 1 AND 2

Improved Technical Specifications
Conversion Submittal

Part D

Package 3.4

REACTOR COOLANT SYSTEMS (RCS)

DISCUSSION OF CHANGES TO CURRENT TECHNICAL SPECIFICATIONS

The proposed changes to PI Operating License Appendix A, TS are discussed below and the specific wording changes are shown in parts B, C and E.

For ease of review, all package part and discussions are organized according to the proposed PI ITS Table of Contents.

NSHD Category	Change Number 3.4-	Discussion of Change
LR	01	CTS 3.10.J. The specific limits for RCS T_{ave} and pressurizer pressure have been relocated to the COLR. This is acceptable since these limits will continue to be met during plant operation. These limits may be changed by the licensee when they are in the COLR; therefore, this is a less restrictive change. However these limits can only be changed in accordance with approved methodology; therefore this change is acceptable. This change is consistent with NUREG-1431 as modified by approved TSTF-339.

NSHD Category	Change Number 3.4-	Discussion of Change
L	02	CTS 3.10.J. CTS includes a note which does not apply pressurizer pressure limits during increasing power ramps or step changes. The note limitation "increase" has been removed so the note applies to ramps and steps which increase or decrease. This is acceptable since the LCO requirements are required to prevent exceeding DNB limits and a ramp or step decrease in power will increase DNBR and maintain the plant in a safe condition. This change is consistent with the guidance of NUREG-1431.
A	03	CTS 3.1 and 4.3. The beginning of each CTS section contains general statements of Applicability and Objectives for that TS section. This Applicability states the systems to which the specifications apply which is a different meaning than the Applicability in NUREG-1431. Since the ITS clearly states within each specification the system to which it applies, administratively these statements have been incorporated. Likewise, the CTS Objectives statement provides an overall purpose for the specifications within the section. These objectives are administratively incorporated in general through the statement of the ITS specification LCO and the supporting Bases. Since these general CTS statements do not establish any regulatory requirements and are incorporated in a broad sense in the ITS, these are considered administrative changes.

NSHD Category	Change Number 3.4-	Discussion of Change
M	04	CTS 3.1.A.1.a (1). The prose description of the conditions of applicability for this specification have been replaced by the applicable NUREG-1431 MODE. These requirements have been changed in conformance with ISTS, since the mode change closest to critical is MODE 2 which begins at K_{eff} > 0.99. This change is acceptable because the same plant operating restrictions will apply at a somewhat sooner time when the reactor K_{eff} > 0.99. This change is included to make the ITS complete and does not place the plant in any unsafe operating condition. Since this change imposes plant equipment requirements earlier in the plant startup evolution, it is a more restrictive change.
L	05	CTS 3.10.J. This change will allow the plant 6 hours to reduce to MODE 2 when the DNB limits are not met. CTS allows 4 hours to reduce to MODE 2 in this condition. The additional 2 hours (for a total time of 6 hours versus 4 hours) to achieve Mode 2 was originally applied during the development of NUREG-1431 as a reasonable amount of time to reduce power without unnecessarily challenging the operators or safety systems. The additional time should reduce the risk of incurring an undesirable transient by providing for a more orderly shutdown. It is also consistent with LCO 3.0.3 which allows 6 hours to achieve Mode 3 in the event of a complete loss of safety function. This change was determined to be generically acceptable to the NRC during the development of NUREG-1431, and it is applicable to PI.

NSHD Category	Change Number 3.4-	Discussion of Change
M	06	New Action Statement. In conformance with ISTS, a restriction on RCS loop inoperability for low power PHYSICS TESTS has been added. Power level is restricted to less than P-7 and an associated action statement has also been included. This is more restrictive since the level for low power testing is currently undefined. This more restrictive change is included to make the PI ITS complete.
M	07	New SRs, 3.4.18.1 and 3.4.18.2, have been included in conformance with ISTS low power PHYSICS TESTS. These SRs will require verification that the power is less than P-7 during PHYSICS TESTS and require performance of a COT on P-7, power range (low setpoint) and intermediate range neutron flux channels. These are more restrictive since PI CTS do not require these surveillances. These more restrictive changes are included to make the PI ITS complete. These changes do not introduce any unsafe plant operating or testing conditions.

NSHD Category	Change Number 3.4-	Discussion of Change
A	08	CTS 3.1.A.1.a(2), 3.1.A.1.b(1), 3.1.A.1.c, 3.1.A.1.d, 3.1.A.2.a(1), 3.1.A.2.a(2), 3.1.A.2.a(3), 3.1.A.2.b(1), 3.1.A.2.c(1), 3.1.A.2.c(1)(a), 3.1.A.2.c(1)(b), 3.1.A.2.c(1)(b)3, 3.1.B.1.b, 3.1.C.1, 3.1.C.2, 3.1.C.3, 3.1.D.2, 3.1.D.2.a, 3.1.D.2.b, 3.10.J and 4.3. The CTS contain prose descriptions of the modes of applicability and conditions in which the plant must be placed or remain when equipment is inoperable. These descriptions have been replaced with the equivalent MODES of applicability for the ITS. Since the modes of applicability and plant conditions in which the plant is placed or remains have not changed this is an administrative change.
	9	Not used.
	10	Not used.
M	11	A new SR, 3.4.4.1, is included in conformance with ISTS which requires verification of each RCS loop in operation while at power. This is more restrictive since PI CTS do not require this surveillance. This more restrictive change is included to make the PI ITS complete.

NSHD Category	Change Number 3.4-	Discussion of Change
M	12	CTS 3.1.A.1.b. The prose description of the conditions of applicability for this specification have been replaced by the applicable NUREG-1431 MODE. The CTS applies this specification "above 350°F" whereas the condition of being equal to 350°F is included in MODE 3; thus this is a change to make the CTS requirements consistent with NUREG-1431 MODE definitions. This change is acceptable because the same plant operating restrictions will apply at a somewhat sooner time when the RCS temperature equals 350°F. Since these new actions place additional requirements on plant operations this change is more restrictive. This change is included to make the ITS complete and does not place the plant in any unsafe operating condition.
M	13	CTS 3.1.A.1.b(1). New requirements have been added to require both RCS loops to operate if the Rod Control System is capable of rod withdrawal or only one loop is required to operate if the Rod Control System is not capable of rod withdrawal. These new requirements are necessary to make plant operations consistent with the supporting plant analyses. These changes also make the PI ITS consistent with NUREG-1431.

NSHD Category	Change Number 3.4-	Discussion of Change
A	14	CTS 3.1.A.1.b(2), 3.1.A.2.a(2), 3.1.A.2.b(1) and 3.1.A.2.c(1)(b). The CTS condition that "STARTUP OPERATION is discontinued until OPERABILITY is restored" or similar statements in the action statements have not been included in the ITS. The ITS, in conformance with the guidance of NUREG-1431, included general rules for use and applicability governing how the plant may operate. Among these rules is Specification 3.0.4 which, unless specifically excepted, does not allow the plant to startup with inoperable TS required equipment. Thus, it is not necessary to include this CTS restriction in each specification. Since the ITS requirements provide the same plant operating restrictions, this is an administrative change.
	15	Not used.
L	16	CTS 3.1.A.1.b(2). The required time to reduce RCS temperature below 350°F is increased from 6 hours to 12 hours. This completion time is compatible with the operations required to achieve cooldown and depressurization under the existing plant conditions in an orderly manner without challenging plant systems. This change is consistent with the guidance of NUREG-1431.

NSHD Category	Change Number 3.4-	Discussion of Change
M	17	CTS 3.1.A.1.b(3). Action statements have been included which address one RCS loop not in operation with the reactor trip breakers closed and the Rod Control System capable of rod withdrawal. This new action statement is necessary to support the new LCO restrictions placed on the plant discussed in Change M3.4-06 above. Since these new actions place additional requirements on plant operations, this is more restrictive. This more restrictive change is included to make the PI ITS complete and consistent with NUREG-1431.
A	18	CTS 3.1.A.1.b(3). CTS specifies the actions to be taken if the Action Statements are not met. This is not included in the ITS since these requirements are defined by the format of ITS and do not need to be stated explicitly. Since there are no substantive changes made, this is an administrative change.
	19	Not used.
	20	Not used.

NSHD Category	Change Number 3.4-	Discussion of Change
M	21	Three new SRs, 3.4.5.1, 3.4.5.2 and 3.4.5.3, have been added in conformance with the guidance of NUREG-1431. These SRs will require verification that the required RCS loop is in operation, verify the SG level is within the required limit and verify power is available to the RCP that is not operating. This is more restrictive since PI CTS do not require these surveillances. These more restrictive changes are included to make the PI ITS complete and do not introduce any unsafe plant conditions.
A	22	CTS 3.1.A.1.b(3). CTS requires the plant to shut down to MODE 4 if the Action Statement requirements are not met. These requirements are not included in the ITS, since once the Action Statement requirements are not met, the plant will enter LCO 3.0.3 which essentially requires the same actions as CTS.

NSHD Category	Change Number 3.4-	Discussion of Change
L	23	CTS 3.1.A.1.b(2). CTS requires the control rod drive mechanisms to be de-energized when both reactor coolant pumps are inoperable or not in operation. This requirement has not been included in the ITS. The PI CTS provision to shutdown both reactor coolant pumps would only allow reactor coolant pump coastdown tests to be performed. By removing the requirement to de-energize the control rod drive mechanisms, control rod drop tests can also be performed. This is less restrictive in that it allows additional plant operations. This change is acceptable because the plant is only allowed to be in these conditions for a short time and other restrictions continue to be placed on plant operations. This change is consistent with NUREG-1431.
LR	24	CTS 3.1.A.1.c(1), 3.1.A.1.c(2), 3.1.A.1.d(1) and 3.1.A.2.a(1). The CTS description of equipment required for system operability has been relocated to the Bases which is consistent with the format and guidance of NUREG-1431. This change is acceptable since the system is required to be operable in accordance with the definition of OPERABILITY and details of the specific equipment are unnecessary in the specification. Since the Bases are under licensee control, this is a less restrictive change.
	25	Not used.

NSHD Category	Change Number 3.4-	Discussion of Change
M	26	CTS 3.1.A.1.c. CTS do not define a minimum SG level for OPERABILITY in this MODE of operation. To make the PI ITS complete and for consistency with NUREG-1431, a specific level requirement is included. This is more restrictive since a new requirement is introduced.
	27	Not used.
A	28	CTS 3.1.A.1.c. CTS has a single specification which requires two methods of cooling the RCS to be operable with one in operation when in MODE 4 and MODE 5, loops filled. This can be any combination of RCS loops or RHR trains. During plant heatup from MODE 5 to MODE 4, one RCS loop may be in operation with one or both RHR loops operable but not operating. NUREG-1431 splits MODE 4 and MODE 5, loops filled into two specifications, LCO 3.4.6 and LCO 3.4.7. Because these MODES or other specified conditions of applicability have been split into two Specifications, LCO Note 4 in 3.4.7 is required to allow one RCS loop in operation with one or both RHR loops operable but not operating. This configuration is required to allow a smooth transition to MODE 4. Since the configuration allowed by Note 4 currently is allowed by CTS, explicit statement of this Note is an administrative change. This change is consistent with the ISTS.
	29	Not used.

NSHD Category	Change Number 3.4-	Discussion of Change
	30	Not used.
M	31	Three new SRs, 3.4.6.1, 3.4.6.2, 3.4.6.3, have been added in conformance with ISTS. These SRs will require verification that the required RHR or RCS loop is in operation, verify the SG level is within the required limit and verify power is available to the required RHR or RCP that is not operating. This is more restrictive since PI CTS do not require these surveillances. These more restrictive changes are included to make the PI ITS complete and do not introduce any unsafe plant operations.
M	32	Three new SRs, 3.4.7.1, 3.4.7.2, 3.4.7.3, have been added in conformance with ISTS. These SRs will require verification that one RHR loop is in operation, verify the SG level is within the required limit and verify power is available to the RHR pump that is not operating. This is more restrictive since PI CTS do not require these surveillances. These more restrictive changes are included to make the PI ITS complete and they do not introduce any unsafe plant operations.

NSHD Category	Change Number 3.4-	Discussion of Change
M	33	CTS 3.1.A.1.d. An additional required action has been included in the action statement when both RHR loops are inoperable to suspend operations involving reduction in RCS boron concentration. This change is included to make the ITS complete and may improve plant safety. Since this change places additional restrictions on plant operations it is more restrictive. This change is also consistent with the guidance of NUREG-1431.
M	34	CTS 3.1.A.1.d. CTS allows all pumps to be shutdown for up to one hour. For ITS Specification 3.4.8, the Note has been modified to only allow all pumps to be shutdown for up to 15 minutes. Also, the Note is modified by addition of a restriction that, in MODE 5 with the loops not filled, no RCS draining operations are permitted when all pumps are shutdown. Restrictions on boron concentration changes have also been included. These changes were made for consistency with the guidance of NUREG-1431 as modified by approved TSTF-286, Revision 2. These changes are more restrictive since they impose requirements on plant operations that are not in the CTS. These changes are acceptable since they may improve plant safety.
	35	Not used.

NSHD Category	Change Number 3.4-	Discussion of Change
L	36	CTS 3.1.A.1.d. One RHR loop may be made inoperable for surveillance testing while the other RHR loop is OPERABLE and in operation. This change is made for consistency with NUREG-1431. This is acceptable since the operating RHR loop provides sufficient heat removal capability and the loss of the operating RHR loop in the two hours allowed for testing is unlikely. This is less restrictive since this provision is not contained in the CTS.
M	37	Two new SRs, 3.4.8.1 and 3.4.8.2, have been added in conformance with NUREG-1431. These SRs will require verification that one RHR loop is in operation and verify power is available to the RHR pump that is not operating. This is more restrictive since PI CTS do not require these surveillances. These more restrictive changes are included to make the PI ITS complete. These changes do not place the plant in any unsafe operating conditions.
M	38	CTS 3.1.A.1.d. CTS allows the pumps to be shutdown for 1 hour. To be consistent with NUREG-1431, this is further restricted to 1 hour in an 8 hour period. Since this imposes additional restrictions on plant operations, this is a more restrictive change. This change is acceptable because this change may limit the time that all pumps are not operating and improve plant safety.

NSHD Category	Change Number 3.4-	Discussion of Change
A	39	CTS 3.1.A.2.a, 3.1.A.2.c.(1), 3.1.B.1.b and 3.1.C.2. The format for CTS and ITS fundamentally differ in the presentation of shutdown tracks but the Completion Time requirements are the same. The CTS format has been changed to the ITS format. Since there is no net change in plant operations, this is an administrative change.
	40	Not used.
M	41	CTS 3.1.A.2.a(1). CTS for pressurizer OPERABILITY require a steam bubble in the pressurizer and group A and B heaters operable. The ITS will require pressurizer level less than or equal to the pressurizer high water level Allowable Value and two heater groups capable of being powered from an emergency power supply. Since the pressurizer high water level Allowable Value is significantly below the level required to assure that there is a steam bubble in the pressurizer, this is a more restrictive change. This change has been made to be consistent with NUREG-1431 and current NRC guidance. This change is acceptable since it will not introduce any unsafe plant operating or test conditions.

NSHD Category	Change Number 3.4-	Discussion of Change
M	42	CTS 3.1.A.2.a(3), 3.1.C.2.c, 3.1.C.2.d and 3.1.C.2.e. For consistency with NUREG-1431, the one hour to initiate actions necessary for shutdown has been deleted. This is more restrictive since the plant has one less hour prior to placing the plant in MODE 3. The time provided in the ITS is adequate to plan for and implement an orderly shutdown if required; thus this change is acceptable. This change does not place the plant in any unsafe operating conditions.
M	43	CTS 3.1.A.2.a(3). New Required Actions have been provided for the possibility that the pressurizer is inoperable. These actions incorporate proposed TSTF-87, Rev. 1. This is more restrictive since PI CTS do not require these actions. This more restrictive change is included to make the PI ITS complete.
M	44	A new SR, 3.4.9.1, has been added in conformance with the guidance of NUREG-1431. This SR will require periodic verification of a steam bubble in the SG. This is more restrictive since PI CTS do not require this surveillance. This more restrictive change is included to make the PI ITS complete.

NSHD Category	Change Number 3.4-	Discussion of Change
M	45	CTS 3.1.A.2.b(1). CTS 3.1.A.2.b(1) requires two RCS pressurizer safety valves (PSVs) to be operable when the RCS temperature is > 350°F. This change will require two RCS PSVs to be operable whenever both RCS cold leg temperatures are greater than the OPPS enable temperature specified in the PTLR. This change is consistent with the guidance of NUREG-1431 and will provide additional overpressure protection when the RCS is between 350°F and the OPPS enable temperature. Since this change will require more components to be operable for more plant conditions, this is a more restrictive change. This change will also require the plant to be cooled down further than CTS; thus the Completion Time is specified as 24 hours. This Completion Time change is consistent with the guidance of NUREG-1431 as modified by approved traveler TSTF-352, Revision 1. These changes are acceptable because they will not cause any unsafe plant operating or testing conditions.
A	46	CTS 3.1.A.2.b(1). PI CTS License Amendment 123 allows pressurizer safety valves to be 2485 psig ±3% when tested for operability. However, because of the particular wording in Specification 3.1.A.2.b(1), it was not revised. The intent of Amendment 123 was that if the operators were aware the setting was outside 2485 psig ±1% then they would not continue with their startup. CTS Table 4.1-2A, Item 3 requires verification that the settings are within ±3. Thus, the change in LCO 3.4.10 is not a technical change since CTS already allow ±3% tolerance on the valve settings and this is simply an administrative change.

NSHD Category	Change Number 3.4-	Discussion of Change
L	47	CTS 3.1.A.2.b(2). In conformance with NUREG-1431 guidance, this specification which requires one pressurizer safety valve operable when the head is on the reactor vessel has been not been retained. Since the pressurizer safety valves do not provide overpressurization protection when the RCS temperature is below the LTOP enable temperature, this specification serves no purpose. NUREG-1431 requires PSVs to be operable when the RCS temperature exceeds the LTOP enable temperature. Below the LTOP enable temperature, NUREG-1431 requires redundant PORVs to be OPERABLE at the LTOP lift settings or the RCS to be vented. Thus, this specification has not been retained. This change results in less restrictions on plant operation.

NSHD Category	Change Number 3.4-	Discussion of Change
L	48	CTS 3.1.A.2.b(1). CTS requires the pressurizer safety valves (PSVs) to be operable in MODES 1, 2, and 3. A new Applicability Note has been included to be consistent with the guidance of NUREG-1431. This Note does not require the PSVs to meet the LCO lift setting limits in MODE 3, when the plant is started up, to allow adjusting the settings under hot conditions. Because this Note allows the PSVs to be potentially inoperable in MODE 3 until the PSVs can be tested and set, this change is less restrictive. This change is acceptable since the Note also requires preliminary lift settings under cold conditions prior to heatup which is consistent with current plant practices and the time during which the valves are potentially inoperable for valve setting adjustments is limited to 36 hours by the Note. The Note may improve plant safety by allowing more accurate settings under hot conditions.
A	49	CTS 3.1.A.2.c(1)(b). CTS required action explicitly requires the operators to "either restore the PORV to OPERABLE status or" take other appropriate actions. The option to restore equipment to operable status is always an available means of exiting the ITS Required Actions and, under the Writers Guide, this option should not be stated unless there are no other actions which should be taken. Thus this statement and similar statements within this CTS paragraph are not included in the ITS. Since it is understood that these are viable operator actions, this is an administrative change.

NSHD Category	Change Number 3.4-	Discussion of Change
L	50	CTS 3.1.A.2.c.(2). When the RCS temperature is greater than the SI pump disable temperature and less than the OPPS enable temperature, CTS requires the plant to be in MODE 5 with the RCS depressurized and vented through at least a 3 square inch vent within 8 hours when a PORV is not restored within the allowed outage time or both PORVs are inoperable. This change will allow 12 hours to place the plant in MODE 5, depressurize and vent the RCS under these conditions. This change is acceptable because 8 hours is considered by the industry to be insufficient time to perform an orderly shutdown. The additional time will maintain plant safety by allowing the operators to plan the shutdown and prevent challenges to plant systems which may initiate an overpressure event which the shutdown intends to prevent. This change is consistent with the guidance of NUREG-1431 as modified by approved TSTF-352, Revision 1. Since this change allows more time to fully implement remedial action, this is a less restrictive change.

NSHD Category	Change Number 3.4-	Discussion of Change
M	51	CTS 3.1.A.2.c(2) and 3.1.A.2.c(3). Restrictions on SI pump operability isolation have been included for clarity. These requirements are specified in CTS LCO 3.3. New Action Statements are included for the possibility one (Specification 3.4.13) or two (Specification 3.4.12) SI pumps are capable of injecting into the RCS. CTS does not contain these action statement requirements; thus in this event CTS LCO 3.0.C (equivalent to PI ITS 3.0.3) would be entered which would allow 1 hour for action to be taken. Under these circumstances, 1 hour would be an inappropriate response time. The new Action Statements require immediate action which is more restrictive than CTS. This more restrictive change is included to make the PI ITS complete.
M	52	CTS 3.1.A.2.c(2) and 3.1.A.2.c(3). Restrictions on ECCS accumulator isolation have been included for clarity. These requirements are specified in CTS LCO 3.3. Associated action statements, clarifying notes for exceptions and applicability, and surveillance requirements, SRs 3.4.12.2 and 3.4.13.2, have been added. Since these are added requirements they are more restrictive on plant operations. These requirements have been included to make the PI ITS complete and consistent with the guidance of NUREG-1431.

NSHD Category	Change Number 3.4-	Discussion of Change
LR	53	CTS 3.1.A.2.c(2) and 3.1.A.2.c(3). The specific status of associated equipment has been relocated to the Bases for consistency with NUREG-1431. These provisions are assumed to be part of the OPERABILITY requirement for the PORVs. Since ITS Bases (under the Bases Control Program in Section 5.5 of the ITS) is licensee controlled, relocation of CTS requirements to the Bases is a less restrictive change.
M	54	New SRs, 3.4.12.1, 3.4.12.2, 3.4.12.3, 3.4.13.1, 3.4.13.2, 3.4.13.3, and 3.4.13.4, have been added in conformance with NUREG-1431. These SRs require verification that only one (or no) SI pump capable of injecting, RCS is vented and PORV block valves are open as applicable for the MODES of operation and the method of providing overpressurization protection. These SRs are more restrictive on plant operations since CTS do not require these surveillances. This is consistent with the guidance of NUREG-1431. These more restrictive changes are included to make the PI ITS complete.
	55	Not used.

NSHD Category	Change Number 3.4-	Discussion of Change
R	56	CTS 3.1.A.3 and 4.18. The reactor vessel head vent system specifications, CTS 3.1.A.3 and associated surveillance requirements in TS 4.18, are not included in the PI ITS since this system does not meet the 10CFR50.36 Technical Specification Selection Criteria. These vents are designed to exhaust noncondensable gases and steam from the RCS which could inhibit natural circulation following an accident with an extended loss of offsite power. However, these vents are not the primary success path and are used by the operators only if both pressurizer PORVs are unavailable. Credit for this vent system is not assumed in the safety analyses nor in the PI IPE. Therefore, the reactor vessel head vent does not meet the TSSC and these requirements have been relocated to the TRM which is maintained under the regulatory controls of 10CFR50.59. This is consistent with the guidance of NUREG-1431 which does not include specifications for the reactor vessel head vent system.

NSHD Category	Change Number 3.4-	Discussion of Change
M	57	New Specification. In conformance with the guidance of NUREG-1431, a new specification, RCS Minimum Temperature for Criticality, is included in the PI ITS. CTS Figure 3.1-1 does include a line labeled "Criticality Limit"; however there are no action statements or applicability associated with this line on the curve. The Bases for this new specification provide four considerations which the minimum temperature for criticality should satisfy. Based on PI calculations and consistency with the Low-Low T_{ave} setpoint, this LCO will specify 540°F as the minimum allowable temperature. Compliance with this specification will assure that plant conditions are conservative with respect to the initial conditions assumed in the safety analyses. Since this is a new specification, it does place additional restrictions on plant operations and is thus categorized as more restrictive. This more restrictive change is included to make the PI ITS complete.
	58	Not used.
	59	Not used.
	60	Not used.

NSHD Category	Change Number 3.4-	Discussion of Change
A	61	CTS 3.1.B.1.b. The requirements for maintaining RCS pressure and temperature limits has been broken into two sets of action statements, for conditions above 200°F and for conditions below 200°F. This is a logical split since the CTS action is to reduce the temperature below 200°F which would be meaningless if the pressure and temperature limits were violated and the temperature were already below 200°F. Since the new action statement is addressed below, this change is considered administrative. This change also includes changing to the NUREG-1431 use of MODES to define plant conditions rather than use of prose descriptions. Since use of MODES does not change any plant operations, this is also an administrative change.
M	62	CTS 3.1.B.1.b. CTS do not specify a time frame for evaluating the integrity of the RCS following an out-of-limit condition. For consistency with NUREG-1431, a limit of 72 hours is included. Since this is a new limit it is more restrictive on plant operation. This more restrictive change is included to make the PI ITS complete.
M	63	CTS 3.1.B.1.b. For consistency with NUREG-1431, new action statements have been included for out-of-limit conditions when the RCS is below 200°F. PI CTS do not address this condition. Since these are new requirements they are considered more restrictive on plant operations. This more restrictive change is included to make the PI ITS complete.

NSHD Category	Change Number 3.4-	Discussion of Change
M	64	A new surveillance requirement, SR 3.4.3.1, has been included for consistency with NUREG-1431 which requires verification that RCS pressure, temperature and heatup and cooldown rates are within the specified limits. Plant operators currently monitor these variables for compliance with Specification 3.1.B.1.b although it is not explicitly written as a TS SR. Since these are new TS requirements they are considered more restrictive on plant operations. This more restrictive change is included to make the PI ITS complete.
	65	Not used.
R	66	CTS 3.1.B.2.c. In conformance with the guidance of NUREG-1431, the pressurizer heatup and cooldown specifications have been relocated from the TS to the PTLR. This change is acceptable since the Bases for Specification 3.4.3 state that the reactor pressure vessel is the most limiting component for brittle fracture; thus the requirements for the pressurizer have not been included in the ITS. The shutdown requirements associated with pressurizer heatup and cooldown limitations have been relocated to the TRM.

NSHD Category	Change Number 3.4-	Discussion of Change
R	67	CTS 3.1.B.3. In conformance with the guidance of NUREG-1431, the Steam Generator pressure/temperature limits have been relocated to the PTLR. This operating restriction does not present a challenge to the integrity of a fission product barrier and this limit is not required for safe operation. These specifications do not meet the Technical Specification Selection Criteria defined in 10CFR50.36. The shutdown requirements associated with this pressure/temperature limit have been relocated to the TRM.

NSHD Category	Change Number 3.4-	Discussion of Change
L	68	CTS 3.1.C.1. CTS requires two means of leakage detection. One of those means is specified as the radionuclide detection instrumentation, the other means is not specified. For consistency with the guidance of NUREG-1431, a second means of detecting RCS leakage, containment sump A pump run time, has been specified. This is a more restrictive change which is acceptable since it is consistent with CTS Bases and USAR. Also in conformance with the guidance of NUREG-1431, action statements and remedial actions, in the event one or both means of leakage detection are inoperable, have been provided. The CTS are unnecessarily restrictive in that inoperability of one means of leakage detection requires shutdown of the plant. There are remedial measures which can be implemented to compensate for an inoperable leakage detection channel as allowed by NUREG-1431. The PI ITS includes provisions for compensatory sampling and surveillance testing if leakage detection channels are inoperable. Implementation of this change will provide increased plant operational flexibility and thus this is a less restrictive change.
	69	Not used.
	70	Not used.

NSHD Category	Change Number 3.4-	Discussion of Change
A	71	CTS 3.1.C.1. The CTS requirements described here are the same as ITS LCO 3.0.3 (CTS LCO 3.0.c) which is the appropriate action if no leakage detection instrumentation is operable. Thus this is rewritten to require immediate entry into LCO 3.0.3. The change of applicability of this requirement is addressed in L3.4-68. The time and the implications of the use of the time are the same; therefore this is an administrative change.
M	72	Four new SRs, 3.4.16.1, 3.4.16.2, 3.4.16.3 and 3.4.16.4, have been included to perform Channel Checks, COTs and Calibrations of containment radiation monitors and Calibration of sump pump run time instrumentation. Since these requirements are new to the PI TS they are considered more restrictive to plant operation. These SRs have been included for consistency with NUREG-1431. These more restrictive changes are included to make the PI ITS complete.
A	73	CTS 3.1.C.2.a. The CTS specifics of where leakage is not originating from have not been included in ITS LCO 3.4.14. The ITS definition of LEAKAGE includes the required details of what is considered leakage; thus these details are not necessary in the specification. Since this change does not change any plant operations, this is an administrative change.

NSHD Category	Change Number 3.4-	Discussion of Change
LR	74	CTS 3.1.C.2.b. The requirement to evaluate the leakage for continued safe operation is a level of detail beyond that contained within NUREG-1431. Therefore this requirement has been relocated to the Bases. Since ITS Bases (under the Bases Control Program in Section 5.5 of the ITS) is licensee controlled, this change is less restrictive.
	75	Not used.
	76	Not used.
A	77	CTS 3.1.C.2.d. The CTS requirement to remain in COLD SHUTDOWN until the condition is corrected is not included in ITS specification 3.4.14. The rules of ITS use in LCO 3.0.4 require ITS LCOs to be met prior to resumption of power operation. Thus, it is unnecessary to include these requirements in ITS LCO 3.4.14. Since the change does not involve any changes to plant operations, this is an administrative change.

NSHD Category	Change Number 3.4-	Discussion of Change
A	78	CTS 3.1.C.2.e. CTS requires inservice steam generator tube inspection in accordance with TS 4.12. Since this TS has been relocated to the Steam Generator Program in ITS Section 5.5, no technical changes are associated with this change; therefore, this is an administrative change.
L	79	CTS 3.1.C.3. A new action statement is included in the ITS which allows 4 hours to isolate the flow path of a PIV with leakage outside the allowed limits and 72 hours to restore the PIV to within limits. CTS allows one hour to initiate shutdown. These changes are consistent with the guidance of NUREG-1431. These changes are acceptable since the safety function of the PIV is met once the PIV flow path is isolated. Furthermore, the isolation valve is also required to meet the same leakage limits as the PIV, in accordance with the additional new requirement for isolation valve leakage limits. An additional three hours to take action is acceptable since these leakage limits are very conservative and it is very unlikely that these valves will suffer complete failure during this limited time. When the PIV flow path is isolated, this may cause some other system, such as RHR, to be inoperable and require entry into an Action Statement for another Specification which would require completion of required actions within 72 hours. Thus this change is consistent with other allowed outage times for significant safety features. Since this change allows additional plant operational flexibility, this is a less restrictive change.

NSHD Category	Change Number 3.4-	Discussion of Change
	80	Not used.
M	81	CTS 3.1.D.2.a. In accordance with the guidance of NUREG-1431, verification that the limits of Figure TS.3.1-3 (ITS Figure 3.4.17-1) are met shall be performed every 4 hours. Since CTS do not specify a time frequency, this is a more restrictive change. This change is acceptable since it will assure that the limits are met on a timely basis and it does not introduce any unsafe plant operating conditions.
L	82	CTS 3.1.C.3 and Table 4.1-2B, Item 4a. CTS RCS specific activity limits when the RCS temperature is below 500°F and above cold shutdown have not been retained. The purpose of Specification 3.4.17 is to limit SGTR releases to a small fraction of 10CFR100 limits. This change is acceptable because below 500°F the release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main stream safety valves. Since this change would not retain CTS requirements it is less restrictive on plant operations. This change is consistent with the guidance of NUREG-1431.

NSHD Category	Change Number 3.4-	Discussion of Change
A	83	CTS 3.3.A.3 and 3.3.A.4. For clarity and to be consistent with the guidance of NUREG-1431, "MODE 4" has been included in the description of when this specification is applicable. This is an administrative change since the plant is by definition in MODE 4 when the RCS temperature meets the criteria for the OPPS enable temperature.
M	84	CTS Table 4.1-1C, Item 25. For consistency with NUREG-1431, the functional test of the low temperature overpressure protection system will be performed monthly when the RCS temperature is below the LTOP enable temperature. Since CTS require this test each refueling outage, this is a more restrictive requirement. This change will provide additional assurance that the LTOP system will perform as required and is consistent with the surveillance interval for instrumentation providing similar plant protection.
M	85	Table 4.1-1C, Note 38. CTS do not specify a time frame within which this SR must be in compliance. Since ITS includes a time limit of 12 hours, this is a more restrictive change. This change conforms to the guidance of NUREG-1431 and is acceptable because it assures that the plant is maintained in a safe condition.

NSHD Category	Change Number 3.4-	Discussion of Change
L	86	CTS 4.6.C and Table 4.1-2A, Item 7. In accordance with GL 91-04, the surveillance interval for PORV functional testing and emergency pressurizer heater power supply are increased to 24 months to accommodate planned future extended reactor fuel cycles. Since this testing will occur less frequently, this is a less restrictive change. PORV functional tests were reviewed for a five year period and no problems were identified. Therefore it was concluded that an increased surveillance interval would have a minimal effect on plant safety. The emergency pressurizer heater power supply is currently tested prior to each refueling outage so that if a problem is identified it could be corrected during the ensuing outage. Review of testing experience on the emergency power supply did not identify any problems. Therefore it was concluded that an increased surveillance interval would have a minimal effect on plant safety.
L	87	CTS 4.3. This proposed change would revise the allowable PIV leakage of 1.0 gpm to 0.5 gpm per inch of nominal valve size up to 5 gpm maximum. This change is acceptable since the CTS 1.0 gpm limit imposes an unjustified limitation on larger valves. The restrictive limit, when applied to the larger valve, would require a repair effort when the relative degradation of the valve does not warrant the cost or exposure. A leakage limit based on valve size is more apt to provide meaningful information with respect to the mechanical condition of the valve, and is considered superior. This change is consistent with the guidance of NUREG-1431.

NSHD Category	Change Number 3.4-	Discussion of Change
L	88	CTS Table 4.1-2B, Item 1. The surveillance interval for RCS gross activity determination would be increased to once per week by this change in conformance with the guidance of NUREG-1431. This change is acceptable because fuel failures are most likely to occur during startup and fast power changes and not during steady state power operation during which the majority of sampling is performed. Gross fuel failures will also result in letdown radiation alarms and possibly containment radiation alarms providing additional operator indication.
L	89	CTS Table TS.4.1-2A, Item 9. This proposed change would increase the surveillance interval of RCS water inventory balances from once per day to 72 hours for consistency with SR 3.4.14.1. This increased surveillance interval is considered acceptable based on the leakage detection systems required to be operable by LCO 3.4.16 and the various indications available to the operators. These other indications include volume control tank level and radiation alarms.
	90	Not used.

NSHD Category	Change Number 3.4-	Discussion of Change
L	91	CTS Table 4.1-2B, Item 4a. The CTS requirement to sample once per 4 hours when specific activity exceeds 100 µCi/gram has not been included. In accordance with ITS 3.4.17 Action B, whenever this limit is not met, the plant is required to be in MODE 3 with T_{ave} < 500°F within 6 hours. Thus the requirement for sampling in 4 hours serves no useful purpose and is unnecessary. Since this change will require less sampling this change is less restrictive. This change is consistent with the guidance of NUREG-1431.
	92	Not used.
	93	Not used.
LR	94	CTS Table 4.1-2B, Items 5 and 6, and Note 2. The purpose of Specification 3.4.17, RCS Specific Activity is to limit the offsite radioactivity dose consequences from a SGTR to a small fraction of 10CFR100. This change will relocate Items 5 and 6, RCS Radiochemistry and RCS Tritium activity, from CTS Table TS.4.1-2B to the TRM since these items are not significant in limiting SGTR offsite dose and therefore should not be in TS. This is less restrictive since the TRM is under licensee control. However this change is acceptable since the TRM is under the controls of 10CFR50.59. This change conforms the PI ITS to the guidance of NUREG-1431.

NSHD Category	Change Number 3.4-	Discussion of Change
	95	Not used.
LR	96	CTS Table 4.1-2B, Item 8 and Note 4. RCS boron concentration measurement at power was not included in the ITS since RCS Chemistry does not meet the NRC Policy Statement for TS Screening Criteria and is not required to be addressed within the TS. This requirement is relocated to the TRM. While this is a less restrictive change since the TRM is under licensee control, this change is acceptable because the TRM is under the controls of 10CFR50.59. This change is consistent with the guidance of NUREG-1431.
LR	97	CTS 4.3. For consistency with NUREG-1431, the list of valves and the test methodology have been relocated to the Bases. This detailed information is not required in the TS to run the plant in a safe manner. Since ITS Bases (under the Bases Control Program in Section 5.5 of the ITS) is licensee controlled, relocation of CTS requirements to the Bases is a less restrictive change.
LR	98	CTS 4.6.C. The methodology for performing this surveillance has been relocated to the Bases. This detailed information is not required in the TS to run the plant in a safe manner. Since ITS Bases (under the Bases Control Program in Section 5.5 of the ITS) is licensee controlled, relocation of CTS requirements to the Bases is a less restrictive change.

NSHD Category	Change Number 3.4-	Discussion of Change
A	99	CTS 4.3. CTS require this SR to be performed prior to resuming power after each refueling. Since the NUREG-1431 format requires SR Frequency statement in months, this is revised to every 24 months. Since this change does not materially change the testing of these valves, this is an administrative change.