

**INSERT THIS PAGE IN FRONT OF VOLUME 6**

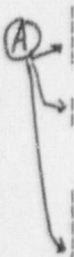
<b>Volume 6: SECTION 3.4</b>	
<b>Remove</b>	<b>Replace</b>
3.4.1 ITS pg 3.4-1 Rev 2	3.4.1 ITS pg 3.4-1 Rev 4
3.4.1 CTS M/U (3/4 4-2) pg 3 of 6	3.4.1 CTS M/U (3/4 4-2) pg 3 of 6 Rev 4
3.4.1 DOCs pg 4 Rev 2	3.4.1 DOCs pg 4 Rev 4
3.4.1 NUREG M/U pg 3.4-1 Rev 2	3.4.1 NUREG M/U pg 3.4-1 Rev 4
B 3.4.3 ITS pg B 3.4.3-4 Rev 0	B 3.4.3 ITS pg B 3.4.3-4 Rev 4
3.4.3 DOCs pg 2 Rev 0	3.4.3 DOCs pg 2 Rev 4
B 3.4.3 NUREG M/U pg B 3.4-15	B 3.4.3 NUREG M/U pg B 3.4-15 Rev 4
3.4.4 CTS M/U (3/4 4-10) pg 1 of 2	3.4.4 CTS M/U (3/4 4-10) pg 1 of 2 Rev 4
3.4.4 DOCs pg 1 Rev 0	3.4.4 DOCs pg 1 Rev 4
3.4.4 DOCs pg 2 Rev 0	3.4.4 DOCs pg 2 Rev 4
3.4.4 DOCs pg 3 Rev 0	3.4.4 DOCs pg 3 Rev 4
3.4.5 ITS pg 3.4-11 Rev 0	3.4.5 ITS pg 3.4-11 Rev 4
3.4.5 ITS pg 3.4-12 Rev 0	3.4.5 ITS pg 3.4-12 Rev 4
3.4.5 CTS M/U (3/4 4-12) pg 3 of 3	3.4.5 CTS M/U (3/4 4-12) pg 3 of 3 Rev 4
3.4.5 DOCs pg 3 Rev 0	3.4.5 DOCs pg 3 Rev 4
3.4.5 NUREG M/U pg 3.4-9	3.4.5 NUREG M/U pg 3.4-9 Rev 4
3.4.5 NUREG M/U pg 3.4-11	3.4.5 NUREG M/U pg 3.4-11 Rev 4
B 3.4.5 NUREG M/U pg B 3.4-25	B 3.4.5 NUREG M/U pg B 3.4-25 Rev 4
3.4.5 JFD's pg 1 Rev 0	3.4.5 JFD's pg 1 Rev 4
3.4.6 ITS pg 3.4-13 Rev 0	3.4.6 ITS pg 3.4-13 Rev 4
3.4.6 ITS pg 3.4-14 Rev 0	3.4.6 ITS pg 3.4-14 Rev 4
B 3.4.6 ITS pg B 3.4.6-5 Rev 0	B 3.4.6 ITS pg B 3.4.6-5 Rev 4
3.4.6 DOCs pg 2 Rev 0	3.4.6 DOCs pg 2 Rev 4
3.4.6 NUREG M/U pg 3.4-12	3.4.6 NUREG M/U pg 3.4-12 Rev 4
B 3.4.6 NUREG M/U pg B 3.4-32	B 3.4.6 NUREG M/U pg B 3.4-32 Rev 4
3.4.6 JFD's pg 1 Rev 0	3.4.6 JFD's pg 1 Rev 4
3.4.7 CTS M/U (3/4 4-17) pg 2 of 3	3.4.7 CTS M/U (3/4 4-17) pg 2 of 3 Rev 4
3.4.7 CTS M/U (3/4 4-18) pg 3 of 3	3.4.7 CTS M/U (3/4 4-18) pg 3 of 3 Rev 4
3.4.7 DOCs pg 1 Rev 0	3.4.7 DOCs pg 1 Rev 4
3.4.7 DOCs pg 2 Rev 0	3.4.7 DOCs pg 2 Rev 4
3.4.7 DOCs pg 3 Rev 0	3.4.7 DOCs pg 3 Rev 4
3.4.7 DOCs pg 4 Rev 0	3.4.7 DOCs pg 4 Rev 4
--	3.4.7 NSHC pg 7 Rev 4

**Volume 6: SECTION 3.4 (cont'd)**

Remove	Replace
3.4.8 DOCs pg 2 Rev 0	3.4.8 DOCs pg 2 Rev 4
3.4.9 DOCs pg 1 Rev 0	3.4.9 DOCs pg 1 Rev 4
3.4.9 DOCs pg 2 Rev 0	3.4.9 DOCs pg 2 Rev 4
3.4.9 DOCs pg 3 Rev 0	3.4.9 DOCs pg 3 Rev 4
3.4.10 ITS pg 3.4-24 Rev 0	3.4.10 ITS pg 3.4-24 Rev 4
3.4.10 CTS M/U (3/4 4-19) pg 5 of 8	3.4.10 CTS M/U (3/4 4-19) pg 5 of 8 Rev 4
3.4.10 DOCs pg 1 Rev 0	3.4.10 DOCs pg 1 Rev 4
3.4.10 DOCs pg 2 Rev 0	3.4.10 DOCs pg 2 Rev 4
3.4.10 DOCs pg 3 Rev 2	3.4.10 DOCs pg 3 Rev 4
3.4.10 NUREG M/U pg 3.4-24 (Insert) Rev 0	3.4.10 NUREG M/U pg 3.4-24 (Insert) Rev 4
3.4.11 CTS M/U (3/4 4-23) pg 1 of 1	3.4.11 CTS M/U (3/4 4-23) pg 1 of 1 Rev 4
3.4.11 DOCs pg 1 Rev 0	3.4.11 DOCs pg 1 Rev 4
3.4.11 DOCs pg 2 Rev 0	3.4.11 DOCs pg 2 Rev 4
3.4.11 NSHC pg 1 Rev 0	3.4.11 NSHC pg 1 Rev 4
--	3.4.11 NSHC pg 2 Rev 4

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating



LCO 3.4.1 The reactor core shall not exhibit core thermal-hydraulic instability or operate in the "Scram" or "Exit" Regions.

AND

a. Two recirculation loops with matched recirculation loop jet pump flows shall be in operation;

OR

b. One recirculation loop may be in operation provided:

1. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Simulated Thermal Power - Upscale) Allowable Value of Table 3.3.1.1-1 is reset for single loop operation, when in MODE 1.

-----NOTE-----  
Required allowable value modification for single loop operation may be delayed for up to 4 hours after transition from two recirculation loop operations to single recirculation loop operation.  
-----

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Recirculation jet pump loop flow mismatch not within limits.	A.1 Declare recirculation loop with lower flow: "not in operation."	2 hours

(continued)

SPECIFICATION 3.4.1

< Also see Specification 3.4.10 >  
< Also see Specification 3.5.1 >

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

See Specification 3.5.1

4.4.1.1.1 Each pump discharge valve shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each STARTUP\* prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER.

~~4.4.1.1.2 DELETED~~

A.1

AMEND # 130

4.4.1.1.3 With one reactor coolant system recirculation loop not in operation, at least once per 12 hours verify that:  
a. THERMAL POWER is less than or equal to 67.2% of RATED THERMAL POWER, and  
b. The individual recirculation pump flow controller for the operating recirculation pump is in the Manual mode, and  
c. The speed of the operating recirculation pump is less than or equal to 75% of rated pump speed.

4.4.1.1.4 With one reactor coolant system loop not in operation with THERMAL POWER less than or equal to 30% of RATED THERMAL POWER or with recirculation loop flow in the operating loop less than or equal to 50% of rated loop flow, verify the following differential temperature requirements are met within no more than 15 minutes prior to either THERMAL POWER increase or recirculation flow increase:

See Specification 3.4.10

- a. Less than or equal to 145°F between reactor vessel steam space coolant and bottom head drain line coolant, and
- b. Less than or equal to 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel\*\*, and
- c. Less than or equal to 50°F between the reactor coolant within the loop not in operation and the operating loop.\*\*

See Specification 3.5.1

\*If not performed within the previous 31 days.

See Specification 3.4.10

\*\*Requirement does not apply when the recirculation loop not in operation is isolated from the reactor pressure vessel.

DISCUSSION OF CHANGES  
ITS: SECTION 3.4.1 - RECIRCULATION LOOPS OPERATING

LA.3 CTS 3.4.10, and Figure 3.4.10-1, detail various power-to-flow operating regions for the reactor core; CTS Actions a, b, and c specify requirements for operation in these regions; and CTS 4.4.10.2 details the specific region to monitor for instability when operating near the instability region. These controls are based on Generic Letter 94-02, "Long Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors" and the June 1994 BWROG Letter (no. 94078). ITS 3.4.1 maintains the scope and intent of CTS requirements and Actions, but relocates the definition of what constitutes core thermal-hydraulic instability, and the Figure that contains the definitions of the various regions, to the ITS Bases. Bases revisions require change control in accordance with ITS 5.5.10, Bases Control Program. These relocations continue to provide adequate protection of the public health and safety since the ITS retain sufficient requirements related to maintaining appropriate response to thermal hydraulic instabilities.

Rev 2  
A

TECHNICAL CHANGES - LESS RESTRICTIVE  
"Specific"

L.1 CTS 3.4.1.3 Applicability applies the recirculation pump speed (revised to "loop flow" in another discussion) mismatch "during two recirculation loop operation." ITS SR 3.4.1.2 also applies the mismatch criteria to "when both recirculation loops [are] in operation"; but adds a Note to the SR allowing 24 hours after establishing both loops in operation before the SR is required to be performed. This allowance is necessary to avoid intentional entry into the Actions each time the second recirculation pump is started and the SR has not been performed within its required Frequency. This change, although less restrictive, will have a negligible impact on safety.

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)  
3.4.1 Recirculation Loops Operating

LCO 3.4.1

**AND**  
a. Two recirculation loops with matched flows shall be in operation,  
*recirculation loop jet pump*  
*The reactor core shall not exhibit core thermal hydraulic instability or operate in the "Scram" or "Exit" Regions.*

**OR**  
b. One recirculation loop may be in operation provided the following limits are applied when the associated LCO is applicable:  
*P.5*

*P.1*  
1. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits [specified in the COLR];

*P.1*  
2. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits [specified in the COLR]; and

*P.1*  
3. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors ~~Flow Biased~~ Simulated Thermal Power ~~High~~), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation *when in MODE 1* *Upscale* *P.4*

{3.4.1.1}  
{3.4.1.3}

Action a.i.e)

INSERT 3.4.1-1

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<del>A.1 Requirements of the LCO not met.</del>	<del>A.1 Satisfy the requirements of the LCO.</del>	<del>24 hours</del>

\*Pending resolution of stability issues

(continued)

INSERT 3.4.1-2

~~Rev 2~~  
Rev 4

## BASES

---

SURVEILLANCE REQUIREMENTS (continued)

The SR gives set pressures for all 15 SRVs installed. However, since only 11 SRVs are required, the SR is met if 11 SRVs are set properly.

The Frequency is required by the Inservice Testing Program and is consistent with the fact that Surveillance must be performed during shutdown conditions.

SR 3.4.3.2

A manual actuation of each required SRV is performed to verify that, mechanically, the valve is functioning properly and no blockage exists in the valve discharge line. This can be demonstrated by the response of the turbine control valves or bypass valves, by a change in the measured steam flow, or by any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the SRVs divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is to be performed is  $\geq 850$  psig (the pressure recommended by the valve manufacturer). Adequate steam flow is represented by turbine bypass valves open at least 20%. Plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME Code requirements, prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. If a valve fails to actuate due only to the failure of the solenoid but is capable of opening on overpressure, the safety function of the SRV is considered OPERABLE.

DISCUSSION OF CHANGES  
ITS: SECTION 3.4.3 - SAFETY RELIEF VALVES (SRVs)

LA.2 CTS 3.4.2.1.c and 4.4.2.1.1 requires the SRV position indicators be Operable. Additionally, CTS 4.4.2.1.1 requires the SRV position indicators be demonstrated Operable with the pressure setpoint of each of the tail-pipe pressure switches verified by performance of a Channel Calibration. The SRV position indicators do not impact the Operability of the SRVs. DECo knows of no plant specific reason that these position indicators would meet the 10 CFR 50.36 criteria for inclusion in the ITS. ITS 3.4.3 requires the Operability of the SRVs, but does not require that the position indicators are Operable. Therefore, these requirements will be relocated to the Technical Requirements Manual (TRM), which require revisions to be controlled by 10 CFR 50.59. This relocation continues to provide adequate protection of the public health and safety since the requirement for SRV Operability continues to be required by the Technical Specifications, and is consistent with the NUREG-1433.

RAI#2  
RAI 0.0-1

LR.1 CTS SR 4.4.2.1.2 requires 1/2 of the SRVs to be set-pressure tested at least once per 18 months, such that all 15 SRVs are set pressure tested at least once per 40 months. These requirements are considered details of methods for performing the surveillance and are not included in the ITS. These details do not impact the Operability of the SRVs. ITS 3.4.3 continues to require the Operability of the SRVs and requires they be tested in accordance with the IST Program. Regulatory control of changes to these requirements (e.g., Technical Specification amendment or 10 CFR 50.59) is not necessary to provide adequate protection of the public health and safety since the requirement for SRV Operability continues to be required by the Technical Specifications. This is consistent with the NUREG-1433.



BASES (continued)

SURVEILLANCE  
REQUIREMENTS

SR 3.4.3.1

This Surveillance requires that the ~~required~~ S/RVs will open at the pressures assumed in the safety analysis of Reference 1. The demonstration of the S/RV safe lift settings must be performed during shutdown, since this is a bench test, to be done in accordance with the Inservice Testing Program. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. The S/RV setpoint is ~~± 3%~~ for OPERABILITY; however, the valves are reset to ~~± 1%~~ during the Surveillance to allow for drift.

P.4

The SR gives set pressures for all 15 SRVs installed. However, since only 11 SRVs are required, the SR is met if 11 SRVs are set properly.

The ~~18 month~~ Frequency ~~was selected because this~~ Surveillance must be performed during shutdown conditions ~~and is based on the time between refuelings~~ ...

is required by the Inservice Testing Program, and is consistent with the fact that

P.3

SR 3.4.3.2

A manual actuation of each ~~required~~ S/RV is performed to verify that, mechanically, the valve is functioning properly and no blockage exists in the valve discharge line. This can be demonstrated by the response of the turbine control valves or bypass valves, by a change in the measured steam flow, or by any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the S/RVs divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is to be performed is ~~[920]~~ psig (the pressure recommended by the valve manufacturer). Adequate steam flow is represented by ~~[at least 1.25 turbine bypass valves open, or total steam flow ≥ 10<sup>6</sup> lb/hr]~~. Plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME Code requirements, prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable

≥ 850

turbine bypass valves open at least 20%

(A)

(continued)

Specification 3.4.4  
(Also see specification 3.4.5)

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

A.1

LIMITING CONDITION FOR OPERATION

LCO  
3.4.4

~~3.4.3.2~~ Reactor coolant system leakage shall be limited to:

3.4.4.a ~~a~~ No PRESSURE BOUNDARY LEAKAGE.

3.4.4.b ~~b~~ 5 gpm UNIDENTIFIED LEAKAGE.

3.4.4.c ~~c~~ 25 gpm total leakage averaged over any 24-hour period.

see  
specification  
3.4.5

d. Leakage specified in Table 3.4.3.2-1 at a reactor coolant system pressure of 1045 ± 10 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.

3.4.4.d ~~d~~ 2 gpm increase in UNIDENTIFIED LEAKAGE within ~~any~~ 24 hour period during OPERATIONAL CONDITION 1.

the previous

A.2

3.4.5

f. 2 gpm increase in UNIDENTIFIED LEAKAGE within any 4 hour period during OPERATIONAL CONDITIONS 2 and 3.

L.2

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

Action C a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

Action A b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Action C

see  
specification  
3.4.5

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one other closed manual, deactivated automatic, or check\* valve, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the high/low pressure interface valve leakage pressure monitors shown in Table 3.4.3.2-2 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

see  
specification  
3.4.5

\*Which has been verified not to exceed the allowable leakage limit at the last refueling outage or after the last time the valve was disturbed, whichever is more recent.

DISCUSSION OF CHANGES  
ITS: SECTION 3.4.4 - RCS OPERATIONAL LEAKAGE

ADMINISTRATIVE

- A.1 In the conversion of the Fermi 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Boiling Water Reactor (BWR) Standard Technical Specifications NUREG-1433, Rev. 1.
- A.2 CTS 3.4.3.2.c and 3.4.3.2.e allow Leakage to be averaged over "any 24-hour period." ITS 3.4.4.c and 3.4.4.d specifies that the Leakage is averaged over the "previous 24 hour period." The intent of the CTS is that the total Leakage limit applies, to the previous 24 hours (not any future or past 24 hour periods). This results in a "rolling average" covering "any 24-hour period." Therefore, changing "any" to "the previous" does not change the intent of the CTS requirement. This change is administrative with no impact on safety.

RAI #3

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

- LA.1 CTS 4.4.3.2.1 and footnote \* detail the specific methods for performing the Surveillance for leakage monitoring. ITS SR 3.4.4.1 requires the performance of the Surveillance, but does not provide specific details on the ways the Surveillance can be performed. Defining the method used to determine RCS Leakage is not necessary to ensure Leakage does not exceed the LCO limits and detect a degradation of the RCPB does not impact the requirement to maintain the RCPB Leakage within specified limits. Therefore, these details can be relocated to the Bases. This change is consistent with NUREG-1433. The information moved to the Bases requires changes to be controlled in accordance with the ITS 5.5.10, Bases Control Program. This relocation continues to provide adequate protection of the public health and safety since the requirement for determining acceptable leakage rates continues to be required by the Technical Specifications.

DISCUSSION OF CHANGES  
ITS: SECTION 3.4.4 - RCS OPERATIONAL LEAKAGE

TECHNICAL CHANGES - LESS RESTRICTIVE  
"Specific"

- L.1 CTS 4.4.3.2.1 requires that leakage be demonstrated to be within limit, in part, by monitoring primary containment atmospheric gaseous radioactivity be monitored at least once per 4 hours. However, the remaining parts of CTS 4.3.2.1 require leakage be demonstrated to be within limit by monitoring once per 12 hours; and this 12 hour monitoring is done on the system that actually quantifies the leakage (the atmospheric gaseous radioactivity monitor is not utilized to quantify the leakage for comparison to the LCO limit, as provided in CTS footnote \* to the referenced surveillance). ITS SR 3.4.4.1 requires verification every 12 hours that the RCS unidentified and total Leakage, and unidentified Leakage increase, are within limits consistent with these latter CTS requirements. RCS Leakage is monitored by a variety of instruments designed to provide alarms when excessive Leakage is indicated and to quantify the various types of Leakage. In conjunction with alarms and administrative controls, a 12 hour Frequency for this Surveillance is appropriate for identifying Leakage and for tracking trends. Note also that in Mode 1 this Frequency is restricted from applying the 25% extension of ITS SR 3.0.2. This change is consistent with Generic Letter 88-01, will provide an effective means to determine any adverse trends and will have a negligible impact on safety.

Handwritten annotations on the right margin:  
A vertical line with several horizontal tick marks.  
An arrow pointing to the right from the top of the line.  
An arrow pointing to the right from the middle of the line.  
The handwritten text "XAI #5" written vertically.  
An arrow pointing to the right from the bottom of the line.

DISCUSSION OF CHANGES  
ITS: SECTION 3.4.4 - RCS OPERATIONAL LEAKAGE

L.2 CTS 3.4.3.2.f, Action f, 4.4.3.2.1.b and 4.4.3.2.1.c contain requirements for Unidentified Leakage during Operational Conditions 2 and 3. ITS 3.4.4 revises the unidentified Leakage rate increase limit to be applicable only in Mode 1, instead of the CTS required Operational Conditions 1, 2, and 3. As the plant starts up and increases pressure, increasing leakage will occur due to increasing pressure. This could result in a requirement for a unit shutdown, even though there is no new leakage source with potential of rapid propagation. This change allows the limit to be applied only after Mode 1 is achieved, which is when reactor pressure has effectively reached normal operating pressure. The overall 5 gpm unidentified Leakage limit is still required to be met during the startup. This limit is much below the expected flow from a critical crack in the primary system, and provides adequate protection during the brief exception to the leakage increase limit. This change is also consistent with NUREG-1433. Therefore, this less restrictive change will have a minimal impact on safety. As discussed in DOC L.1 above, the 12 hour Frequency for monitoring for RCS Leakage is adequate. The CTS 4.4.3.2.1.b and c frequency of once per 4 hours in MODES 2 and 3 is related to the appropriate monitoring frequency for the unidentified leakage rate increase limit, which is eliminated for MODE 2 and 3 (as discussed in this L.2 DOC). As such, this once per 4 hour frequency can be eliminated, leaving the once per 12 hour surveillance.

RAI #5

RELOCATED SPECIFICATIONS

None

TECHNICAL SPECIFICATION BASES

The CTS Bases for this Specification have been replaced by Bases that reflect the format and applicable content of ITS 3.4.4 consistent with the BWR STS, NUREG-1433, Rev. 1.

ACTIONS

-----NOTES-----

1. Separate Condition entry is allowed for each flow path.
  2. Enter applicable Conditions and Required Actions for systems made inoperable by PIVs.
- 

RAI #7

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more flow paths with leakage from one or more RCS PIVs not within limit.</p>	<p>-----NOTE----- Each check valve used to satisfy Required Action A.1 must have been verified to meet SR 3.4.5.1 at the last refueling outage or after the last time the valve was disturbed, whichever is more recent.</p> <p>-----</p> <p>A.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of one other closed manual, de-activated automatic, or check valve.</p>	<p>4 hours</p>
<p>B. Required Action and associated Completion Time not met.</p>	<p>B.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>B.2 Be in MODE 4.</p>	<p>12 hours</p> <p>36 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.5.1 -----NOTE----- Not required to be performed in MODE 3. -----</p> <p>Verify equivalent leakage of each RCS PIV, at an RCS pressure <math>\geq 1035</math> and <math>\leq 1055</math> psig:</p> <p>a. For PIVs other than LPCI loop A and B injection isolation valves is <math>\leq 0.5</math> gpm per nominal inch of valve size up to a maximum of 5 gpm;</p> <p>b. For LPCI loop A and B outboard injection isolation valves is <math>\leq 0.4</math> gpm through-seat, and <math>\leq 5</math> ml/min external leakage; and</p> <p>c. For LPCI loop A and B inboard injection isolation testable check valves is <math>\leq 10</math> gpm.</p>	<p>In accordance with the Inservice Testing Program</p>

RAI # 62

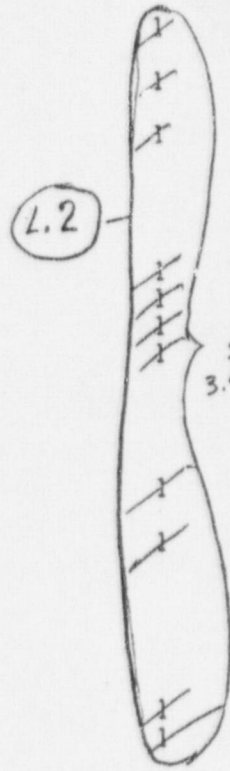
# Specification 3.4.5

**TABLE 3.4.3.2-1**  
**REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES**

VALVE NUMBER	VALVE DESCRIPTION
<b>1. RHR System</b>	
E11-F015A	LPCI Loop A Injection Isolation Valve
E11-F015B	LPCI Loop B Injection Isolation Valve
E11-F050A	LPCI Loop A Injection Line Testable Check Valve
E11-F050B	LPCI Loop B Injection Line Testable Check Valve
E11-F008	Shutdown Cooling RPV Suction Outboard Isolation Valve
E11-F009	Shutdown Cooling RPV Suction Inboard Isolation Valve
E11-F608	Shutdown Cooling Suction Isolation Valve
<b>2. Core Spray System</b>	
E21-F005A	Loop A Inboard Isolation Valve
E21-F005B	Loop B Inboard Isolation Valve
E21-F006A	Loop A Containment Check Valve
E21-F006B	Loop B Containment Check Valve
<b>3. High Pressure Coolant Injection System</b>	
E41-F007	Pump Discharge Outboard Isolation Valve
E41-F006	Pump Discharge Inboard Isolation Valve
<b>4. Reactor Core Isolation Cooling System</b>	
E51-F012	Pump Discharge Isolation Valve
E51-F013	Pump Discharge to Feedwater Header Isolation Valve

MAXIMUM LEAKAGE (gpm)

SR 3.4.5.1.b { 0.4<sup>(a)</sup>  
0.4<sup>(a)</sup>  
SR 3.4.5.1.c { 10  
10



SR 3.4.5.1.b

(a) External Leakage from this valve shall be limited to 5 ml/min.

(A)

**TABLE 3.4.3.2-2**  
**REACTOR COOLANT SYSTEM INTERFACE VALVES LEAKAGE PRESSURE MONITORS**

VALVE NUMBER	SYSTEM	ALARM SETPOINT (psig)
E11-F015A & B, E11-F050A & B	RHR LPCI	≤ 449
E11-F008, F009, F608	RHR Shutdown Cooling	≤ 135
E21-F005A & B, E21-F006A & B	Core Spray	≤ 452
E41-F006, F007	HPCI	≤ 71
E51-F012, F013	RCIC	≤ 71

LR.1



DISCUSSION OF CHANGES

ITS: SECTION 3.4.5 - RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

LR.1

CTS 3.4.3.2 Action d 4.4.3.2.3, and Table 3.4.3.2-2 contain Actions, Surveillance details, and a specific list of PIV leakage pressure monitors related to alarm-only functions, which are removed from Technical Specifications. These alarm-only functions are not assumed in any accident analysis. Alarm-only functions do not relate directly to the Operability requirements for the Reactor Coolant System. ITS does not specify indication-only, alarm-only, or test equipment to be Operable to support Operability of a system or component. Additionally, DECo knows of no Fermi-specific reason these monitors would be required to be retained. Regulatory control of changes to these requirements (e.g., Technical Specification amendment or 10 CFR 50.59) is not necessary to provide adequate protection of the public health and safety since alarm-only functions do not relate directly to the Operability requirements for the system or analysis assumptions and the requirement for RCS and PIV leakage limits continues to be required by the Technical Specifications.

RAI 3.4-1

RAI 3.4-9

LR.2

CTS 4.4.3.2.2.b requires that any time the leakage rate of a PIV is affected by maintenance, repair, or replacement, post maintenance testing is required to demonstrate Operability of the PIV. ITS requires the PIV Operability, but does not direct the performance of testing when repair activities have been performed. The requirement to perform post maintenance testing is applicable to all plant equipment. The majority of CTS LCOs do not contain this requirement to perform post maintenance testing although this requirement is applicable. Therefore, these post maintenance testing requirement details can be removed from Technical Specifications consistent with NUREG-1433. Regulatory control of changes to these requirements (e.g., Technical Specification amendment or 10 CFR 50.59) is not necessary to provide adequate protection of the public health and safety since the requirement for PIV Operability continues to be required by the Technical Specifications.

RAI 3.4-1

3.4 REACTOR COOLANT SYSTEM (RCS)

<CTS>

3.4.5 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.5 The leakage from each RCS PIV shall be within limit. <3.4.3.2.d>

APPLICABILITY: MODES 1 and 2.  
MODE 3, except valves in the residual heat removal (RHR) shutdown cooling flow path when in, or during the transition to or from, the shutdown cooling mode of operation. <DOC L.1>

ACTIONS

NOTES

1. Separate Condition entry is allowed for each flow path.
2. Enter applicable Conditions and Required Actions for systems made inoperable by PIVs. <DOC A.2>

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more flow paths with leakage from one or more RCS PIVs not within limit.	<p>check</p> <p>NOTE</p> <p>Each valve used to satisfy Required Action A.1 and Required Action A.2 must have been verified to meet SR 3.4.5.1, and be in the reactor coolant pressure boundary for the high pressure portion of the system.</p>	<p>&lt;3.4.3.2 Action C&gt;</p> <p>RAI # 7</p>
		(continued)

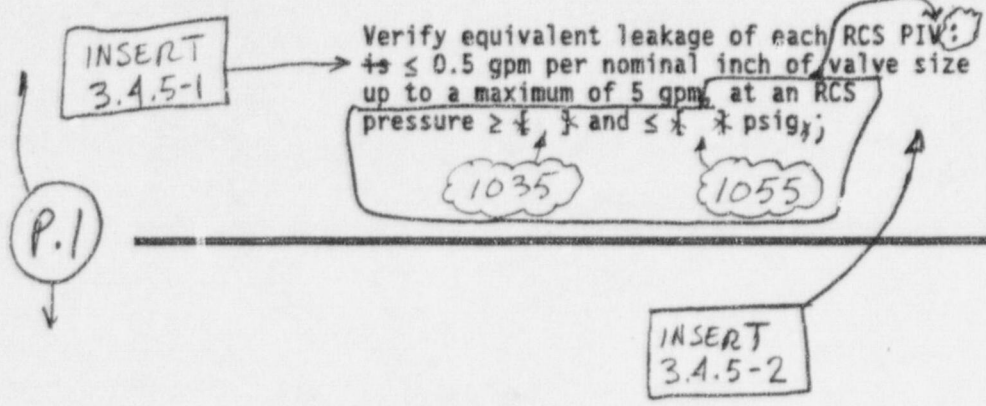
P.I

at the last refueling outage or after the last time the valve was disturbed, whichever is more recent.

<CTS>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.5.1</p> <p>-----NOTE----- Not required to be performed in MODE 3.</p> <p>Verify equivalent leakage of each RCS PIV:  <del>is</del> <math>\leq 0.5</math> gpm per nominal inch of valve size                      up to a maximum of 5 gpm, at an RCS                      pressure <math>\geq</math> * * and <math>\leq</math> * * psig;</p>	<p>&lt;4.4.3.2.2&gt; &lt;3.4.3.2.d&gt;</p> <p>In accordance with the Inservice Testing Program or                      18 months</p>



RAI 6

Rev 4

BASES

ACTIONS

P.3

A.1 and A.2 (continued) check

is

INSERT  
B 3.4.5-2

Required Action A.1 ~~and Required Action A.2~~ are modified by a Note stating that ~~the~~ valves used for isolation must meet the same leakage requirements as the PIVs ~~and must be on the RCPB for the high pressure portion of the system~~.

RAI-7

Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the flow path if leakage cannot be reduced while corrective actions to reseal the leaking PIVs are taken. The 4 hours allows time for these actions and restricts the time of operation with leaking valves.

P.3

~~Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing another valve qualified for isolation or restoring one leaking PIV. The 72 hour Completion Time considers the time required to complete the action, the low probability of a second valve failing during this time period, and the low probability of a pressure boundary rupture of the low pressure ECCS piping when overpressurized to reactor pressure (Ref. 7)~~

B.1 and B.2

If leakage cannot be reduced or the system isolated, 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 within 12 hours and MODE 4 within 36 hours. This action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The Completion Times are reasonable, based on operating experience, to achieve the required plant conditions from full power conditions in an orderly manner and without challenging plant system

SURVEILLANCE  
REQUIREMENTS

SR 3.4.5.1

P.3

INSERT  
B 3.4.5-3

Performance of leakage testing on each RCS PIV 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 to 5 gpm maximum applies to each valve.~~ Leakage testing requires a stable pressure condition. For the two PIVs in series, the leakage

(continued)

JUSTIFICATION FOR DIFFERENCES FROM NUREG - 1433  
ITS: SECTION 3.4.5 - RCS PIV LEAKAGE

NON-BRACKETED PLANT SPECIFIC CHANGES

- P.1 These changes are made to NUREG-1433 to reflect Fermi 2 current licensing basis; including design features, existing license requirements and commitments. Refer to CTS Discussion Of Changes to the related requirement for a detailed justification of changes made to the current licensing basis which are also reflected in the ITS as presented. Additional rewording, reformatting, and revised numbering is made to incorporate these changes consistent with Writer's Guide conventions. Specifically:
- a. ISTS 3.4.5 Action A provides a generic Note detailing limitations on valves used for isolating a flow path with an inoperable PIV. The Fermi-2 CTS provides a Fermi-specific current licensing basis note (3.4.3.2 footnote \*) with the applicable limitations. These Fermi-specific limitations are incorporated in to the ITS, replacing the generic limitations.
- P.2 Bases changes are made to reflect plant specific design details, equipment terminology, and analyses.
- P.3 Bases changes are made to reflect changes made to the Specification. Refer to the Specification change (and associated JFD) for additional detail.
- P.4 Change made for editorial preference or clarity.
- P.5 The Bases "definition" of RCS PIVs is not an accurate presentation. First, some PIV pairs consist of one normally closed valve and one interlocked-to-close (but normally open) valve. Second, many pairs of in-series, normally closed, valves are within the RCPB, but are not PIVs. This inaccurate statement can be eliminated without any loss of appropriate detail.
- P.6 The reference to the NRC Policy Statement has been replaced with a more appropriate reference to the Improved Technical Specification "split" criteria found in 10 CFR 50.36(c)(2)(ii).

RAI #7

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Leakage Detection Instrumentation

LCC 3.4.6 The following RCS leakage detection instrumentation shall be OPERABLE:

RAI 510

- a. Drywell floor drain sump flow monitoring system;
- b. The primary containment atmosphere gaseous radioactivity monitoring system; and
- c. Drywell floor drain sump level monitoring system.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell floor drain sump flow monitoring system inoperable.	A.1 Restore drywell floor drain sump flow monitoring system to OPERABLE status.	30 days
B. Required primary containment atmosphere gaseous radioactivity monitoring system inoperable.	B.1 Analyze grab samples of primary containment atmosphere.	Once per 24 hours

(continued)

ACTIONS (continued)

RAI #12

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. Drywell floor drain sump level monitoring system inoperable.</p>	<p>C.1 -----NOTE----- Not applicable when primary containment atmosphere gaseous radioactivity monitoring system is inoperable. ----- Perform SR 3.4.6.1.</p>	<p>Once per 8 hours</p>
<p>D. Primary containment atmosphere gaseous radioactivity monitoring system inoperable.</p> <p><u>AND</u></p> <p>Drywell floor drain sump level monitoring system inoperable.</p>	<p>D.1 Restore primary containment atmosphere gaseous radioactivity monitoring system to OPERABLE status.</p> <p><u>OR</u></p> <p>D.2 Restore drywell floor drain sump level monitoring system to OPERABLE status.</p>	<p>30 days</p> <p>30 days</p>
<p>E. Required Action and associated Completion Time of Condition A, B, C, or D not met.</p>	<p>E.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>E.2 Be in MODE 4.</p>	<p>12 hours</p> <p>36 hours</p>
<p>F. All required leakage detection systems inoperable.</p>	<p>F.1 Enter LCO 3.0.3.</p>	<p>Immediately</p>

BASES

---

ACTIONS (continued)

Action is to restore either of the inoperable monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a degraded configuration for a lengthy time period.

E.1 and E.2

If any Required Action of Condition A, B, C, or D cannot be 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 3 within 12 hours and MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to perform the actions in an orderly manner and without challenging plant systems.

F.1

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

---

SURVEILLANCE  
REQUIREMENTS

SR 3.4.6.1

This SR is for the performance of a CHANNEL CHECK of the required primary containment atmosphere gaseous radioactivity monitoring system. 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 3.4.6.2

This SR is for the performance of a CHANNEL FUNCTIONAL TEST of the required RCS leakage detection instrumentation. The test ensures that the monitors can perform their function in the desired manner. The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

(A) |



DISCUSSION OF CHANGES

ITS: SECTION 3.4.6 - RCS LEAKAGE DETECTION INSTRUMENTATION

NUREG-1433. The information moved to the Bases requires changes to be controlled in accordance with the ITS 5.5.10, Bases Control Program. This relocation continues to provide adequate protection of the public health and safety since the requirement for RCS leakage detection system Operability continues to be required by the Technical Specifications.

- LA.2 CTS 3.4.3.1.b.2 requires operational requirements for "The drywell equipment drain sump level, flow and pump-run-time system," which is utilized for monitoring identified leakage rates. ITS 3.4.6 contains only requirements related to meeting Regulatory Guide 1.45 for unidentified leakage. Based on comparison to the Technical Specification "split" criteria of 10 CFR 50.36(c)(2)(ii) the drywell equipment drain system will be relocated to the Technical Requirements Manual (TRM), which require revisions to be controlled by 10 CFR 50.59. The drywell equipment drain sump level, flow and pump-run-time system is not a part of a primary success path in the mitigation of a DBA or transient, and is a non-significant risk contributor to core damage frequency and offsite releases. This relocation continue to provide adequate protection of the public health and safety since the ITS retain sufficient requirements related to controlling the magnitude of the identified leakage rate.

RA1 0.0-1

TECHNICAL CHANGES - LESS RESTRICTIVE  
"Specific"

- L.1 CTS 3.4.3.1 Action requires two leakage detection systems to remain Operable when allowing a 30-day restoration time. ITS Action statements are provided which allow unlimited continued operation with either required primary containment atmosphere gaseous radioactivity monitor or drywell floor drain sump level monitoring system inoperable. In either case, the primary system for identifying and quantifying unidentified leakage in the containment (i.e., the drywell floor drain sump flow monitoring system) remains OPERABLE. Since the inoperable system(s) provide actions for more frequent leakage measurements (ITS 3.4.6 Required Actions B.1 and/or C.1), operation is allowed to continue with the system(s) inoperable. However, with both backup systems inoperable, this continued operation is limited to 30 days since the diversity is significantly reduced.

(A)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Leakage Detection Instrumentation

<CTS>

LCO 3.4.6 The following RCS leakage detection instrumentation shall be OPERABLE:

- a. Drywell floor drain sump <sup>flow</sup> monitoring system; ~~and~~ <3.4.3.1.b>
- b. ~~One channel of either~~ <sup>The</sup> primary containment ~~atmospheric~~ <sup>radioactivity</sup> particulate or atmospheric gaseous monitoring system; <3.4.3.1.a>
- c. ~~Primary containment air cooler condensate flow rate monitoring system~~ <3.4.3.1.c>

RAI#10

Drywell floor drain sump level monitoring system.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell floor drain sump monitoring system inoperable. <sup>flow</sup> (P.1)	NOTE LCO 3.0.4 is not applicable. A.1 Restore drywell floor drain sump monitoring system to OPERABLE status. <sup>flow</sup>	<3.4.3.1 Action> 30 days

(continued)

BASES (continued)

SURVEILLANCE  
REQUIREMENTS

SR 3.4.6.1

p.2 e gaseous radioactivity

This SR is for the performance of a CHANNEL CHECK of the required primary containment atmospheric monitoring system. 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 3.4.6.2

p.2

This SR is for the performance of a CHANNEL FUNCTIONAL TEST of the required RCS leakage detection instrumentation. The test ensures that the monitors can perform their function in the desired manner. The test also verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

A

SR 3.4.6.3

This SR is for the performance of a CHANNEL CALIBRATION of required leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of ~~18~~ months is a typical refueling cycle and considers channel reliability. Operating experience has proven this Frequency is acceptable.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. UFSAR, Section ~~5.2.7.2.1~~ <sup>1.3</sup>.
4. GEAP-5620, April 1968.
5. NUREG-75/067, October 1975.
6. UFSAR, Section ~~[5.2.7.5.2]~~ 5.2.7.4.3.3

JUSTIFICATION FOR DIFFERENCES FROM NUREG - 1433  
ITS: SECTION 3.4.6 - RCS LEAKAGE DETECTION INSTRUMENTATION

NON-BRACKETED PLANT SPECIFIC CHANGES

- P.1 These changes are made to NUREG-1433 to reflect Fermi 2 current licensing basis; including design features, existing license requirements and commitments. Additional rewording, reformatting, and revised numbering is made to incorporate these changes consistent with Writer's Guide conventions. Refer to CTS Discussion Of Changes to the related requirement for a detailed justification of changes made to the current licensing basis which are also reflected in the ITS as presented. Some of these changes are specifically discussed below:
- a. The ISTS bracketed options reflect designs that include leakage detection from containment air cooler condensate flow rate. Fermi-2 design does not include containment air cooler condensate flow rate as leakage detection instrumentation. Revisions reflect the ISTS intended Actions for removal of this option.
- P.2 Bases changes are made to reflect plant specific design details, equipment terminology, and analyses. Some of these changes are specifically discussed below:
- a. The CHANNEL FUNCTIONAL TEST (ITS SR 3.4.6.2) ensures that the channel performs its intended function, but does not verify the setpoint of all alarms, timers, and switches. These setpoints would be the subject of the CHANNEL CALIBRATION. As such, the Bases sentence referencing verification of the alarm setpoint and string accuracy is deleted. This reflects the existing Fermi-2 implementation of the CHANNEL FUNCTIONAL TEST and is consistent with the Bases description for other instrument channels. (A)
- P.3 Bases changes are made to reflect changes made to the Specification. Refer to the Specification, and associated JFD if applicable, for additional detail.
- P.4 The reference to the NRC Policy Statement has been replaced with a more appropriate reference to the Improved Technical Specification "split" criteria found in 10 CFR 50.36(c)(2)(ii).

GENERIC CHANGES

- C.1 TSTF-60: NRC approved change to NUREG-1433.

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

- c. In OPERATIONAL CONDITION 1 or 2, with:
1. THERMAL POWER changed by more than 15% of RATED THERMAL POWER in 1 hour\*, or
  2. The off-gas level, at the delay pipe, increased by more than 10,000 microcuries per second in 1 hour during steady-state operation at release rates less than 75,000 microcuries per second, or
  3. The off-gas level, at the delay pipe, increased by more than 15% in one hour during steady-state operation at release rates greater than 75,000 microcuries per second,
- perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit.

L.4

RAI #15

SURVEILLANCE REQUIREMENTS

SR  
3.4.7.1

4.4.5 The specific activity of the reactor coolant shall be demonstrated to be within the limits by performance of the sampling and analysis program of Table 4.4.5-1.

~~\*Not applicable during the start-up test program.~~

L.4

RAI #15

TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	OPERATIONAL CONDITIONS IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. <del>Gross Beta and Gamma Activity Determination</del>	<del>At least once per 72 hours</del>	<del>1, 2, 3</del> L.1
Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	At least once per 31 days	M.1 L.1
3. Radiochemical for I-Determination	At least once per 6 months*	1, 2, 3** L.1
4. Isotopic Analysis for Iodine	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION b.	1** L.2
Required Action A.1 and B.1		
5. Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for at least Xe-133, Xe-135 and Kr-88	b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION c.	1, 2 L.4
	At least once per 31 days	A.3 L.1

Specification 3.4.7  
(Also See Specification 3.7.5)

FERMI - UNIT 2

3/4 4-18

Rev 4

SR 3.4.7.1 →

See Specification 3.7.5

\*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.

\*\*Until the specific activity of the primary coolant system is restored to within its limits.

Required Action A.1 and B.1

DISCUSSION OF CHANGES  
ITS: SECTION 3.4.7 - RCS SPECIFIC ACTIVITY

ADMINISTRATIVE

- A.1 In the conversion of the Fermi 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Boiling Water Reactor (BWR) Standard Technical Specifications NUREG-1433, Rev. 1.
- A.2 (Not used.)
- A.3 CTS Table 4.4.5-1 Item 5, requires "Isotopic Analysis of an Offgas Sample..." ITS 3.4.7 does not retain this requirement. This is acceptable because the CTS requirement is the same as ITS SR 3.7.5.1 which requires the same sample to be taken every 31 days. Therefore, this is an administrative change with no impact on safety because it only eliminates a duplicate requirement contained in CTS.

RAI #15

TECHNICAL CHANGES - MORE RESTRICTIVE

- M.1 CTS Table 4.4.5-1 Item 2 requires that isotopic analysis for Dose Equivalent I-131 concentration be made at least once per 31 days. ITS SR 3.4.7.1 requires that this sample be taken once every 7 days. The increased surveillance frequency provides a compensatory measure for the removal of the requirement that gross specific activity remain less than or equal to 100/E-bar  $\mu\text{Ci}/\text{gram}$ . This more restrictive change will have no negative impact on safety, based on the fact that the increased sampling will provide an earlier detection of a degraded condition.

DISCUSSION OF CHANGES  
ITS: SECTION 3.4.7 - RCS SPECIFIC ACTIVITY

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Specific"

- L.1 CTS 3.4.5 requirement to maintain specific activity  $\leq 100/E\text{-bar}$   $\mu\text{Ci/gm}$  and Surveillance for gross Beta and Gamma activity, as well as the radiochemical analysis for E-bar (Table 4.4.5-1, Items 1 and 3) have been deleted. The CTS Bases state that the intent of the requirement to limit the specific activity of the reactor coolant is to ensure that whole body and thyroid doses at the site boundary would not exceed a small fraction of the limits stated in 10 CFR 100 (i.e., 10% of 25 rem and 300 rem, respectively) in the event of a main steam line failure outside containment or an instrument line break. To ensure that offsite thyroid doses do not exceed 30 rem, reactor coolant dose equivalent I-131 (DEI) is limited to less than or equal to  $0.2 \mu\text{Ci/gm}$ . CTS also limits reactor coolant gross specific activity to less than or equal to  $100/E\text{-bar}$   $\mu\text{Ci/gm}$  to ensure that whole body doses do not exceed 2.5 rem. CTS LCO 3.11.2.7 (ITS LCO 3.7.5) associated with radioactive effluents requires that the gross gamma radioactivity rate of the noble gases Xe-133, Xe-135, Xe-138, Kr-85m, Kr-87, and Kr-88 measured at the main condenser evacuation system pretreatment monitor station be limited to less than or equal to 340 mCi/second. The CTS Bases for LCO 3.11.2.7 state that restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total-body exposure to an individual at the exclusion area boundary will not exceed a small fraction (10%) of the limits of 10 CFR 100 in the event this effluent is inadvertently discharged without treatment directly to the environment.

The Main Condenser Offgas Treatment System, as required by CTS LCO 3.11.2.7 (ITS LCO 3.7.5) provides reasonable assurance the reactor coolant gross specific activity is maintained at a sufficiently low level to preclude offsite doses from exceeding a small fraction of the limits of 10 CFR 100 in the event of a main steam line failure. Therefore, CTS LCO 3.4.5.b is redundant and places an unnecessary burden on the licensee without a commensurate

RAI #17



DISCUSSION OF CHANGES  
ITS: SECTION 3.4.7 - RCS SPECIFIC ACTIVITY

increase in safety. Elimination of CTS LCO 3.4.5.b and Surveillance Items 1 and 3 of Table 4.4.5-1 will allow plant personnel to focus attention on efficient, safe operation of the plant without the unnecessary distraction of the redundant Surveillance Requirement. Additional assurance that the offsite doses will not exceed a small fraction of the 10 CFR 100 limits is provided by increasing the frequency of sampling and analysis of the reactor coolant for DEI from at least once per 31 days to at least once per 7 days.

RAI #17

Since (1) the reactor coolant limit on DEI adequately assures that offsite doses will not exceed small fractions of the limits of 10 CFR 100 in the event of a main steam line failure outside containment and (2) gross gamma radioactivity rate of the noble gases measured at the condenser evacuation system pretreatment monitor station is limited by ITS 3.7.5 to a value that provides reasonable assurance the reactor coolant gross specific activity is maintained at a sufficiently low level to preclude offsite doses from exceeding a small fraction of the limits of 10 CFR 100, the requirements associated with LCO 3.4.5.b and Surveillance Items 1 and 3 of Table 4.4.5-1 are unnecessary.

RAI #17

The associated Actions and Surveillance Requirements are also being deleted, consistent with the LCO requirement deletion.

L.2

CTS 3.4.5 Applicability includes Operational Condition 2, 3, and 4. ITS 3.4.7 Applicability is limited to only those conditions which represent a potential for release of significant quantities of radioactive coolant to the environment. Mode 4 is omitted since the reactor is not pressurized and the potential for leakage is significantly reduced. In Modes 2 and 3, with the main steam lines isolated, no escape path exists for significant releases and requirements for limiting the specific activity are not required. The Required Actions are also modified to reflect the new Applicability, and an option for exiting the applicable Modes is provided for cases where isolation is not desired. Based on the fact that the ITS Applicability is consistent with plant conditions where event consequences are significant, this less restrictive change will have a negligible impact on safety.

DISCUSSION OF CHANGES  
ITS: SECTION 3.4.7 - RCS SPECIFIC ACTIVITY

- L.3 CTS 3.4.5 Actions do not allow Mode changes while Dose Equivalent I-131 is not within limit (CTS 3.0.4). ITS 3.4.7 Action A provides a Note to the Required Actions to indicate that LCO 3.0.4 is not applicable. Entry into the applicable Modes should not be restricted due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is 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. Therefore, operation during the allowed time frame would not represent a significant impact to the health and safety of the public. This less restrictive change will have a negligible impact on safety.
- L.4 CTS 3.4.5 Action c requires increased sampling of Item 4b of CTS Table 4.4.5-1. CTS 3.4.5 Action c associates this sampling with changing plant conditions. However, as required by CTS 3.0.1, this Action applies only when the LCO is not met. Similarly, CTS 3.4.5 Action b requires the sampling of Item 4a of Table 4.4.5-1 when the LCO is not met. Since sampling of Item 4a provides sampling requirements equivalent to Item 4b, the Action c requirement (and associated Item 4b) are unnecessary. Therefore, elimination of the duplicative requirements of CTS Action c (including associated footnote \*) involves no impact on safety.

RAI #15 #16

RELOCATED SPECIFICATIONS

None

TECHNICAL SPECIFICATION BASES

The CTS Bases for this Specification have been replaced by Bases that reflect the format and applicable content of ITS 3.4.7 consistent with the BWR STS, NUREG-1433, Rev. 1.

NO SIGNIFICANT HAZARDS EVALUATION  
ITS: SECTION 3.4.7 - RCS SPECIFIC ACTIVITY

TECHNICAL CHANGES - LESS RESTRICTIVE  
(Specification 3.4.7 "L.4" Labeled Comments / Discussions)

Detroit Edison has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria specified by 10 CFR 50.92 and has determined that the proposed change does not involve a significant hazards consideration.

The bases for the determination that the proposed change does not involve a significant hazards consideration is an evaluation of these changes against each of the criteria in 10 CFR 50.92. The criteria and the conclusions of the evaluation are presented below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change eliminates the requirements for increased sampling of isotopic analysis for iodine (which is stated in terms of changing plant conditions; but required only by CTS 3.4.5 Action c and thus applicable only when the associated LCO is not met). Equivalent sampling requirements, also applicable when the LCO is not met, are included in CTS Action b. The removal of an action covered by an equivalent action is not considered as an initiator of any previously evaluated accident, nor will it have any impact on the consequences of any previously evaluated accident. Therefore, the proposed change will not increase the probability of any accident previously evaluated and will not increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not introduce a mode of plant operation, not previously analyzed, and does not involve a physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

This change does not involve a significant reduction in a margin of safety since no limitation, operating parameter, or analysis assumption is modified. Therefore, no question of safety is involved and these changes do not involve any reduction in a margin of safety.

RAI #15

DISCUSSION OF CHANGES

ITS: SECTION 3.4.8 - RHR SHUTDOWN COOLING SYSTEM - HOT SHUTDOWN

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 CTS 3.4.9.1 LCO details system design. ITS 3.4.8 requires that RHR SDC be Operable, but does not detail the system design. This is acceptable because the system design does not impact the requirement to have the RHR SDC system Operable. Therefore, these details can be relocated to the Bases. This change is consistent with NUREG-1433. The information moved to the Bases requires changes to be controlled in accordance with the ITS 5.5.10, Bases Control Program. This relocation continues to provide adequate protection of the public health and safety since the requirement for RHR SDC Operability continues to be required by the Technical Specifications.

LR.1 CTS SR 4.4.9.1.2 requires that "At least once per 12 hours verify the required RHR shutdown cooling mode loop(s) are capable of taking suction from the reactor vessel through the RHR heat exchanger(s) with their associated cooling water available." This is essentially equivalent to "verify the required RHR subsystem is Operable." ISTS surveillances prescribe specific acceptance criteria or require verification of a specific feature; but ISTS surveillances do not require non-specific verification of system Operability without specifying details for this verification. Tracking the status of system Operability is an ongoing activity. Therefore, the requirements of CTS SR 4.4.9.1.2 are removed from Technical Specifications, consistent with the NUREG-1433. Regulatory control of changes to these requirements (e.g., Technical Specification amendment or 10 CFR 50.59) is not necessary to provide adequate protection of the public health and safety since the requirement for RHR Shutdown Cooling system Operability continues to be required by the Technical Specifications.

RAI # 3.4-1

## DISCUSSION OF CHANGES

### ITS: SECTION 3.4.9 - RHR SHUTDOWN COOLING SYSTEM - COLD SHUTDOWN

#### ADMINISTRATIVE

- A.1 In the conversion of the Fermi 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Boiling Water Reactor (BWR) Standard Technical Specifications NUREG-1433, Rev. 1.
- A.2 ITS 3.4.9 adds a Note to the Actions that "Separate Condition entry is allowed for each RHR shutdown cooling subsystem." This Note provides explicit instructions for proper application of the Actions for Technical Specification compliance. In conjunction with the ITS 1.3 - "Completion Times," this Note provides direction consistent with the intent of the CTS Actions. Therefore, this is an administrative change with no impact on safety.
- A.3 CTS 3.4.9.2 Applicability is stated as Operational Condition 4 (i.e., Cold Shutdown), but is further modified with "irradiated fuel is in the reactor vessel and the water level is less than 20 feet 6 inches above the top of the reactor pressure vessel flange..." This additional modifier is eliminated with no resultant change in requirement or interpretation. The Cold Shutdown condition is defined such that there is fuel in the vessel and the reactor vessel head is bolted in place. With the head on, the reactor cavity is not flooded and water level is not (and could not be) 20 feet 6 inches above the vessel flange. Therefore, eliminating this explicit modifier will not change the requirements, and is considered administrative.

RAI #19

#### TECHNICAL CHANGES - MORE RESTRICTIVE

- M.1 CTS 3.4.9.2 Action d requires "The provisions of Specification 3.0.4 are not applicable for up to 4 hours for the purpose of establishing the RHR system in the shutdown cooling mode once the reactor vessel pressure is less than the RHR cut-in permissive setpoint." Since no benefit or rationale could be determined for this Action, it has been deleted.

DISCUSSION OF CHANGES  
ITS: SECTION 3.4.9 - RHR SHUTDOWN COOLING SYSTEM - COLD SHUTDOWN

TECHNICAL CHANGES - LESS RESTRICTIVE  
"Generic"

- LA.1 CTS LCO 3.4.9.2 details the system design. ITS 3.4.9 requires the RHR Shutdown Cooling System to be Operable, but does not provide details as to the system design. This is acceptable because the system design details do not impact the requirement to maintain the RHR Shutdown Cooling System Operable. This change is consistent with NUREG-1433. The information is being moved to the Bases, which requires changes to be controlled in accordance with the ITS 5.5.10, Bases Control Program. This relocation continues to provide adequate protection of the public health and safety since RHR Shutdown Cooling System Operability continues to be required by the Technical Specifications.
- LA.2 CTS 3.4.9.2 and 4.4.9.2.2 during Mode 4 operation with the reactor cavity not completely flooded requires reactor water level to be maintained  $\geq 214$  inches in addition to a means of forced circulation. CTS 3.9.11.2 Action c requires two means of forced circulation from either two recirculation pumps or two RHR-SDC loops in the event level is not maintained  $\geq 214$  inches. ITS 3.4.9 does not retain these restrictions -- only requiring one means of forced circulation regardless of reactor water level. These will be relocated to the Technical Requirements Manual (TRM). This is consistent with the NUREG-1433. This water level requirement can be adequately defined and controlled in the TRM, which requires revisions to be controlled by 10 CFR 50.59. This relocation continues to provide adequate protection of the public health and safety since the requirement for forced circulation, and alternate means of reactor coolant circulation on loss of forced circulation, continues to be required by the Technical Specifications.

RAI 0.0-1

DISCUSSION OF CHANGES

ITS: SECTION 3.4.9 - RHR SHUTDOWN COOLING SYSTEM - COLD SHUTDOWN

LR.1 CTS SR 4.4.9.2.3 requires that "At least once per 12 hours verify the required RHR shutdown cooling mode loop(s) are capable of taking suction from the reactor vessel through the RHR heat exchanger(s) with their associated cooling water available." This is essentially equivalent to "verify the required RHR subsystem is Operable." ISTS surveillances prescribe specific acceptance criteria or require verification of a specific feature; but ISTS surveillances do not require non-specific verification of system Operability without specifying details for this verification. Tracking the status of system Operability is an ongoing activity. Therefore, the requirements of CTS SR 4.4.9.1.2 are removed from Technical Specifications, consistent with the NUREG-1433. Regulatory control of changes to these requirements (e.g., Technical Specification amendment or 10 CFR 50.59) is not necessary to provide adequate protection of the public health and safety since the requirement for RHR Shutdown Cooling system Operability continues to be required by the Technical Specifications.

RAY# 3.4-1

TECHNICAL CHANGES - LESS RESTRICTIVE  
"Specific"

None

RELOCATED SPECIFICATIONS

None

TECHNICAL SPECIFICATION BASES

The CTS Bases for this Specification have been replaced by Bases that reflect the format and applicable content of ITS 3.4.9 consistent with the BWR STS, NUREG-1433, Rev. 1.





REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

(A.1)

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.4.10

SR 3.4.10.1.a

SR 3.4.10.2

~~3.4.6.1~~ The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 (1) Curve A for hydrostatic or leak testing; (2) Curve B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) Curve C for operations with a critical core other than low power PHYSICS TESTS, with:

SR 3.4.10.1.b.1

SR 3.4.10.1.b.2

SR 3.4.10.7

- a. A maximum heatup of 100°F in any 1-hour period,
- b. A maximum cooldown of 100°F in any 1-hour period,
- c. A maximum temperature change of less than or equal to 20°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and
- d. The reactor vessel flange and head flange temperature greater than or equal to 71°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

REQUIRED ACTION  
A.1 & C.1

REQUIRED ACTION  
A.2 & C.2

ACTION B

With any of the above limits exceeded, 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; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

ADD ACTION A NOTE  
ACTION C NOTE

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

~~4.4.6.1.1~~ During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figure 3.4.6.1-1 Curves A, B, or C, as applicable, at least once per 30 minutes.

RAI # 25

RAI # 25

DISCUSSION OF CHANGES  
ITS: SECTION 3.4.10 - RCS P/T LIMITS

ADMINISTRATIVE

- A.1 In the conversion of the Fermi 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Boiling Water Reactor (BWR) Standard Technical Specifications NUREG-1433, Rev. 1.
- A.2 CTS 3.4.1.4.a, 3.4.1.4.b, and 4.4.1.1.4.b and c address a 50°F limitation between the idle loop and either the "reactor coolant" or an "operating recirculation loop." ITS SR 3.4.10.4 and SR 3.4.10.6 presents these limitations together with the idle loop compared to "RPV coolant temperature." Since the operating loop temperature is considered synonymous with RPV coolant temperature, this change is considered an editorial presentation preference. As such, the stated requirements for "both loops have been idle" (CTS 3.4.1.4.a) and for "only one loop has been idle" (CTS 3.4.1.4.b) will be the same and separate discussion is no longer necessary. These will be combined in ITS SR 3.4.10.4.
- A.3 CTS 3.4.1.4 Action requires "suspend startup of any idle recirculation loop" if any required differential temperature is not within limit. ITS SRs 3.4.10.3 and 3.4.10.4 present the same required differential temperatures, however, with the requirement that SRs be met (ITS SR 3.0.1) the action to preclude startup with unacceptable temperatures is inherent without an explicit action. Therefore, elimination of this CTS Action is an administrative presentation preference.
- A.4 CTS 3.4.6.1 Action for any out of limits condition requires "restore...within 30 minutes." ITS 3.4.10 Required Action C.1 for Modes 4 and 5 states "initiate action to restore ... Immediately." The CTS Action provides 30 minutes in which pressure and temperature requirements could exceed the limits, and if the parameters are incapable of being restored to within the limits within 30 minutes, provides no further action. The intent of the action is more appropriately presented in ITS Required Action C.1, which provides for continuous Actions during all out-of-limits conditions in Mode 4 and 5. As an enhanced presentation of the CTS intent, the change is deemed to be administrative.

RAI #23

(A)

DISCUSSION OF CHANGES  
ITS: SECTION 3.4.10 - RCS P/T LIMITS

- A.5 (Not used)
- A.6 CTS 4.4.6.1.4.b requirement to verify the vessel flange and head flange temperature within 30 minutes "prior to" tensioning of the head bolting studs has been deleted. This requirement is a duplication of CTS 4.0.4 and ITS SR 3.0.4, which require the Surveillances to be current prior to entering the condition in which the SR is applicable. Therefore, this change is an administrative change with no impact on safety.

RAI #25

TECHNICAL CHANGES - MORE RESTRICTIVE

- M.1 CTS 3.4.6.1 Action requires an evaluation to determine that the RCS remains acceptable for continued operation. ITS imposes a specific limitation for the engineering evaluation with a specific completion time. The Completion Time of 72 hours is considered reasonable for operation in Modes 1, 2, and 3 (Required Action A.2) because the P/T limits represent long term vessel fatigue and usage factors. In Modes 4 and 5, the Completion Time of "Prior to entering MODE 2 or 3" (Required Action C.2) specifically prevents entry into operating Modes. This is consistent with the CTS intent in applying LCO 3.0.4. The open-ended time limit on completing the evaluation when the unit is shutdown is a reasonable interpretation of the CTS requirement, since it is not expected that a P/T limit violation while shutdown would present an immediate threat to the RCS integrity. As a specific limitation consistent with the NUREG, and an enhanced presentation of the CTS intent, the change will not introduce any adverse impact on safety.

RAI #25

Additionally, the CTS Actions are interpreted to imply that the required evaluation must be completed regardless of the restoration. ITS Notes to Condition A and Condition C provide explicit clarification of this intent. As an enhanced presentation of the CTS intent, the change is deemed to be administrative (and discussed here for completeness).

DISCUSSION OF CHANGES  
ITS: SECTION 3.4.10 - RCS P/T LIMITS

TECHNICAL CHANGES - LESS RESTRICTIVE  
"Generic"

- LA.1 CTS 4.4.6.1.3 and Table 4.4.6.1.3-1 details the schedule for removal of the reactor vessel material surveillance specimens, and states the function of the examination results is to update P/T curves. ITS does not retain these details. The requirement for an NRC approved withdrawal schedule and for use of the specimen results are contained in 10 CFR 50, Appendix H, and therefore, are not necessary to be repeated in the ITS to provide adequate protection of the public health and safety.
- LA.2 CTS 3.4.6.1 Actions state the requirement to "perform an engineering evaluation to determine..." ITS 3.4.10 Required Actions A.2 and C.2 state "determine." The understanding that the determination is made by performance of an engineering evaluation is relocated to the Bases, which requires changes to be controlled in accordance with the ITS 5.5.10, Bases Control Program. This relocation continues to provide adequate protection of the public health and safety since the requirement to "determine" continues to be required by the Technical Specifications.
- LR.1 Not used.

RAI #25

INSERT 3.4.10 - 1

SURVEILLANCE REQUIREMENTS	
SURVEILLANCE	FREQUENCY
SR 3.4.10.1 .. .. ..  Verify:  a. RCS pressure and RCS temperature are to the right of the limits specified in Figure 3.4.10-1; and  b. RCS heatup and cooldown rates are limited to:  1. $\leq 100^{\circ}\text{F}$ in any 1 hour period; and 2. $\leq 20^{\circ}\text{F}$ in any 1 hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.	 RAI # 26

# Specification 3.4.11

## REACTOR COOLANT SYSTEM

A.1

## REACTOR STEAM DOME

### LIMITING CONDITION FOR OPERATION

LCO  
3.4.11

~~3.4.6.2~~ The pressure in the reactor steam dome shall be less than <sup>or equal to</sup> 1045 psig.

L.1

RAI #27

APPLICABILITY: OPERATIONAL CONDITIONS 1<sup>st</sup> and 2<sup>nd</sup>.

M.1

### ACTION:

Action A With the reactor steam dome pressure exceeding 1045 psig, reduce the pressure to less than 1045 psig within 15 minutes <sup>or</sup> be in at least HOT SHUTDOWN within 12 hours.

Action B

### SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

~~3.4.6.2~~ The reactor steam dome pressure shall be verified to be less than 1045 psig at least once per 12 hours.

or equal to

L.1

RAI #27

\*Nnt applicable during anticipated transients.

M.1

DISCUSSION OF CHANGES  
ITS: SECTION 3.4.11 - REACTOR STEAM DOME PRESSURE

ADMINISTRATIVE

- A.1 In the conversion of the Fermi 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Boiling Water Reactor (BWR) Standard Technical Specifications NUREG-1433, Rev. 1.
- A.2 Not used.

RAY #27

TECHNICAL CHANGES - MORE RESTRICTIVE

- M.1 CTS 3.4.6.2 Applicability footnote identifying that the reactor steam dome pressure limit is not applicable during anticipated transients is deleted. ITS 3.4.11 is constructed with the reactor steam dome pressure limit as an operational limit. It is assumed that anticipated transients will result in exceeding operational limits. It is further assumed that action (either automatic or operator, or both) is initiated to return the plant to acceptable steady state operation. No benefit or rationale is seen for the CTS exception. Therefore, the reactor steam dome pressure limit is applicable during such transients, and the Required Actions should be taken as part of the mitigation of the transient. This more restrictive change will have no negative impact on safety, since it eliminates a CTS exception which could be misinterpreted.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

None

DISCUSSION OF CHANGES  
ITS: SECTION 3.4.11 - REACTOR STEAM DOME PRESSURE

TECHNICAL CHANGES - LESS RESTRICTIVE  
"Specific"

- L.1 CTS 3.4.6.2 and 4.4.6.2 requires the reactor steam dome pressure to be "less than 1045 psig," however, the associated Action applies with reactor steam dome pressure "exceeding 1045 psig." The requirement associated with pressure exactly equal to 1045 psig is not specifically addressed. ITS 3.4.11 limit on reactor steam dome pressure is specified as less than or equal to 1045 psig, which agrees with the Fermi-2 vessel overpressurization analyses. Since the difference is infinitesimal, and resolves an obvious discontinuity consistent with the CTS Action presentation, the change will not result in any significant impact on safety.

RELOCATED SPECIFICATIONS

None

TECHNICAL SPECIFICATION BASES

The CTS Bases for this Specification have been replaced by Bases that reflect the format and applicable content of ITS 3.4.11 consistent with the BWR STS, NUREG-1433, Rev. 1.



NO SIGNIFICANT HAZARDS EVALUATION  
ITS: SECTION 3.4.11 - REACTOR STEAM DOME PRESSURE

TECHNICAL CHANGES - LESS RESTRICTIVE  
(Specification 3.4.11 "L.1" Labeled Comments / Discussions)

Detroit Edison has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria specified by 10 CFR 50.92 and has determined that the proposed change does not involve a significant hazards consideration.

The bases for the determination that the proposed change does not involve a significant hazards consideration is an evaluation of these changes against each of the criteria in 10 CFR 50.92. The criteria and the conclusions of the evaluation are presented below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed change resolves an obvious discontinuity between the LCO statement and its surveillance, and the associated Action statement. The operating limit on reactor steam dome pressure is not considered as an initiator for any previously evaluated accident. Therefore, the proposed change will not increase the probability of any accident previously evaluated. The difference is infinitesimal, and resolves an obvious discontinuity in a manner consistent with the Fermi-2 vessel overpressurization analyses and the CTS Action presentation. Therefore, the change will not significantly increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

This proposed change will not involve any physical changes to plant systems, structures, or components (SSC), or measurable changes in normal plant operation. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

NO SIGNIFICANT HAZARDS EVALUATION  
ITS: SECTION 3.4.11 - REACTOR STEAM DOME PRESSURE

TECHNICAL CHANGES - LESS RESTRICTIVE  
(Specification 3.4.11 "L.1" Labeled Comments / Discussions)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not involve a significant reduction in a margin of safety because the change is consistent with the vessel overpressurization analysis assumptions. Furthermore, the difference is infinitesimal, and resolves an obvious discontinuity in a manner consistent with the CTS Action presentation as well as the Fermi-2 analyses.

**INSERT THIS PAGE IN FRONT OF VOLUME 10**

<b>Volume 10: SECTION 3.9</b>	
<b>Remove</b>	<b>Replace</b>
B 3.9.2 ITS pg B 3.9.2-4 Rev 0	B 3.9.2 ITS pg B 3.9.2-4 Rev 4
B 3.9.2 NUREG M/U pg B 3.9-7	B 3.9.2 NUREG M/U pg B 3.9-7 Rev 4
--	B 3.9.2 ITS pg B 3.9-7 (Insert) Rev 4
3.9.2 JFDs pg 1 Rev 0	3.9.2 JFDs pg 1 Rev 4
3.9.4 DOCs pg 3 Rev 0	3.9.4 DOCs pg 3 Rev 4
3.9.4 NSHC pg 1 Rev 0	3.9.4 NSHC pg 1 Rev 4
B 3.9.5 ITS pg B 3.9.5-2 Rev 0	B 3.9.5 ITS pg B 3.9.5-2 Rev 4
B 3.9.5 NUREG M/U pg B 3.9-17	B 3.9.5 NUREG M/U pg B 3.9-17 Rev 4
3.9.5 JFDs pg 1 Rev 0	3.9.5 JFDs pg 1 Rev 4
B 3.9.6 NUREG M/U pg B 3.9-19	B 3.9.6 NUREG M/U pg B 3.9-19 Rev 4
3.9.7 ITS pg 3.9-10 Rev 0	3.9.7 ITS pg 3.9-10 Rev 4
B 3.9.7 ITS pg B 3.9.7-2 Rev 0	B 3.9.7 ITS pg B 3.9.7-2 Rev 4
3.9.7 CTS M/U (3/4 9-16) pg 1 of 1	3.9.7 CTS M/U (3/4 9-16) pg 1 of 1 Rev 4
3.9.7 NUREG M/U pg 3.9-11	3.9.7 NUREG M/U pg 3.9-11 Rev 4
B 3.9.7 NUREG M/U pg 3.9-26	B 3.9.7 NUREG M/U pg 3.9-26 Rev 4
B 3.9.7 NUREG M/U pg B 3.9-26 (Insert) Rev 0	B 3.9.7 NUREG M/U pg B 3.9-26 (Insert) Rev 4
3.9.8 ITS pg 3.9-12 Rev 0	3.9.8 ITS pg 3.9-12 Rev 4
B 3.9.8 ITS pg B 3.9.8-2 Rev 0	B 3.9.8 ITS pg B 3.9.8-2 Rev 4
3.9.8 CTS M/U (3/4 9-17) pg 1 of 1	3.9.8 CTS M/U (3/4 9-17) pg 1 of 1 Rev 4
3.9.8 DOCs pg 2 Rev 0	3.9.8 DOCs pg 2 Rev 4
3.9.8 NUREG M/U pg 3.9-14	3.9.8 NUREG M/U pg 3.9-14 Rev 4
B 3.9.8 NUREG M/U pg B 3.9-30	B 3.9.8 NUREG M/U pg B 3.9-30 Rev 4
B 3.9.8 NUREG M/U pg B 3.9-30 (Insert) Rev 0	B 3.9.8 NUREG M/U pg B 3.9-30 (Insert) Rev 4

BASES

---

SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.2.2

(B) Performance of a CHANNEL FUNCTIONAL TEST demonstrates the associated refuel position one-rod-out interlock will function properly when a simulated or actual signal indicative of a required condition is injected into the logic. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other 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 CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping, or total channel steps so that the entire channel is tested. The 7 day Frequency is considered adequate because of demonstrated circuit reliability, procedural controls on control rod withdrawals, and visual and audible indications available in the control room to alert the operator to control rods not fully inserted. To perform the required testing, the applicable condition must be entered (i.e., a control rod must be withdrawn from its full-in position). Therefore, SR 3.9.2.2 has been modified by a Note that states the CHANNEL FUNCTIONAL TEST is not required to be performed until 1 hour after any control rod is withdrawn.

---

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. UFSAR, Section 7.6.1.1.
3. UFSAR, Section 15.4.1.1.

BASES

---

ACTIONS

A.1 and A.2 (continued)

fuel assemblies. Action must continue until all such control rods are fully inserted. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and, therefore, do not have to be inserted.

---

SURVEILLANCE  
REQUIREMENTS

SR 3.9.2.1

Proper functioning of the refueling position one-rod-out interlock requires the reactor mode switch to be in Refuel. During control rod withdrawal in MODE 5, improper positioning of the reactor mode switch could, in some instances, allow improper bypassing of required interlocks. Therefore, this Surveillance imposes an additional level of assurance that the refueling position one-rod-out interlock will be OPERABLE when required. By "locking" the reactor mode switch in the proper position (i.e., removing the reactor mode switch key from the console while the reactor mode switch is positioned in refuel), an additional administrative control is in place to preclude operator errors from resulting in unanalyzed operation.

The Frequency of 12 hours is sufficient in view of other administrative controls utilized during refueling operations to ensure safe operation.

SR 3.9.2.2

(P.2) Performance of a CHANNEL FUNCTIONAL TEST ~~on each channel~~ demonstrates the associated refuel position one-rod-out interlock will function properly when a simulated or actual signal indicative of a required condition is injected into the logic. The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping, or total channel steps so that the entire channel is tested. The 7 day Frequency is considered adequate because of demonstrated circuit reliability, procedural controls on control rod withdrawals, and visual and audible indications available in the control room to alert the operator to control rods not fully inserted. To perform the required testing, the applicable condition must be entered (i.e., a control rod

INSERT  
B 3.9.2-1

(C.1)

(B)

(continued)

INSERT B 3.9.2-1

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

(B)

JUSTIFICATION FOR DIFFERENCES FROM NUREG - 1433  
ITS: SECTION 3.9.2 - REFUEL POSITION ONE-ROD-OUT INTERLOCK

NON-BRACKETED PLANT SPECIFIC CHANGES

- P.1 Not used.
- P.2 Bases changes are made to reflect plant specific design details, equipment terminology, and analyses. Specifically, the Bases description of the one-rod-out interlock as consisting of two channels is revised to reflect the Fermi-2 design. This interlock is not designed as a function with two redundant channels.
- P.3 Not used.
- P.4 The reference to the NRC Policy Statement has been replaced with a more appropriate reference to the Improved Technical Specification "split" criteria found in 10 CFR 50.36(c)(2)(ii).

GENERIC CHANGES

- C.1 TSTF-205: NRC approved change to NUREG-1433.

ⓑ

DISCUSSION OF CHANGES  
ITS: SECTION 3.9.4 - CONTROL ROD POSITION INDICATION

TECHNICAL CHANGES - LESS RESTRICTIVE  
"Specific"

- L.1 CTS 3.1.3.7 LCO and Action b, and CTS 4.1.3.7, require the position indication system to indicate the current position of any withdrawn control rod in Mode 5. ITS 3.9.4 requires only "full-in" position indication capability. This change is acceptable because the Operability of the control rod "full-in" position indication for each control rod (whether the control rod is inserted or withdrawn) is required to support Operability of the refueling interlocks (ITS 3.9.1) and Operability of the one-rod-out interlock (ITS 3.9.2). Requiring only the "full-in" position indication is acceptable because in Mode 5 the safety analysis is based on the assurance that interlocks ensure that either all or all but one control rod is fully inserted. There is no assumption in the safety analysis about rod position other than it is "full-in" or not "full-in." Therefore, only the position indication channel associated with the "full-in" indication is required to be Operable in Mode 5. (A)

ITS 3.9.4 modifies Surveillance Requirements to be consistent with the requirement that only the full-in indicator must be Operable. The new Surveillance (ITS SR 3.9.4.1) requires that each time a control rod is withdrawn from the full-in position, the full-in indication is indicating correctly (i.e., it is not indicating full-in when a control rod is withdrawn). (Note that failure to indicate "full-in" when the control rod is fully inserted results in conservative actuation of the one-rod-out and refueling interlocks, and therefore, is not explicitly required to be verified.) Additionally, the current requirements to verify the position of the control rod every 24 hours (CTS 4.1.3.7.a), that the control rod position changes during exercise tests (CTS 4.1.3.7.b), and that the full-out indicator functions during rod coupling checks (CTS 4.1.3.7.c), are deleted because these tests do not support Operability of the "full-in" position indication channel. These less restrictive changes have no impact on safety.

RELOCATED SPECIFICATIONS

None



NO SIGNIFICANT HAZARDS EVALUATION  
ITS: SECTION 3.9.4 - CONTROL ROD POSITION INDICATION

TECHNICAL CHANGES - LESS RESTRICTIVE  
(Specification 3.9.4 "L.1" Labeled Comments / Discussions)

Detroit Edison has evaluated the proposed Technical Specification change identified as "Less Restrictive" in accordance with the criteria specified by 10 CFR 50.92 and has determined that the proposed change does not involve a significant hazards consideration.

The bases for the determination that the proposed change does not involve a significant hazards consideration is an evaluation of these changes against each of the criteria in 10 CFR 50.92. The criteria and the conclusions of the evaluation are presented below.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

This change replaces a Mode 5 requirement that the position indication system indicate the current position of any withdrawn control rod with a requirement that only the control rod "full-in" position indication is Operable for all control rods including those not withdrawn. This change will not result in an increase in the probability or consequences of an accident previously evaluated because the Operability of the control rod "full-in" position indication for each control rod (whether the control rod is inserted or withdrawn) is sufficient to support Operability of the refueling interlocks and Operability of the one-rod-out interlock. Requiring Operability of the "full-in" position indication only is acceptable because in Mode 5 the safety analysis is based on the assurance that interlocks ensure that either all or all but one control rod is fully inserted. There is no assumption in the safety analysis about rod position other than it is "full-in" or not "full-in." Therefore, only the position indication channel associated with the "full-in" indication is required to be Operable in Mode 5. (A)

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

This proposed change will not involve any physical changes to plant systems, structures, or components (SSC), or changes in normal plant operation. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

BASES

---

LCO Each withdrawn control rod must be OPERABLE. The withdrawn control rod is considered OPERABLE if the scram accumulator pressure is  $\geq 940$  psig and the control rod is capable of being automatically inserted upon receipt of a scram signal; however, no specific scram time limit is imposed. Inserted control rods have already completed their reactivity control function, and therefore are not required to be OPERABLE.

---

APPLICABILITY During MODE 5, withdrawn control rods must be OPERABLE to ensure that in a scram the control rods will insert and provide the required negative reactivity to maintain the reactor subcritical.

For MODES 1 and 2, control rod requirements are found in LCO 3.1.2, "Reactivity Anomalies," LCO 3.1.3, "Control Rod OPERABILITY," LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators." During MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod OPERABILITY during these conditions.

RAI 3.9-2

ACTIONS

A.1

With one or more withdrawn control rods inoperable, action must be immediately initiated to fully insert the inoperable control rod(s). Inserting the control rod(s) ensures the shutdown and scram capabilities are not adversely affected. Actions must continue until the inoperable control rod(s) is fully inserted.

however, no specific scram time limit is imposed (P.1)

BASES (continued)

LCO Each withdrawn control rod must be OPERABLE. The withdrawn control rod is considered OPERABLE if the scram accumulator pressure is  $\geq 1940$  psig and the control rod is capable of being automatically inserted upon receipt of a scram signal. Inserted control rods have already completed their reactivity control function, and therefore are not required to be OPERABLE.

APPLICABILITY

During MODE 5, withdrawn control rods must be OPERABLE to ensure that in a scram the control rods will insert and provide the required negative reactivity to maintain the reactor subcritical.

For MODES 1 and 2, control rod requirements are found in LCO 3.1.2, "Reactivity Anomalies," LCO 3.1.3, "Control Rod OPERABILITY," LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators." During MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod OPERABILITY during these conditions.

RAI 3.9-2

ACTIONS

A.1

With one or more withdrawn control rods inoperable, action must be immediately initiated to fully insert the inoperable control rod(s). Inserting the control rod(s) ensures the shutdown and scram capabilities are not adversely affected. Actions must continue until the inoperable control rod(s) is fully inserted.

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1 and SR 3.9.5.2

During MODE 5, the OPERABILITY of control rods is primarily required to ensure a withdrawn control rod will automatically insert if a signal requiring a reactor shutdown occurs. Because no explicit analysis exists for automatic shutdown during refueling, the shutdown function is satisfied if the withdrawn control rod is capable of

(continued)

JUSTIFICATION FOR DIFFERENCES FROM NUREG - 1433  
ITS: SECTION 3.9.5 - CONTROL ROD OPERABILITY - REFUELING

NON-BRACKETED PLANT SPECIFIC CHANGES

- P.1 These changes are made to NUREG-1433 to reflect Fermi 2 current licensing basis; including design features, existing license requirements and commitments.
- P.2 Not used.
- P.3 Not used.
- P.4 Not used.
- P.5 Editorial change made for consistency with other ITS Bases that discuss the same function.
- P.6 The reference to the NRC Policy Statement has been replaced with a more appropriate reference to the Improved Technical Specification "split" criteria found in 10 CFR 50.36(c)(2)(ii).

RAI 3.9-2

B 3.9 REFUELING OPERATIONS

B 3.9.6 Reactor Pressure Vessel (RPV) Water Level [Irradiated Fuel]

LRAI 3.9-4

BASES

BACKGROUND

INSERT  
B 3.9.6-1

P.4

P.2

sufficient

The movement of [irradiated] fuel assemblies [or handling of control rods] within the RPV requires a minimum water level of 23 ft above the top of the RPV flange. During refueling, this maintains a sufficient water level in the reactor vessel cavity and spent fuel pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to  $\leq 25\%$  of 10 CFR 100 limits, as provided by the guidance of Reference 3.

an adequate

which

assumed to be released

APPLICABLE SAFETY ANALYSES

In accordance with Regulatory Guide 1.25,

P.2

P.4

over other irradiated assemblies seated within the RPV. During movement of [irradiated] fuel assemblies [or handling of control rods], the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet-to-cladding gap of all the dropped fuel assembly rods is retained by the water. The fuel pellet-to-cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).

above the damaged fuel

damaged

secondary

Analysis of the fuel handling accident inside containment is described in Reference 2. With a minimum water level of 23 ft and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water and that offsite doses are maintained within allowable limits (Ref. 4).

postulated by Regulatory Guide 1.25

While the worst case assumptions include the dropping of the irradiated fuel assembly being handled onto the reactor core, the possibility exists of the dropped assembly striking the RPV flange and releasing fission products. Therefore, the minimum depth for water coverage to ensure

also

(continued)

3.9 REFUELING OPERATIONS

3.9.7 Residual Heat Removal (RHR) - High Water Level

LCO 3.9.7 One RHR shutdown cooling subsystem shall be OPERABLE.

APPLICABILITY: MODE 5 with irradiated fuel in the reactor pressure vessel (RPV), the water level  $\geq$  20 ft 6 inches above the top of the RPV flange, and heat losses to ambient not greater than or equal to heat input to reactor coolant.

RAI 3.9-5

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required RHR shutdown cooling subsystem inoperable.	A.1 Verify an alternate method of decay heat removal is available.	1 hour <u>AND</u> Once per 24 hours thereafter
B. Required Action and associated Completion Time of Condition A not met.	B.1 Suspend loading irradiated fuel assemblies into the RPV.  <u>AND</u>	Immediately   (continued)

BASES

---

LCO (continued)

line may be used to allow pumps in one loop to discharge into the opposite loop's recirculation line to make a complete subsystem.

Additionally, each RHR shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. Operation (either continuous or intermittent) of one subsystem can maintain and reduce the reactor coolant temperature as required.

---

APPLICABILITY

One RHR shutdown cooling subsystem must be OPERABLE in MODE 5, with irradiated fuel in the reactor pressure vessel, with the water level  $\geq$  20 ft 6 inches above the top of the RPV flange, and heat losses to ambient not greater than or equal to heat input to the reactor coolant to provide decay heat removal. RHR System requirements in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS); Section 3.5, Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System; and Section 3.6, Containment Systems. RHR Shutdown Cooling System requirements in MODE 5 with irradiated fuel in the reactor pressure vessel and with the water level  $<$  20 ft 6 inches above the RPV flange are given in LCO 3.9.8.

ACTIONS

A.1

With no RHR shutdown cooling subsystem OPERABLE, the availability of an alternate method of decay heat removal must be established within 1 hour. In this condition, the volume of water above the RPV flange provides adequate capability to remove decay heat from the reactor core. However, the overall reliability is reduced because loss of water level could result in reduced decay heat removal capability. The 1 hour Completion Time is based on decay heat removal function and the probability of a loss of the available decay heat removal capabilities. Furthermore, verification of the functional availability of these

---

REFUELING OPERATIONS

A.1

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

LCO 3.9.7

~~3.9.11.1~~ At least one shutdown cooling mode loop of the residual heat removal (RHR) system shall be OPERABLE with at least:

- a. One OPERABLE RHR pump, and
- b. One OPERABLE RHR heat exchanger.

LA.1

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is greater than or equal to 20 feet 6 inches above the top of the reactor pressure vessel flange and heat losses to ambient\* are not sufficient to maintain OPERATIONAL CONDITION 5.

Applicability

ACTION:

With the RHR shutdown cooling mode loop OPERABLE, within 1 hour and at least once every 24 hours thereafter, verify the OPERABILITY of at least one alternate method capable of decay heat removal. Otherwise, suspend all operations involving an increase in the reactor decay heat load and establish SECONDARY CONTAINMENT INTEGRITY within 4 hours.

Action A

A.2

Action B

M.1

A.3

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

~~4.9.11.1~~ At least once per 12 hours verify at least one RHR shutdown cooling mode loop is capable of taking suction from the reactor vessel and discharging back to the reactor vessel through an RHR heat exchanger with available cooling water.

LA.2

Applicability

\*Ambient losses must be such that no increase in reactor vessel water temperature will occur (even though COLD SHUTDOWN conditions are being maintained).

RAI 3.9-5



7

3.9 REFUELING OPERATIONS

3.9.8 Residual Heat Removal (RHR)—High Water Level

<CTS>

LCO 3.9.8 One RHR shutdown cooling subsystem shall be OPERABLE ~~and in~~ *operation.* (3.9.11.1)

P.1

~~operation.~~  
-NOTE-  
The required RHR shutdown cooling subsystem may be removed from operation for up to 2 hours per 8 hour period.

20 ft, 6 inches,

APPLICABILITY: MODE 5 with irradiated fuel in the reactor pressure vessel (RVP), ~~and~~ the water level  $\geq$  [23] ft above the top of the RPV flange; and heat losses to ambient not greater than or equal to heat input to reactor coolant. (3.9.11.1 Applicability)

P.1

RA139-5

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required RHR shutdown cooling subsystem inoperable.	A.1 Verify an alternate method of decay heat removal is available.	1 hour <u>AND</u> <i>&lt; 3.9.11.1, action &gt;</i> Once per 24 hours thereafter
B. Required Action and associated Completion Time of Condition A not met.	B.1 Suspend loading irradiated fuel assemblies into the RPV.  <u>AND</u>	Immediately <i>&lt; 3.9.11.1, action &gt;</i>  (continued)

BASES

LCO  
(continued)

The RHR cross-tie line may be used (P.2) into

An OPERABLE RHR shutdown cooling subsystem consists of an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path. In MODE 5, the RHR cross-tie valve is not required to be closed; thus, the valve may be opened to allow pumps in one loop to discharge through the opposite loop's heat exchanger to make a complete subsystem. recirculation line

(P.1)

Additionally, each RHR shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. Operation (either continuous or intermittent) of one subsystem can maintain and reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required. A Note is provided to allow a 2-hour exception to shut down the operating subsystem every 8 hours.

APPLICABILITY  
(P.1)

(P.1)  
Insert  
3.9.7-1

One RHR shutdown cooling subsystem must be OPERABLE and in operation in MODE 5, with irradiated fuel in the reactor pressure vessel, and with the water level  $\geq$  [23] feet above the top of the RPV flange, to provide decay heat removal. RHR System requirements in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS); Section 3.5, Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System; and Section 3.6, Containment Systems. RHR Shutdown Cooling System requirements in MODE 5 with irradiated fuel in the reactor pressure vessel and with the water level  $<$  [23] ft above the RPV flange are given in LCO 3.9.8. 20 ft, 6 inches

20 ft, 6 inches  
RA13A-5

ACTIONS  
(P.4)

(P.4)

A.1

the availability of

With no RHR shutdown cooling subsystem OPERABLE, an alternate method of decay heat removal must be established within 1 hour. In this condition, the volume of water above the RPV flange provides adequate capability to remove decay heat from the reactor core. However, the overall reliability is reduced because loss of water level could

(continued)

Insert B 3.9.7-1

... , and heat losses to ambient not greater than or equal to  
heat input to the reactor core ...

RAI 3.9-5

3.9 REFUELING OPERATIONS

3.9.8 Residual Heat Removal (RHR) - Low Water Level

LCO 3.9.8 Two RHR shutdown cooling subsystems shall be OPERABLE, and with no recirculation pump in operation, one RHR shutdown cooling subsystem shall be in operation.

-----NOTES-----

1. The required operating RHR shutdown cooling subsystem may be removed from operation for up to 2 hours per 8 hour period.
  2. One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for surveillance testing.
- 

RAI 3.9-6

APPLICABILITY: MODE 5 with irradiated fuel in the reactor pressure vessel (RPV), the water level < 20 ft 6 inches above the top of the RPV flange, and heat losses to ambient not greater than or equal to heat input to reactor coolant.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two required RHR shutdown cooling subsystems inoperable.	A.1 Verify an alternate method of decay heat removal is available for each inoperable required RHR shutdown cooling subsystem.	1 hour <u>AND</u> Once per 24 hours thereafter

(continued)

BASES

LCO (continued)

opposite loop's recirculation line to make a complete subsystem.

Additionally, each RHR shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. Operation (either continuous or intermittent) of one subsystem can maintain and reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation of either an RHR pump or a recirculation pump is required. Note 1 is provided to allow a 2 hour exception to shut down the operating subsystem every 8 hours.

Note 2 is provided to allow a 2 hour exception for a single subsystem inoperability due to surveillance testing.

APPLICABILITY

Two RHR shutdown cooling subsystems are required to be OPERABLE, and one RHR pump or recirculation pump must be in operation in MODE 5, with irradiated fuel in the RPV, with the water level < 20 ft 6 inches above the top of the RPV flange, and heat losses to ambient not greater than or equal to heat input to the reactor coolant to provide decay heat removal. RHR System requirements in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS); Section 3.5, Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System; and Section 3.6, Containment Systems. RHR Shutdown Cooling System requirements in MODE 5 with irradiated fuel in the RPV and with the water level  $\geq$  20 ft 6 inches above the RPV flange are given in LCO 3.9.7, "Residual Heat Removal (RHR) - High Water Level."

RAI 3.9-67

# Specification 3.9.8

## REFUELING OPERATIONS LOW WATER LEVEL LIMITING CONDITION FOR OPERATION

A.1

LCO  
3.9.8

~~3.9.11.2 Reactor water level shall be maintained greater than or equal to 214 inches and two shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and, at least one recirculation pump shall be in operation, or at least one shutdown cooling mode loop shall be in operation\*\*\* with each loop consisting of at least:~~

LA.3

- a. One OPERABLE RHR pump, and
- b. One OPERABLE RHR heat exchanger.

LA.1

APPLICABILITY: OPERATIONAL CONDITION 5 when irradiated fuel is in the reactor vessel and the water level is less than 20 feet 6 inches above the top of the reactor pressure vessel flange and heat losses to ambient\*\* are not sufficient to maintain OPERATIONAL CONDITION 5.

Applicability

< ADD: ACTION B >

M.1

### ACTION:

Action A

a. With less than the above required RHR shutdown cooling mode loops OPERABLE, within 1 hour and at least once per 24 hours thereafter, verify the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop.

Action C

b. With neither a recirculation pump nor an RHR shutdown cooling mode loop in operation immediately initiate corrective action to return either at least one recirculation pump or at least one RHR shutdown cooling mode loop to operation as soon as possible. Within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

L.1

c. With reactor water level less than 214 inches, within 1 hour restore reactor water level to the required level or place two recirculation pumps in operation or place two RHR shutdown cooling mode loops in operation.

LA.3

### SURVEILLANCE REQUIREMENTS

SR 3.9.8.1

~~4.9.11.2.1 At least one shutdown cooling mode loop of the residual heat removal system or at least one recirculation pump shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.~~

LA.3

~~4.9.11.2.2 Verify reactor water level to be greater than or equal to 214 inches at least once per 12 hours.~~

SR 3.9.8.2

~~4.9.11.2.3 At least once per 12 hours verify the required RHR shutdown cooling mode loop(s) are capable of taking suction from the reactor vessel and discharging back to the reactor vessel through the RHR heat exchanger(s) with their associated cooling water available.~~

LA.2

LCO 3.9.8  
Note 2  
LCO 3.9.8  
Note 1

#One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing.

\*The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

\*\*The RHR shutdown cooling pump may be removed from operation during hydrostatic testing.

A.2

Applicability

\*\*Ambient losses must be such that no increase in reactor vessel water temperature will occur (even though COLD SHUTDOWN conditions are being maintained).

RAM 3.9-6

RAM 3.9-6

DISCUSSION OF CHANGES  
ITS: SECTION 3.9.8 - RHR - LOW WATER LEVEL

TECHNICAL CHANGES - LESS RESTRICTIVE  
"Generic"

- LA.1 CTS 3.9.11.2 includes details relating to system design, function, and Operability for the Residual Heat Removal System in Shutdown Cooling Mode. ITS 3.9.8 includes only a requirement for Operability and moves details of system design and specific Operability requirements to the Bases. This is acceptable because the requirement to maintain the function Operable is not impacted by the relocation of the design information. Therefore, these details can be adequately defined and controlled in the Bases which require change control in accordance with ITS 5.5.10, Bases Control Program.
- LA.2 CTS 4.9.11.2.3 requires verification of RHR shutdown cooling capability, and includes details of the required capability. ITS SR 3.9.8.2 requires this same surveillance, but relocates the details of RHR shutdown cooling capability to the Bases. This is acceptable because these details do not impact the ITS requirement to maintain the system Operable. These details can be adequately defined and controlled in the Bases which require change control in accordance with ITS 5.5.10, Bases Control Program. The system configuration and capability requirements are also described in the UFSAR, which are controlled in accordance with 10 CFR 50.59.
- LA.3 CTS 3.9.11.2 during Mode 5 operation with the reactor cavity not completely flooded requires reactor water level to be maintained  $\geq 214$  inches in addition to a means of forced circulation. CTS 3.9.11.2 Action c requires two means of forced circulation from either two recirculation pumps or two RHR-SDC loops in the event level is not maintained  $\geq 214$  inches. ITS 3.9.8 does not retain these restrictions -- only requiring one means of forced circulation regardless of reactor water level. This is consistent with the NUREG-1433. This water level requirement can be adequately defined and controlled in the Technical Requirements Manual (TRM), which requires revisions to be controlled by 10 CFR 50.59. This relocation continues to provide adequate protection of the public health and safety since the requirement for forced circulation, and alternate means of reactor coolant circulation on loss of forced circulation, continues to be required by the Technical Specifications. (A)

<CTS>

3.9 REFUELING OPERATIONS

3.9.8 Residual Heat Removal (RHR)—Low Water Level

LCO 3.9.8 Two RHR shutdown cooling subsystems shall be OPERABLE, and one RHR shutdown cooling subsystem shall be in operation. <sup>(P.1)</sup> <sup>(5)</sup> *with no recirculation pump in operation.* <3.9.11.2>

NOTE  
1. The required operating shutdown cooling subsystem may be removed from operation for up to 2 hours per 8 hour period. <3.9.11.2, X>

2. One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for surveillance testing. <3.9.11.2, X>

APPLICABILITY: MODE 5 with irradiated fuel in the reactor pressure vessel (RPV), and the water level <23> ft above the top of the [RPV flange], <sup>(P.1)</sup> <sup>(5)</sup> *20ft, 6 inches.* <3.9.11.2, Applicability, X>

ACTIONS *and heat losses to ambient not greater than or equal to heat input to reactor coolant.* RM 3.9-10

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two required RHR shutdown cooling subsystems inoperable.	A.1 Verify an alternate method of decay heat removal is available for each inoperable required RHR shutdown cooling subsystem.	1 hour AND Once per 24 hours hereafter
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action to restore <del>secondary</del> containment to OPERABLE status.  AND	Immediately  (continued)

<3.9.11.2, Action a>

<Doc M.1>

Rev 4



BASES

LCO  
(continued)

recirculation line

P.1

allow pumps in one loop to discharge through the opposite loop's heat exchanger to make a complete subsystem.

into P.1

Additionally, each RHR shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. Operation (either continuous or intermittent) of one subsystem can maintain and reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required. A Note is provided to allow a 2 hour excursion to shut down the operating subsystem every 8 hours.

INSERT  
B3.9.8-3

of either an RHR pump or a recirculation pump

APPLICABILITY

20 ft., 6 inches

P.1

INSERT  
B3.9.8-1

Two RHR shutdown cooling subsystems are required to be OPERABLE, and one must be in operation in MODE 5, with irradiated fuel in the RPV and with the water level  $\geq$  [23] ft above the top of the RPV flange, to provide decay heat removal. RHR System requirements in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS); Section 3.5, Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System; and Section 3.6, Containment Systems. RHR Shutdown Cooling System requirements in MODE 5 with irradiated fuel in the RPV and with the water level  $\geq$  [23] ft above the RPV flange are given in LCO 3.9.8, "Residual Heat Removal (RHR)—High Water Level."

RHR pump or recirculation pump

RA13.9-6

20 ft., 6 inches

ACTIONS

A.1

P.4

made available

With one of the two required RHR shutdown cooling subsystems inoperable, the remaining subsystem is capable of providing the required decay heat removal. However, the overall reliability is reduced. Therefore an alternate method of decay heat removal must be provided. With both required RHR shutdown cooling subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial RHR shutdown cooling subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour Completion Time is based on the decay heat removal function and the probability of a loss of the

(continued)

Rev 4

Insert B 3.9.8-1

... and heat losses to ambient not greater than or equal to heat input to the reactor core ...

RA1 3.9-6

Insert B 3.9.8-3

Note 2 is provided to allow a 2 hour exception for a single subsystem inoperability due to surveillance test ...

**INSERT THIS PAGE IN FRONT OF VOLUME 11**

<b>Volume 11: CTS MARKUP COMPILATION</b>	
<b>Remove</b>	<b>Replace</b>
3.4.1 CTS M/U (3/4 4-2) pg 3 of 6	3.4.1 CTS M/U (3/4 4-2) pg 3 of 6 Rev 4
3.4.4 CTS M/U (3/4 4-10) pg 1 of 2	3.4.4 CTS M/U (3/4 4-10) pg 1 of 2 Rev 4
3.4.5 CTS M/U (3/4 4-12) pg 3 of 3	3.4.5 CTS M/U (3/4 4-12) pg 3 of 3 Rev 4
3.4.7 CTS M/U (3/4 4-17) pg 2 of 3	3.4.7 CTS M/U (3/4 4-17) pg 2 of 3 Rev 4
3.4.7 CTS M/U (3/4 4-18) pg 3 of 3	3.4.7 CTS M/U (3/4 4-18) pg 3 of 3 Rev 4
3.4.10 CTS M/U (3/4 4-19) pg 5 of 8	3.4.10 CTS M/U (3/4 4-19) pg 5 of 8 Rev 4
3.4.11 CTS M/U (3/4 4-23) pg 1 of 1	3.4.11 CTS M/U (3/4 4-23) pg 1 of 1 Rev 4
3.9.7 CTS M/U (3/4 9-16) pg 1 of 1	3.9.7 CTS M/U (3/4 9-16) pg 1 of 1 Rev 4
3.9.8 CTS M/U (3/4 9-17) pg 1 of 1	3.9.8 CTS M/U (3/4 9-17) pg 1 of 1 Rev 4

# SPECIFICATION 3.4.1

< Also see Specification 3.4.10 >  
< Also see Specification 3.5.1 >

## REACTOR COOLANT SYSTEM

### SURVEILLANCE REQUIREMENTS

See  
Specification  
3.5.1

4.4.1.1.1 Each pump discharge valve shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each STARTUP\* prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER.

~~4.4.1.1.2 DELETED~~

A.1

4.4.1.1.3 With one reactor coolant system recirculation loop not in operation, at least once per 12 hours verify that:

- THERMAL POWER is less than or equal to 67.2% of RATED THERMAL POWER, and
- The individual recirculation pump flow controller for the operating recirculation pump is in the Manual mode, and
- The speed of the operating recirculation pump is less than or equal to 75% of rated pump speed.

4.4.1.1.4 With one reactor coolant system loop not in operation with THERMAL POWER less than or equal to 30% of RATED THERMAL POWER or with recirculation loop flow in the operating loop less than or equal to 50% of rated loop flow, verify the following differential temperature requirements are met within no more than 10 minutes prior to either THERMAL POWER increase or recirculation flow increase:

- Less than or equal to 145°F between reactor vessel steam space coolant and bottom head drain line coolant, and
- Less than or equal to 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel\*\*, and
- Less than or equal to 50°F between the reactor coolant within the loop not in operation and the operating loop.\*\*

See  
Specification  
3.4.10

See  
Specification  
3.5.1

\*If not performed within the previous 31 days.

See  
Specification  
3.4.10

\*\*Requirement does not apply when the recirculation loop not in operation is isolated from the reactor pressure vessel.

REACTOR COOLANT SYSTEM

A.1

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

LCO  
3.4.4

~~3.4.3.2~~ Reactor coolant system leakage shall be limited to:

- 3.4.4.a ~~a~~ No PRESSURE BOUNDARY LEAKAGE.
- 3.4.4.b ~~b~~ 5 gpm UNIDENTIFIED LEAKAGE.
- 3.4.4.c ~~c~~ 25 gpm total leakage averaged over any 24-hour period.

see  
specification  
3.4.5

d. Leakage specified in Table 3.4.3.2-1 at a reactor coolant system pressure of 1045 ± 10 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.

the previous

A.2

3.4.4.d ~~d~~ 2 gpm increase in UNIDENTIFIED LEAKAGE within ~~any~~ 24 hour period during OPERATIONAL CONDITION 1.

L.2

f. 2 gpm increase in UNIDENTIFIED LEAKAGE within any 4 hour period during OPERATIONAL CONDITIONS 2 and 3.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

Action C a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

Action A b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Action C

see  
specification  
3.4.5

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one other closed manual, activated automatic, or check\* valve, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the high/low pressure interface valve leakage pressure monitors shown in Table 3.4.3.2-2 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

see  
specification  
3.4.5

\*Which has been verified not to exceed the allowable leakage limit at the last refueling outage or after the last time the valve was disturbed, whichever is more recent.

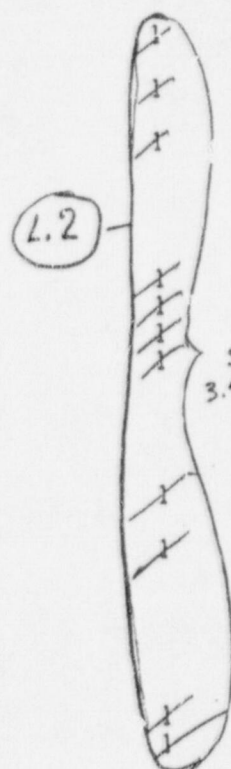
# Specification 3.4.5

**TABLE 3.4.3.2-1  
REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES**

VALVE NUMBER	VALVE DESCRIPTION
<b>1. RHR System</b>	
E11-F015A	LPCI Loop A Injection Isolation Valve
E11-F015B	LPCI Loop B Injection Isolation Valve
E11-F050A	LPCI Loop A Injection Line Testable Check Valve
E11-F050B	LPCI Loop B Injection Line Testable Check Valve
E11-F008	Shutdown Cooling RPV Suction Outboard Isolation Valve
E11-F009	Shutdown Cooling RPV Suction Inboard Isolation Valve
E11-F608	Shutdown Cooling Suction Isolation Valve
<b>2. Core Spray System</b>	
E21-F005A	Loop A Inboard Isolation Valve
E21-F005B	Loop B Inboard Isolation Valve
E21-F006A	Loop A Containment Check Valve
E21-F006B	Loop B Containment Check Valve
<b>3. High Pressure Coolant Injection System</b>	
E41-F007	Pump Discharge Outboard Isolation Valve
E41-F006	Pump Discharge Inboard Isolation Valve
<b>4. Reactor Core Isolation Cooling System</b>	
E51-F012	Pump Discharge Isolation Valve
E51-F013	Pump Discharge to Feedwater Header Isolation Valve

MAXIMUM LEAKAGE (gpm)

SR 3.4.5.1.b { 0.4<sup>(a)</sup>  
0.4<sup>(a)</sup>  
TO  
SR 3.4.5.1.c { 10



SR 3.4.5.1.b

(a) External Leakage from this valve shall be limited to 5 ml/min.

(A)

**TABLE 3.4.3.2-2  
REACTOR COOLANT SYSTEM INTERFACE VALVES  
LEAKAGE PRESSURE MONITORS**

VALVE NUMBER	SYSTEM	ALARM SETPOINT (psig)
E11-F015A & B, E11-F050A & B	RHR LPCI	≤ 449
E11-F008, F009, F608	RHR Shutdown Cooling	≤ 135
E21-F005A & B, E21-F006A & B	Core Spray	≤ 452
E41-F006, F007	HPCI	≤ 71
E51-F012, F013	RCIC	≤ 71

(LR.1)

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

- c. In OPERATIONAL CONDITION 1 or 2, with:
1. THERMAL POWER changed by more than 15% of RATED THERMAL POWER in 1 hour\*, or
  2. The off-gas level, at the delay pipe, increased by more than 10,000 microcuries per second in 1 hour during steady-state operation at release rates less than 75,000 microcuries per second, or
  3. The off-gas level, at the delay pipe, increased by more than 15% in one hour during steady-state operation at release rates greater than 75,000 microcuries per second;
- perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit.

L.4  
RAI #15

SURVEILLANCE REQUIREMENTS

SR  
3.4.7.1

4.4.5 The specific activity of the reactor coolant shall be demonstrated to be within the limits by performance of the sampling and analysis program of Table 4.4.5-1.

~~\*Not applicable during the startup test program.~~ L.4

RAI #15

TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	OPERATIONAL CONDITIONS IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Beta and Gamma Activity Determination	At least once per 72 hours	1, 2, 3 L.1
Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	At least once per 30 days	1 M.1 L.1
3 Radiochemical for I Determination	At least once per 6 months*	1** 2**, 3** 4** L.2
4. Isotopic Analysis for Iodine	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION b.	L.4 RAI #15
Required Action A.1 and B.1		
5. Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for at least Xe-133, Xe-135 and Kr-88	b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION c.	1, 2 L.4 RAI #14
	At least once per 31 days	A.3 L.1

Specification 3.4.7  
(Also see Specification 3.7.5)

\*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.  
\*\*Until the specific activity of the primary coolant system is restored to within its limits.

Required Action A.1 and B.1

See Specification 3.7.5



REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

(A.1)

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.4.10

SR 3.4.10.1.a

SR 3.4.10.2

~~3.4.6.1~~ The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 (1) Curve A for hydrostatic or leak testing; (2) Curve B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) Curve C for operations with a critical core other than low power PHYSICS TESTS, with:

SR 3.4.10.1.b.1

SR 3.4.10.1.b.2

SR 3.4.10.7

- a. A maximum heatup of 100°F in any 1-hour period,
- b. A maximum cooldown of 100°F in any 1-hour period,
- c. A maximum temperature change of less than or equal to 20°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and
- d. The reactor vessel flange and head flange temperature greater than or equal to 71°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

REQUIRED ACTION

A.1 & C.1

REQUIRED ACTION

A.2 & C.2

ACTION B

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 0 minutes; ~~perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations of Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.~~

ADD ACTION A NOTE  
ACTION C NOTE

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

~~3.4.6.1.1~~ During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figure 3.4.6.1-1 Curves A, B, or C, as applicable, at least once per 30 minutes.

RAI#25  
RAI#25

# Specification 3.4.11

## REACTOR COOLANT SYSTEM

A.1

## REACTOR STEAM DOME

### LIMITING CONDITION FOR OPERATION

LCO  
3.4.11

~~3.4.6.2~~ The pressure in the reactor steam dome shall be less than <sup>or equal to</sup> 1045 psig.

L.1

RAI #27

APPLICABILITY: OPERATIONAL CONDITIONS 1<sup>st</sup> and 2<sup>nd</sup>.

M.1

### ACTION:

Action A With the reactor steam dome pressure exceeding 1045 psig, reduce the pressure to less than 1045 psig within 15 minutes <sup>or</sup> be in at least HOT SHUTDOWN within 12 hours.  
Action B

### SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

~~3.4.5.2~~ The reactor steam dome pressure shall be verified to be less than 1045 psig at least once per 12 hours.

or equal to

L.1

RAI #27

M.1

\*Not applicable during anticipated transients.

REFUELING OPERATIONS

(A.1)

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

LCO 3.9.7 ~~3.9.11.1~~ At least one shutdown cooling mode loop of the residual heat removal (RHR) system shall be OPERABLE with at least:

- a. One OPERABLE RHR pump, and
  - b. One OPERABLE RHR heat exchanger.
- (LA.1)

Applicability

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is greater than or equal to 20 feet 6 inches above the top of the reactor pressure vessel flange and heat losses to ambient\* are not sufficient to maintain OPERATIONAL CONDITION 5.

ACTION:

Action A

With no RHR shutdown cooling mode loop OPERABLE, within 1 hour and at least once per 24 hours thereafter, verify the OPERABILITY of at least one alternate method capable of decay heat removal. Otherwise, suspend all operations

Action B

~~involving an increase in the reactor decay heat load and establish SECONDARY CONTAINMENT INTEGRITY within 4 hours~~

(M.1) (A.2) (A.3)

SURVEILLANCE REQUIREMENTS

SR3.9.7.1

~~4.9.11.1~~ At least once per 12 hours verify at least one RHR shutdown cooling mode loop is capable of taking suction from the reactor vessel and discharging back to the reactor vessel through an RHR heat exchanger with available cooling water.

(LA.2)

Applicability

\*Ambient losses must be such that no increase in reactor vessel water temperature will occur (even though COLD SHUTDOWN conditions are being maintained).

RA13.9-5

RA13.9-5

# Specification 3.9.8

## REFUELING OPERATIONS LOW WATER LEVEL LIMITING CONDITION FOR OPERATION

A.1

LCO 3.9.8

3.9.11.2 Reactor water level shall be maintained greater than or equal to 214 inches and two shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and, at least one recirculation pump shall be in operation, or at least one shutdown cooling mode loop shall be in operation\*\* with each loop consisting of at least:

LA.3

- a. One OPERABLE RHR pump, and
- b. One OPERABLE RHR heat exchanger.

LA.1

APPLICABILITY: OPERATIONAL CONDITION 5 when irradiated fuel is in the reactor vessel and the water level is less than 20 feet 6 inches above the top of the reactor pressure vessel flange and heat losses to ambient\*\* are not sufficient to maintain OPERATIONAL CONDITION 5.

Applicability

< ADD: ACTION B >

M.1

### ACTION:

Action A

a. With less than the above required RHR shutdown cooling mode loops OPERABLE, within 1 hour and at least once per 24 hours thereafter, verify the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop.

Action C

b. With neither a recirculation pump nor an RHR shutdown cooling mode loop in operation immediately initiate corrective action to return either at least one recirculation pump or at least one RHR shutdown cooling mode loop to operation as soon as possible. Within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

L.1

c. With reactor water level less than 214 inches, within 1 hour restore reactor water level to the required level or place two recirculation pumps in operation or place two RHR shutdown cooling mode loops in operation.

LA.3

### SURVEILLANCE REQUIREMENTS

SR 3.9.8.1

4.9.11.2.1 At least one shutdown cooling mode loop of the residual heat removal system or at least one recirculation pump shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

LA.3

4.9.11.2.2 Verify reactor water level to be greater than or equal to 214 inches at least once per 12 hours.

SR 3.9.8.2

4.9.11.2.3 At least once per 12 hours verify the required RHR shutdown cooling mode loop(s) are capable of taking suction from the reactor vessel and discharging back to the reactor vessel through the RHR heat exchanger(s) with their associated cooling water available.

LA.2

LCO 3.9.8 Note 2  
LCO 3.9.8 Note 1

\*One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing.

\*The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.

\*\*The RHR shutdown cooling pump may be removed from operation during hydrostatic testing.

A.2

Applicability

\*\*Ambient losses must be such that no increase in reactor vessel water temperature will occur (even though COLD SHUTDOWN conditions are being maintained).

RAI 3.9-6

RAI 3.9-6

INSERT THIS PAGE IN FRONT OF VOLUME 12

<b>Volume 12: IMPROVED TECHNICAL SPECIFICATIONS</b>	
<b>Remove</b>	<b>Replace</b>
3.4.1 ITS pg 3.4-1 Rev 2	3.4.1 ITS pg 3.4-1 Rev 4
3.4.5 ITS pg 3.4-11 Rev 0	3.4.5 ITS pg 3.4-11 Rev 4
3.4.5 ITS pg 3.4-12 Rev 0	3.4.5 ITS pg 3.4-12 Rev 4
3.4.6 ITS pg 3.4-13 Rev 0	3.4.6 ITS pg 3.4-13 Rev 4
3.4.6 ITS pg 3.4-14 Rev 0	3.4.6 ITS pg 3.4-14 Rev 4
3.4.10 ITS pg 3.4-24 Rev 0	3.4.10 ITS pg 3.4-24 Rev 4
3.9.7 ITS pg 3.9-10 Rev 0	3.9.7 ITS pg 3.9-10 Rev 4
3.9.8 ITS pg 3.9-12 Rev 0	3.9.8 ITS pg 3.9-12 Rev 4

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating



LCO 3.4.1 The reactor core shall not exhibit core thermal-hydraulic instability or operate in the "Scram" or "Exit" Regions.

AND

a. Two recirculation loops with matched recirculation loop jet pump flows shall be in operation;

OR

b. One recirculation loop may be in operation provided:

1. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Simulated Thermal Power - Upscale) Allowable Value of Table 3.3.1.1-1 is reset for single loop operation, when in MODE 1.

-----NOTE-----  
 Required allowable value modification for single loop operation may be delayed for up to 4 hours after transition from two recirculation loop operations to single recirculation loop operation.  
 -----

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Recirculation jet pump loop flow mismatch not within limits.	A.1 Declare recirculation loop with lower flow: "not in operation."	2 hours

(continued)

ACTIONS

-----NOTES-----

1. Separate Condition entry is allowed for each flow path.
  2. Enter applicable Conditions and Required Actions for systems made inoperable by PIVs.
- 

RAI #7

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. One or more flow paths with leakage from one or more RCS PIVs not within limit.</p>	<p>-----NOTE----- Each check valve used to satisfy Required Action A.1 must have been verified to meet SR 3.4.5.1 at the last refueling outage or after the last time the valve was disturbed, whichever is more recent.</p> <p>-----</p> <p>A.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of one other closed manual, de-activated automatic, or check valve.</p>	<p>4 hours</p>
<p>B. Required Action and associated Completion Time not met.</p>	<p>B.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>B.2 Be in MODE 4.</p>	<p>12 hours</p> <p>36 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.5.1 -----NOTE----- Not required to be performed in MODE 3. -----</p> <p>Verify equivalent leakage of each RCS PIV, at an RCS pressure <math>\geq 1035</math> and <math>\leq 1055</math> psig:</p> <ul style="list-style-type: none"> <li>a. For PIVs other than LPCI loop A and B injection isolation valves is <math>\leq 0.5</math> gpm per nominal inch of valve size up to a maximum of 5 gpm;</li> <li>b. For LPCI loop A and B outboard injection isolation valves is <math>\leq 0.4</math> gpm through-seat, and <math>\leq 5</math> ml/min external leakage; and</li> <li>c. For LPCI loop A and B inboard injection isolation testable check valves is <math>\leq 10</math> gpm.</li> </ul>	<p>In accordance with the Inservice Testing Program</p>

RAI # 62



3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Leakage Detection Instrumentation

LCO 3.4.6 The following RCS leakage detection instrumentation shall be OPERABLE:

RAI 10

- a. Drywell floor drain sump flow monitoring system;
- b. The primary containment atmosphere gaseous radioactivity monitoring system; and
- c. Drywell floor drain sump level monitoring system.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell floor drain sump flow monitoring system inoperable.	A.1 Restore drywell floor drain sump flow monitoring system to OPERABLE status.	30 days
B. Required primary containment atmosphere gaseous radioactivity monitoring system inoperable.	B.1 Analyze grab samples of primary containment atmosphere.	Once per 24 hours

(continued)

ACTIONS (continued)

RAI #12

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. Drywell floor drain sump level monitoring system inoperable.</p>	<p>C.1 -----NOTE----- Not applicable when primary containment atmosphere gaseous radioactivity monitoring system is inoperable. ----- Perform SR 3.4.6.1.</p>	<p>Once per 8 hours</p>
<p>D. Primary containment atmosphere gaseous radioactivity monitoring system inoperable.</p> <p><u>AND</u></p> <p>Drywell floor drain sump level monitoring system inoperable.</p>	<p>D.1 Restore primary containment atmosphere gaseous radioactivity monitoring system to OPERABLE status.</p> <p><u>OR</u></p> <p>D.2 Restore drywell floor drain sump level monitoring system to OPERABLE status.</p>	<p>30 days</p> <p>30 days</p>
<p>E. Required Action and associated Completion Time of Condition A, B, C, or D not met.</p>	<p>E.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>E.2 Be in MODE 4.</p>	<p>12 hours</p> <p>36 hours</p>
<p>F. All required leakage detection systems inoperable.</p>	<p>F.1 Enter LCO 3.0.3.</p>	<p>Immediately</p>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>C. -----NOTE----- Required Action C.2 shall be completed if this Condition is entered. -----</p> <p>Requirements of the LCO not met in other than MODES 1, 2, and 3.</p>	<p>C.1 Initiate action to restore parameter(s) to within limits.</p> <p><u>AND</u></p> <p>C.2 Determine RCS is acceptable for operation.</p>	<p>Immediately</p> <p>Prior to entering MODE 2 or 3</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.10.1 -----NOTE----- Only required to be performed as applicable during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. -----</p> <p>Verify:</p> <p>a. RCS pressure and RCS temperature are to the right of the limits specified in Figure 3.4.10-1; and</p> <p>b. RCS heatup and cooldown rates are limited to:</p> <p>1. <math>\leq 100^{\circ}\text{F}</math> in any 1 hour period; and</p> <p>2. <math>\leq 20^{\circ}\text{F}</math> in any 1 hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.</p>	<p>30 minutes</p>

RAI # 26

(continued)



3.9 REFUELING OPERATIONS

3.9.8 Residual Heat Removal (RHR) - Low Water Level

LCO 3.9.8 Two RHR shutdown cooling subsystems shall be OPERABLE, and with no recirculation pump in operation, one RHR shutdown cooling subsystem shall be in operation.

- NOTES-----
1. The required operating RHR shutdown cooling subsystem may be removed from operation for up to 2 hours per 8 hour period.
  2. One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for surveillance testing.
- 

RAI 3.9-6

APPLICABILITY: MODE 5 with irradiated fuel in the reactor pressure vessel (RPV), the water level < 20 ft 6 inches above the top of the RPV flange, and heat losses to ambient not greater than or equal to heat input to reactor coolant.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two required RHR shutdown cooling subsystems inoperable.	A.1 Verify an alternate method of decay heat removal is available for each inoperable required RHR shutdown cooling subsystem.	1 hour <u>AND</u> Once per 24 hours thereafter

(continued)

INSERT THIS PAGE IN FRONT OF VOLUME 13

<b>Volume 13: IMPROVED TECHNICAL SPECIFICATIONS BASES</b>	
<b>Remove</b>	<b>Replace</b>
B 3.4.3 ITS pg B 3.4.3-4 Rev 0	B 3.4.3 ITS pg B 3.4.3-4 Rev 4
B 3.4.6 ITS pg B 3.4.6-5 Rev 0	B 3.4.6 ITS pg B 3.4.6-5 Rev 4
B 3.9.2 ITS pg B 3.9.2-4 Rev 0	B 3.9.2 ITS pg B 3.9.2-4 Rev 4
B 3.9.5 ITS pg B 3.9.5-2 Rev 0	B 3.9.5 ITS pg B 3.9.5-2 Rev 4
B 3.9.7 ITS pg B 3.9.7-2 Rev 0	B 3.9.7 ITS pg B 3.9.7-2 Rev 4
B 3.9.8 ITS pg B 3.9.8-2 Rev 0	B 3.9.8 ITS pg B 3.9.8-2 Rev 4

## BASES

---

SURVEILLANCE REQUIREMENTS (continued)

The SR gives set pressures for all 15 SRVs installed. However, since only 11 SRVs are required, the SR is met if 11 SRVs are set properly.

The Frequency is required by the Inservice Testing Program and is consistent with the fact that Surveillance must be performed during shutdown conditions.

SR 3.4.3.2

A manual actuation of each required SRV is performed to verify that, mechanically, the valve is functioning properly and no blockage exists in the valve discharge line. This can be demonstrated by the response of the turbine control valves or bypass valves, by a change in the measured steam flow, or by any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the SRVs divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is to be performed is  $\approx$  850 psig (the pressure recommended by the valve manufacturer). Adequate steam flow is represented by turbine bypass valves open at least 20%. Plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME Code requirements, prior to valve installation. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. If a valve fails to actuate due only to the failure of the solenoid but is capable of opening on overpressure, the safety function of the SRV is considered OPERABLE.

BASES

ACTIONS (continued)

Action is to restore either of the inoperable monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a degraded configuration for a lengthy time period.

E.1 and E.2

If any Required Action of Condition A, B, C, or D cannot be 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 3 within 12 hours and MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to perform the actions in an orderly manner and without challenging plant systems.

F.1

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

SURVEILLANCE  
REQUIREMENTS

SR 3.4.6.1

This SR is for the performance of a CHANNEL CHECK of the required primary containment atmosphere gaseous radioactivity monitoring system. 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 3.4.6.2

This SR is for the performance of a CHANNEL FUNCTIONAL TEST of the required RCS leakage detection instrumentation. The test ensures that the monitors can perform their function in the desired manner. The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

(A)



BASES

---

SURVEILLANCE REQUIREMENTS (continued)

SR 3.9.2.2

(B) Performance of a CHANNEL FUNCTIONAL TEST demonstrates the associated refuel position one-rod-out interlock will function properly when a simulated or actual signal indicative of a required condition is injected into the logic. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other 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 CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping, or total channel steps so that the entire channel is tested. The 7 day Frequency is considered adequate because of demonstrated circuit reliability, procedural controls on control rod withdrawals, and visual and audible indications available in the control room to alert the operator to control rods not fully inserted. To perform the required testing, the applicable condition must be entered (i.e., a control rod must be withdrawn from its full-in position). Therefore, SR 3.9.2.2 has been modified by a Note that states the CHANNEL FUNCTIONAL TEST is not required to be performed until 1 hour after any control rod is withdrawn.

---

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. UFSAR, Section 7.6.1.1.
3. UFSAR, Section 15.4.1.1.

BASES

---

LCO Each withdrawn control rod must be OPERABLE. The withdrawn control rod is considered OPERABLE if the scram accumulator pressure is  $\geq 940$  psig and the control rod is capable of being automatically inserted upon receipt of a scram signal; however, no specific scram time limit is imposed. Inserted control rods have already completed their reactivity control function, and therefore are not required to be OPERABLE.

---

RAI 3.9-2

APPLICABILITY During MODE 5, withdrawn control rods must be OPERABLE to ensure that in a scram the control rods will insert and provide the required negative reactivity to maintain the reactor subcritical.

For MODES 1 and 2, control rod requirements are found in LCO 3.1.2, "Reactivity Anomalies," LCO 3.1.3, "Control Rod OPERABILITY," LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators." During MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod OPERABILITY during these conditions.

---

ACTIONS

A.1

With one or more withdrawn control rods inoperable, action must be immediately initiated to fully insert the inoperable control rod(s). Inserting the control rod(s) ensures the shutdown and scram capabilities are not adversely affected. Actions must continue until the inoperable control rod(s) is fully inserted.

---

BASES

---

LCO (continued)

line may be used to allow pumps in one loop to discharge into the opposite loop's recirculation line to make a complete subsystem.

Additionally, each RHR shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. Operation (either continuous or intermittent) of one subsystem can maintain and reduce the reactor coolant temperature as required.

---

APPLICABILITY

One RHR shutdown cooling subsystem must be OPERABLE in MODE 5, with irradiated fuel in the reactor pressure vessel, with the water level  $\geq$  20 ft 6 inches above the top of the RPV flange, and heat losses to ambient not greater than or equal to heat input to the reactor coolant to provide decay heat removal. RHR System requirements in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS); Section 3.5, Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System; and Section 3.6, Containment Systems. RHR Shutdown Cooling System requirements in MODE 5 with irradiated fuel in the reactor pressure vessel and with the water level  $<$  20 ft 6 inches above the RPV flange are given in LCO 3.9.8.

RA13.9-5

RA13.9-5

ACTIONS

A.1

With no RHR shutdown cooling subsystem OPERABLE, the availability of an alternate method of decay heat removal must be established within 1 hour. In this condition, the volume of water above the RPV flange provides adequate capability to remove decay heat from the reactor core. However, the overall reliability is reduced because loss of water level could result in reduced decay heat removal capability. The 1 hour Completion Time is based on decay heat removal function and the probability of a loss of the available decay heat removal capabilities. Furthermore, verification of the functional availability of these

---

BASES

LCO (continued)

opposite loop's recirculation line to make a complete subsystem.

Additionally, each RHR shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. Operation (either continuous or intermittent) of one subsystem can maintain and reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation of either an RHR pump or a recirculation pump is required. Note 1 is provided to allow a 2 hour exception to shut down the operating subsystem every 8 hours.

Note 2 is provided to allow a 2 hour exception for a single subsystem inoperability due to surveillance testing.

APPLICABILITY

Two RHR shutdown cooling subsystems are required to be OPERABLE, and one RHR pump or recirculation pump must be in operation in MODE 5, with irradiated fuel in the RPV, with the water level < 20 ft 6 inches above the top of the RPV flange, and heat losses to ambient not greater than or equal to heat input to the reactor coolant to provide decay heat removal. RHR System requirements in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS); Section 3.5, Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System; and Section 3.6, Containment Systems. RHR Shutdown Cooling System requirements in MODE 5 with irradiated fuel in the RPV and with the water level  $\geq$  20 ft 6 inches above the RPV flange are given in LCO 3.9.7, "Residual Heat Removal (RHR) - High Water Level."

RAI 3.9-67