ITS DISCUSSION OF DIFFERENCES
ITS Section 3.4B: Reactor Coolant System

Note: The ITS Section 3.4B package addresses the following NUREG-1430 RSTS:

RSTS 3.4.9	Pressurizer
RSTS 3.4.10	Pressurizer Safety Valves
RSTS 3.4.11	Pressurizer Power Operated Relief Valve (PORV) -- Not used
RSTS 3.4.12	Low Temperature Overpressure Protection (LTOP) System
RSTS 3.4.13	RCS Operational Leakage
RSTS 3.4.14	RCS Pressure Isolation Valve Leakage
RSTS 3.4.15	RCS Leakage Detection Instrumentation
RSTS 3.4.16	RCS Specific Activity

1 NUREG 3.4.13 and Bases – The CTS 3.1.6.3.b limitations on primary to secondary leakage are retained in the ITS. Thus, NUREG 3.4.13.d is shown as deleted since there is no CTS equivalent and it largely replicates the requirements of NUREG 3.4.13.e which is revised to specify the CTS limits on leakage. This change is consistent with current license basis.

The CTS 3.1.6 Completion Times are retained for restoring the identified and unidentified leakage to within limits. CTS requires only that the unit be shutdown within 24 hours. ITS 3.4.13 Required Action B.1 will allow 18 hours for attempts to restore and Required Action C.1 will require the unit to be in MODE 3, i.e., shutdown, within the following 6 hours. This results in an equivalent 24 hours to shutdown the unit. This change is consistent with current license basis.

2 NUREG 3.4.10 and Bases - The LCO is revised to require only that the pressurizer safety valves be OPERABLE without specifying the specific setpoint. The setpoints are not included in the CTS. ITS SR 3.4.10.1 will require that the valves be determined OPERABLE in accordance with the Inservice Testing (IST) Program. The IST Program will include the setpoints and be subject to NRC review as required by 10 CFR 50.59. Compliance with the ASME Code as described in the CTS continues to be required by design controls and by the IST Program. Both are subject to NRC review and provide adequate control of the pressurizer safety valve setpoints. Additionally, the CTS 3.1.1 Bases indicate that the pressurizer safety valve setpoint tolerance range is +1%, -3%. This tolerance range is incorporated into the ITS Bases. This change is consistent with current license basis.

3 NUREG 3.4.10 and Bases - The LCO and Applicability are proposed to be consistent with the CTS 3.1.1.3.A, i.e., MODES 1 and 2, and to include MODE 3 and MODE 4 with the RCS above the LTOP enable temperature as currently required by CTS 3.1.1.3.B. For ANO-1, the LTOP enable temperature occurs in MODE 4 at 262°F. This is consistent with the NUREG position that the safety valves provide overpressure protection above the LTOP enable temperature and the LTOP requirements (see ITS 3.4.11) provide overpressure protection in conditions below the LTOP enable temperature. However, ITS 3.4.10 is proposed to require only one pressurizer safety valve in MODE 3 and in MODE 4 above the LTOP enable temperature. This is consistent with CTS 3.1.1.3.B and is based on the capability of one safety valve to remove the equivalent of all available heat sources as discussed in

3.4B-23 the Bases for CTS 3.1.1.3.B and SAR Section 4.3.11.4. Additionally, when the reactor is subcritical, CTS 3.1.1.3.B specifically allows both code safety valves to be inoperable during hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III, and states that LCO 3.0.3 is not applicable. These allowances have been incorporated in the ITS as LCO 3.4.10 LCO Notes 3 and 4. The addition of these allowances retains the requirements of the current license basis. Also, a discussion of the ANO treatment of the instrument uncertainty for this parameter has been inserted. This change is consistent with current license basis.

The Conditions are also revised to reflect that the LCO only requires one safety valve in MODES 3 and 4 (by including "required" in ITS Condition C). These proposed Conditions are equivalent to or more restrictive than the CTS. In MODES 3 and 4, the CTS has no required actions and LCO 3.0.3 is not applicable. Therefore, requiring that the unit cool to less than the LTOP enable temperature is more restrictive than CTS.

3.4B-02 The proposed format is consistent with the format used in other LCOs that specify different levels of system Operability in different Modes of operation such as NUREG 1430 for LCO 3.7.5, "Emergency Feedwater (EFW) System" and proposed ITS LCO 3.6.5, "Reactor Building Spray and Cooling Systems.

4 NUREG 3.4.10 and Bases – The Applicability Note has been relocated to the LCO in the ITS. This provides consistency since the note modifies the LCO requirements, not the Applicability. This change is considered to be administrative in nature.

5 NUREG 3.4.11 and Bases - This LCO is not adopted for ANO-1. NUREG 3.4.12 is renumbered to ITS 3.4.11, and NUREG 3.4.16 is renumbered to ITS 3.4.12. The ANO response (dated December 21, 1990) to Generic Letter 90-06 indicates that the pressurizer electromatic (power operated) relief valve (ERV) is not depended upon for a safety related function in MODES 1, 2, or 3, nor in MODE 4 above the LTOP enable temperature. Therefore, it is not significant to risk and does not satisfy Criterion 4 of 10 CFR 50.36 during these MODES. The ANO-1 CTS does include an exercise of the PORV (ERV) in Table 4.1-2, item 17 in order to demonstrate OPERABILITY for purposes of LTOP. Therefore, this Surveillance is reflected in the LTOP ITS as SR 3.4.11.5. This change is consistent with current license basis.

6 NUREG 3.4.12 and Bases - The LTOP LCO requirements are provided in ITS 3.4.11
3.4B-05 and revised to be consistent with unit specific design, analysis, and licensing basis as reflected in CTS 3.1.2.9, 3.1.2.10, 3.1.2.11, and Table 4.1-2, item 17. The ACTIONS and Surveillance Requirements are similarly revised to reflect the LCOs and the LTOP analysis. The Applicability Note concerning CFT pressurization has been relocated to the LCO for consistency with other Notes retained from the CTS.

<u>a. Makeup Pumps</u>

3.4B-04 NUREG 3.4.12 includes requirements to limit the makeup pumps to a maximum of one capable of injecting into the RCS. No such explicit requirements are included in the CTS. The ANO LTOP analysis assumes that flow from one HPI/makeup pump is introduced through the failed open makeup valve. However, assuming that a second

HPI/makeup pump starts would be an additional failure, since the postulated failure of the makeup valve fully open is the assumed limiting single failure. Therefore, additional requirements limiting the availability of the HPI/makeup pumps is not required. It should be noted that during pump swaps, two HPI/makeup pumps would be inservice at the same time. This is acceptable due to the infrequent nature of this activity, the low probability of a makeup valve failure during this short period of time, and due to the attention of the operating crew during this evolution. NUREG 3.4.12 Condition A and NUREG SR 3.4.12.1 are also not incorporated for makeup pumps since the LCO requirements for makeup pumps were not incorporated. This change is consistent with current license basis.

b. High Pressure Injection

NUREG 3.4.12 includes requirements for HPI to be de-activated. This requirement is included in ITS LCO 3.4.11 along with the previously approved CTS allowances to be capable of injecting under administrative controls for specific purposes, i.e., ITS LCO 3.4.11 Notes 1, 2, 3, and 4. NUREG 3.4.12 Condition B is incorporated in ITS 3.4.11 Condition E, and NUREG SR 3.4.12.2 is incorporated in ITS SR 3.4.11.2 (also including the previously approved CTS allowances to be capable of injecting under administrative controls for specific purposes). This change is consistent with current license basis.

c. Core Flood Tanks

NUREG 3.4.12 includes requirements for the core flood tank(s) (CFTs) to be isolated when the CFT pressure is greater than or equal to the maximum RCS pressure for the existing RCS temperature allowed by the pressure and temperature limit curves. This requirement is included in ITS LCO 3.4.11 along with the previously approved CTS allowances to be capable of unisolated under administrative controls for ASME Section XI testing. NUREG 3.4.12 Conditions C and D are not incorporated in ITS 3.4.11. Rather, an unisolated pressurized CFT results in entry into ITS 3.4.11 Condition E. Since no ACTION is presented in CTS, an immediate restoration type action is more consistent with the CTS application. Also, since restoration is expected to take less than the one hour provided by NUREG Condition C, the proposed Action is more restrictive than NUREG 3.4.12 Conditions C and D. NUREG SR 3.4.12.3 is incorporated in ITS SR 3.4.11.3 (also including the previously approved CTS allowances to be capable of unisolated under administrative controls for ASME Section XI testing). This change is consistent with current license basis.

d. Pressure Relief Capability

NUREG 3.4.12 includes requirements for pressure relief capability by requiring either an OPERABLE PORV with a designated pressurizer level, or a depressurized RCS with a specified minimum vent size. These requirements are included in ITS LCO 3.4.11 items a and b. The CTS 3.1.2.11 requirement to prevent operation in a water solid condition is included as ITS LCO 3.4.11 item a in lieu of a specified pressurizer water level, and CTS Table 4.1-2, item 17 is also incorporated in ITS LCO 3.4.11 item a. Details of what constitutes a water solid condition are provided in the

3.4B-19

Bases. This is consistent with the current license basis, as these requirements are currently maintained in implementing procedures that are referenced from SAR Section 12.5.2.1. This requirement to not operate the RCS in a water solid condition when the RCS pressure boundary is intact was incorporated in the CTS by Amendment 95, dated March 4, 1985. Placing specific pressurizer level criteria in the ITS LCO would decrease the flexibility ANO currently has to evaluate the unit conditions and establish suitable RCS pressure versus pressurizer level requirements based on the needs of the unit, while maintaining a 10 minute criteria for operator action. Since the values are contained in the ITS Bases, changes to these values would be evaluated in accordance with the requirements of 10 CFR 50.59.

NUREG LCO 3.4.12 item b contains the requirement that the RCS be depressurized with a vent of a minimum specified size. ITS LCO 3.4.11 item b specifies that the RCS be depressurized and the RCS is open. This is consistent with the CTS 3.1.2.11 requirements to prevent operation in a water solid condition "when the RCS pressure boundary is intact." Currently, openings that satisfy the minimum vent size assumed in the LTOP analysis are administratively controlled in the implementing procedures. Acceptable openings that satisfy the minimum vent size have been added to the Bases for ITS 3.4.11. Changes to the Bases are evaluated in accordance with the requirements of 10 CFR 50.59, as required by the Technical Specifications Bases Control Program. The terminology of "RCS open" and "RCS intact" have been used at ANO for many years, and the operations personnel are well acquainted with their usage. This change is consistent with the current license basis.

NUREG 3.4.12 Conditions E and F are incorporated in ITS 3.4.11 Conditions A and B, and NUREG SR 3.4.12.4 is incorporated in ITS SR 3.4.11.1 (also including the previously approved CTS exceptions under administrative controls for specific purposes). NUREG 3.4.12 Conditions G and H are incorporated in ITS 3.4.11 Conditions C and D. However, NUREG Required Action H.2 is not incorporated in

3.4B-06

ITS Condition D because ANO-1 is not equipped with a low low makeup tank level interlock to the BWST suction valves. Therefore, the BWST is not automatically aligned if the makeup tank level is low. NUREG SR 3.4.12.5 and SR 3.4.12.6 are incorporated in ITS SR 3.4.11.4.

3.4B-11

NUREG SR 3.4.12.7 and SR 3.4.12.8 are revised to reflect the CTS Table 4.1-2, item 17 testing requirements as ITS SR 3.4.11.5. CTS Table 4.1-2 requires a test of the ERV (PORV) at the end of each refueling. This is revised to require a Channel Calibration of the ERV each 18 months. This is consistent with the interpretation of the current license basis. A separate Channel Functional Test is not required to be performed by the CTS. Since the CTS Table 4.1-2 testing frequency is 18 months, and since the Channel Calibration encompasses the Channel Functional Test, per the definitions of these two tests, deletion of the NUREG Channel Functional Test is acceptable. The ANO testing of the ERV was most recently reviewed in conjunction with our Generic Letter 90-06 responses dated December 21, 1990 (1CAN129013), and February 5, 1993 (1CAN029301).

Finally, NUREG 3.4.12 Condition I is reflected in ITS 3.4.11 Condition E. Again, since no ACTION is presented in CTS for these conditions, an immediate restoration type action is more consistent with the CTS application. Also, since restoration is expected to take less than the 12 hours provided by NUREG Condition I, the proposed Action is more restrictive than NUREG 3.4.12 Condition I. This change is consistent with current license basis and current practice.

NUREG 3.4.12 includes an LTOP enable temperature in the Applicability statement which is incorporated in ITS 3.4.11 Applicability beginning at the LTOP enable temperature of 262°F in MODE 4. Statement clarifying the ANO treatment of the instrument uncertainties associated with this parameter have been inserted. The CTS requirements were determined to be sufficient LTOP requirements, as indicated in the ANO response (dated December 21, 1990) to Generic Letter 90-06, and are proposed to be retained as modified to include appropriate Actions and SRs. This change is consistent with current license basis.

Bases changes are included as necessary to reflect the unit specific LTOP evaluations and the aforementioned changes.

7 NUREG 3.4.16 Bases - The Applicable Safety Analyses and References sections are revised to incorporate a thyroid dose conversion factor reference to the defined term DOSE EQUIVALENT I-131 in Section 1.1, Definitions and to provide unit specific dose analysis information.

3.4B-20

NUREG SR 3.4.16.1 Bases have been revised to incorporate details on performing the gross specific activity analysis consistent with information contained in CTS Table 4.1-3, Note 1. This change is consistent with the current license basis.

3.4B-15

Information from CTS Table 4.1-3, Note 2 providing details on the determination of E-bar have been incorporated in the Bases for SR 3.4.12-3. In addition, two references have been added to the Bases Reference section. This change is consistent with the current license basis.

8 NUREG 3.4.9 and Bases - The required pressurizer heaters are permanently connected to a bus powered by an emergency (ES) power source. Therefore, as indicated in the Bases for NUREG SR 3.4.9.3, there is no need to periodically verify the capability of these heaters to be powered from an emergency power supply. Further, with such a design, the power supply and distribution system is considered to be a support system for the required pressurizer heaters. However, since ANO-1 has both ES powered and non-ES powered heaters, the ITS is revised to retain the ES bus power requirement for clarity. SR 3.4.9.2 is also revised to reflect this permanent connection. Finally, the documentation related to this design was provided in response to NUREG-0578 (prior to NUREG-0737). This change is consistent with current license basis.

9 NUREG 3.4.10 Bases - This change provides unit specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures. These changes are consistent with current license basis.

B 3.4.10 BACKGROUND - An additional plant specific method for pressurizer safety valve discharge flow monitoring is identified.
B 3.4.10 ASA - Unit specific analysis information is incorporated.

10 NUREG 3.4.9 Bases - This change provides unit specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures. These changes are consistent with current license basis.
B 3.4.9 BACKGROUND - An additional plant specific method for maintaining "subcooled conditions" is identified.
B 3.4.9 ASA - The description of the applicable safety analysis is revised to reflect the unit specific analysis. The Bases for CTS 3.1.3 indicate the applicable safety analysis for pressurizer water level are rod withdrawal and startup events.
B 3.4.9 SR 3.4.9.1 - The statement regarding use of "indicated level" is replaced with a unit specific statement regarding instrument error.

11 NUREG 3.4.13 and Bases – SR 3.4.13.1 is revised to remove the Frequency column Note and to modify the Surveillance column Note in accordance with TSTF-116, Rev. 2. A unit specific clarification of the conditions required for the performance of this SR has been incorporated by stipulating that the SR is only required to the performed following the establishment of steady state operation at near operating pressure. Without this latter condition, the inventory balance results are not reliable. The changes in the Notes are consistent with Generic Traveler TSTF-116, Rev. 2.

12 The NUREG 3.4.14 Bases are revised to reflect unit specific requirements imposed by NRC Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves, issued April 20, 1981. Further, changes were made to reflect unit specific design and operational features. These changes are consistent with current license basis.

13 NUREG 3.4.14 and Bases - The requirements of CTS 3.1.6.9 are retained. If RCS pressure isolation valve (PIV) leakage exceeds the allowed limits, the high pressure portion of the affected subsystem must be isolated from the low pressure portion by 3.4B-18 closing at least two valves in the high pressure portion of the piping. These requirements were incorporated in the current operating license by an Order For Modification dated April 20, 1981. The Technical Specifications incorporated as a result of this Order included surveillance testing requirements to perform leakage tests on the RCS PIVs per Table 3.1.6.9. CTS Table 3.1.6.9 contains requirements for leak testing only the check valves contained in high pressure system/low pressure system interface. CTS does not contain a requirement to perform a leak test on the MOVs also contained in these lines. ITS SR 3.4.14.1 requires leak testing of the PIV check valves only, consistent with the CTS. Since the only valves that are required to be leak tested are the check valves, incorporation of the Action A Note would require leak testing of the MOVs, since they are also used to isolate the interface lines. Since only check valves are addressed in ITS SR 3.4.14.1, the NUREG Action A Note is not adopted. This is acceptable because the only check valves that could be used to isolate the lines are leak tested per ITS SR 3.4.14.1, and the CTS, as implemented per the NRC's Order do not require leak testing of the MOVs. This change is consistent with the current license basis.

Also included from the CTS is the more restrictive requirements to isolate with both valves (the MOV and the remaining OPERABLE check valve) within 4 hours. When using the MOV for isolation, deactivation makes the low pressure injection subsystem of the ECCS inoperable since the MOV must automatically open to provide the LPI ECCS function. Therefore, no additional actions to "restore" are necessary since the ECCS Specification will effectively limit continued operation. This change is consistent with the current license basis.

Also, the Conditions are re-ordered to provide a default shutdown Required Action if the DHR System autoclosure interlock is inoperable and the Required Action to "isolate" can not be met within 4 hours. This change is considered an editorial enhancement in the usage of the ITS.

14 NUREG 3.4.14 and Bases – SR 3.4.14.1 is revised to provide only a 5 gpm limit since the valves are large enough so the 0.5 gpm per nominal inch of valve size would exceed the 5 gpm limit. The Surveillance is also revised to refer to a differential test pressure consistent with the CTS method for performing this leakage testing. Note 3 and the last Frequency are omitted since this plant was licensed prior to 1980 (see Bases for SR 3.4.14.1 which identifies this criterion). Finally, the allowance for indirect measurement provided in CTS Table 3.1.6.9 footnote (c) is retained in the Bases as an acceptable method for performance of the SR. Other Bases discussions were also revised to reflect these changes. This change is consistent with current license basis.

15 NUREG 3.4.14 and Bases – SR 3.4.14.2 and SR 3.4.14.3 are revised to omit the Notes since the ITS 3.4.11 LTOP requirements do not require disabling the DHR autoclosure interlocks. This change is consistent with current license basis.

16 NUREG 3.4.14 and Bases - Two additional SRs are provided to periodically test the portion of the DHR autoclosure interlock associated with closed Core Flood Tank isolation valves. This function either closes, if open, or prevents opening, if closed, of the DHR System suction MOVs if the CFT isolation valves are not fully closed. This retains CTS Table 4.1-1, item 30, and Table 4.1-2, item 11 requirements. This change is consistent with current license basis.

17 NUREG 3.4.15 and Bases – Required Actions A.1 and B.1.2 are revised to include a Note indicating that SR 3.4.13.1, an RCS water inventory balance, is not required to be performed until 12 hours after establishing steady state operating conditions. This Note recognizes the fact that performance of SR 3.4.13.1 during non-steady state operation results in the generation of calculational data that is not reliable. This change is consistent with TSTF-116, Rev. 2. In addition, the ITS is revised to retain the at or near operating pressure requirement since calculations performed during other conditions have historically proven unreliable. Steady state operation at or near operating pressure is required to perform a proper inventory balance.

18 NUREG 3.4.10 Bases – Incorporated TSTF-057.

19 NUREG 3.4.16 and Bases - The CTS 3.1.4 requirements for RCS specific activity are retained. Editorially, the specific activity limits are listed in the LCO. The ACTIONS are revised to retain the single requirement to restore within 24 hours with no dependence on the Figure related to iodine spiking. These ACTIONS are consistent with CTS and with the CTS Bases which does not consider iodine spiking. The retention of the CTS requirements also combines NUREG Conditions B and C since these Required Actions are the same after including TSTF-028 which omits Required Action C.1. Also, the second entry condition of NUREG Condition B is unnecessary because: 1) it is the same as the first entry condition, i.e., if specific activity is too high Condition A is entered and Required Action A.1 is completed; and 2) the referenced figure does not exist due to the LCO modification. This change is consistent with current license basis.

3.4B-17 20 Not used.

21 NUREG 3.4.9 and Bases - Maximum and minimum pressurizer water level limits are specified by CTS 3.1.3.4. These unit specific multiple pressurizer water level limits (maximum and minimum) are required by ITS LCO 3.4.9 and identified in ITS SR 3.4.9.1. This change is consistent with the current license basis.

In addition, the MODE 4 LTOP temperature Applicability is revised from "≥ 262 F" to "> 262 F" and a discussion of the ANO treatment of the associated instrument uncertainties has been inserted to be consistent with other similar Applicabilities, e.g., NUREG 3.4.10 and 3.4.11, and to eliminate the overlap with NUREG 3.4.9 Required Action B.2 (which includes the "= 262 F").

ANO-344 22 Not used.

23 NUREG 3.4.15 Bases - The Bases for RCS Leakage Detection Instrumentation are revised to reflect unit specific design, capabilities, and licensing basis. This change is consistent with current license basis.

24 NUREG 3.4.14 and Bases - The actual setpoints for high RCS pressure which prevent valve opening are not included in NUREG SR 3.4.14.2 for the DHR System autoclosure interlocks. The CTS administrative controls for these interlock function setpoints have been adequate to assure OPERABILITY of the system and are proposed to continue as such. This change is consistent with current license basis.

NUREG SR 3.4.1.2 and SR 3.4.1.3 Bases - Discussion of the design of the interlocks has been revised to reflect the ANO design as stated in the ANO SAR, Section 9.5.2.7. Additional information has been added to discuss the treatment of instrument uncertainty. These changes are consistent with the current license basis.

25 Not used

26 NUREG 3.4.13 Bases - Incorporates TSTF-054, Rev. 1.

ITS DISCUSSION OF DIFFERENCES

27 NUREG 3.4.13 and Bases - Incorporates TSTF-061.

28 NUREG 3.4.15 and Bases - Incorporates TSTF-060.

29 NUREG 3.4.16 and Bases - The second Frequency of NUREG SR 3.4.16.2 is not adopted. This Frequency is not required by CTS. Administratively controls for verification of reactor coolant specific activity during power maneuvering have been adequate to date and are retained. This change is consistent with current license basis.

30 NUREG 3.4.15 and Bases - The references to "containment" are revised to "reactor building" consistent with ANO-1 terminology in the license basis documents. References to "containment" in the NUREG-1430 text are changed to "reactor building," "the reactor building," or the abbreviation "RB" as appropriate for the ITS context. However, marking up the NUREG pages to show these changes introduces significant clutter to the page with little value for the purpose of the markup. Therefore, only one reference to this DOD item will be placed on each page of the NUREG/ITS markup for this section at the first occurrence with subsequent changes on that page not marked or annotated with this DOD number to conserve margin space

31 NUREG 3.4.13 Bases - This change provides unit specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures. This change is consistent with current license basis.

32 NUREG Bases - ANO-1 was designed and licensed to the AEC's General Design Criteria (GDC) which was published in the Federal Register on July 11, 1967 [32FR10213]. Appendix A to 10 CFR 50 effective in 1971 [36FR3256] and subsequently amended, is somewhat different from the proposed 1967 criteria. SAR Section 1.4 includes an evaluation of ANO with respect to the 1967 criteria. The NUREG statement concerning the GDC criteria is modified in the ITS to reference the current licensing basis description in the SAR. This change is consistent with current license basis.

33 NURGE-1430 SR 3.4.14.1 Note 2 is deleted. It provides a performance exception during times when DHR is in service. In accordance with the LCO Note (see DOD-20) the valves are not required to meet the LCO (and therefore the SRs) when operating in the DHR mode in MODE 4. For ANO, this LCO exception encompasses the intended allowance of this SR Note. As such the Note serves no purpose and can be removed.

34 Not used

ITS DISCUSSION OF DIFFERENCES

35 NUREG Bases - The Criterion statement at the conclusion of the Applicable Safety Analysis section was modified at each occurrence to refer to 10 CFR 50.36 instead of the NRC Policy Statement. This is an editorial change associated with the implementation of the 10 CFR 50.36 rule changes after NUREG-1430, Revision 1 was issued.

For ITS LCOs 3.4.9, 3.4.10, 3.4.13, 3.4.14, and 3.4.15, the 10 CFR 50.36 Criterion satisfied by the respective ITS LCOs was modified to preserve consistency with the ANO-1 license basis. Specifically, the MODE dependency of the safety analyses was represented in establishing which criterion is met as a function of the operating MODE. For lower MODE LCOs, Criterion 4 of 10 CFR 50.36 was cited as the basis for inclusion of these LCOs. This change is consistent with current license basis and 10 CFR 50.36.

ANO-249 36. Incorporated Generic Change TSTF-352, Rev. 1.

CTS

3.4B-01

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with: (21)

a. Pressurizer water level ~~≤ [290] inches~~ ≥ 45 inches and ≤ 320 inches; and 3.1.3.4

b. A minimum of ~~[126]~~ 99 kW of Engineered Safeguards (ES) bus powered pressurizer heaters OPERABLE. ~~[and capable of being powered from an emergency power supply]~~. 3.1.3.6 (8)

---NOTE---
OPERABILITY requirements on pressurizer heaters do not apply in MODE 4. NA

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with RCS temperature ~~≥~~ > ~~[275]~~ 262 °F. 3.1.3.4 3.1.3.6 (21)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. Pressurizer water level not within ~~limit~~ limits.	A.1 Restore level to within ~~limit~~ limits.	1 hour	3.1.3.7 (21)
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 with RCS temperature ≤ ~~[275]~~ 262 °F.	6 hours ~~12~~ 24 hours	3.1.3.7 NA (36)

(continued)

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≥ 45 inches and ≤ 320 inches

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME	CTS
C. Capacity of ES bus powered pressurizer heaters ~~[capable of being powered by emergency power supply]~~ less than limit.	C.1	Restore pressurizer heater ~~capability~~ capacity.	72 hours	3.1.3.6 edit 8
D. Required Action and associated Completion Time of Condition C not met.	D.1	Be in MODE 3.	6 hours	NA
	AND			
	D.2	Be in MODE 4.	12 hours	3.1.3.6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	CTS
SR 3.4.9.1	Verify pressurizer water level ~~≤ [290] inches~~.	12 hours	NA 3.1.3.4 21
SR 3.4.9.2	Verify Capacity ~~≥ [126] kW~~ of ES bus powered pressurizer heaters ~~are capable of being powered from an emergency power supply~~.	~~[18]~~ months	Tbl 4.1-2 #7 8
~~SR 3.4.9.3~~	~~Verify emergency power supply for pressurizer heaters is OPERABLE.~~	~~[18] months~~	8

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Two pressurizer safety valves shall be OPERABLE. ~~with lift settings ≥ [2475] psig and ≤ [2525] psig.~~ 2.2.2 3.1.1.3.A (2)

3.4B-23

<INSERT 3.4-18A>

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with ~~all~~ RCS ~~cold leg~~ temperature~~s~~ > ~~[283]~~ 262 °F. 3.1.1.3.A 3.1.1.3.B (3)

---NOTE---

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

1. Only one pressurizer safety valve is required to be OPERABLE in MODE 3, and in MODE 4 with RCS temperature > 262°F. 3.1.1.3.B (3)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable in MODES 1 or 2.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time of Condition A not met. OR Two pressurizer safety valves inoperable in MODES 1 or 2.	B.1 Be in MODE 3. ~~AND~~ ~~B.2~~ C.1 Be in MODE 4 with ~~any~~ RCS ~~cold leg~~ temperature ≤ ~~[283]~~ 262 °F.	6 hours ~~12~~ ~~24~~ 18 hours
C. Required pressurizer safety valve inoperable in MODE 3, or MODE 4 with RCS temperature > 262 F.		

3.1.1.3.A (3)
3.1.1.3.A
NA (36)
(3)
NA (3)

AND-249

<INSERT 3.4-18A>

3.4B-23

3. Not applicable in MODE 3, and in MODE 4 with RCS temperature > 262°F during hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III.

4. The provisions of LCO 3.0.3 are not applicable in MODE 3, and in MODE 4 with RCS temperature > 262°F.

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each required pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, as-left lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program

2

T4.1-2 #3

3.4B-03

Not used.

5

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Pressurizer Power Operated Relief Valve (PORV)

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

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. PORV inoperable.	A.1 Close block valve. AND A.2 Remove power from block valve.	1 hour 1 hour
B. Block valve inoperable.	B.1 Close block valve. AND B.2 Remove power from block valve.	1 hour 1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

5

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.11.1	---------------------NOTE-------------------- Not required to be performed with block valve closed in accordance with the Required Actions of this LCO. --- Perform one complete cycle of the block valve.	92 days
SR 3.4.11.2	Perform one complete cycle of the PORV.	18 months
[SR 3.4.11.3	Verify PORV and block valve are capable of being powered from an emergency power source.	18 months]

CTS

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3.4B-05

5

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.~~12~~ 11 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.~~12~~ 11 An LTOP System shall be OPERABLE with ~~a maximum of [one] makeup pump capable of injecting into the RCS,~~ high pressure injection (HPI) deactivated, and the core flood tanks (CFTs) isolated and: 3.1.2.10 3.1.2.9

<INSERT 3.4-22A> a. Pressurizer level ~~≤ [220] inches~~ such that the unit is not in a water solid condition and an OPERABLE ~~power operated~~ electromatic relief valve ~~(PORV)~~ (ERV) with a lift setpoint of ≤ ~~[555]~~ 460 psig; or 3.1.2.11 N/A

<INSERT 3.4-22B> b. The RCS depressurized and ~~an RCS vent of ≥ [0.75] square inch~~ the RCS open. 3.1.2.11 N/A

6

APPLICABILITY: MODE 4 when any RCS cold leg temperature is ≤ ~~[283]~~ 262 °F, 3.1.2.10
MODE 5,
MODE 6 when the reactor vessel head is on. N/A

~~NOTE~~
~~CFT isolation is only required when CFT pressure is greater than or equal to the maximum RCS pressure for the existing RCS temperature allowed by the pressure and temperature limit curves provided in the PTLR.~~

<INSERT 3.4-22A>

3.4B-05

		CTS
	-----NOTES-----	
1.	HPI deactivation and CFT isolation not applicable during ASME Section XI testing.	3.1.2.10 3.1.2.9
2.	HPI deactivation not applicable during fill and vent of the RCS.	3.1.2.10
3.	HPI deactivation not applicable during emergency RCS makeup.	3.1.2.10
4.	HPI deactivation not applicable during valve maintenance.	3.1.2.10
5.	CFT isolation is only required when CFT pressure is greater than or equal to the maximum RCS pressure for the existing RCS temperature allowed by the pressure and temperature curves provided in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."	3.1.2.9

<INSERT 3.4-22B>

3.4B-05

	-----NOTES-----	
1.	Pressurizer level not applicable as allowed by Emergency Operating Procedures.	3.1.2.11
2.	Pressurizer level not applicable during system hydrotest.	3.1.2.11

AWO-344

<INSERT 3.4-23A>

6

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. More than [one] makeup pump capable of injecting into the RCS.	A.1	--------NOTE--------- Two makeup pumps may be capable of injecting into the RCS during pump swap operation for ≤ 15 minutes. --------------------- Initiate action to verify only [one] makeup pump is capable of injecting into the RCS.	Immediately
B. HPI activated.	B.1	Initiate action to verify HPI deactivated.	Immediately
C. A CFT not isolated when CFT pressure is greater than or equal to the maximum RCS pressure for existing temperature allowed in the PTLR.	C.1	Isolate affected CFT.	1 hour
D. Required Action C.1 not met within the required Completion Time.	D.1	Increase RCS temperature to > 175°F.	12 hours
	OR		
	D.2	Depressurize affected CFT to < [555] psig.	12 hours

(continued)

<INSERT 3.4-23A>

CTS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. Pressurizer level not within required limits.	A.1 Restore pressurizer level to within required limits.	1 hour	NA
B. Required Action and associated Completion Time of Condition A not met.	B.1 Close and maintain closed the makeup control valve and its associated isolation valve. AND B.2 Stop RCS heatup.	12 hours 12 hours	NA
C. Required Electromatic Relief Valve (ERV) inoperable.	C.1 Restore required ERV to OPERABLE status.	1 hour	NA
D. Required Action and associated Completion Time of Condition C not met.	D.1 Reduce makeup tank level to ≤ 73 inches.	12 hours	NA
E. LCO requirements not met for any reason other than Condition A through Condition D.	E.1 Initiate action to restore compliance with LCO requirements.	Immediately	NA

AND-344

6

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Pressurizer level > [220] inches.	E.1	Restore pressurizer level to ≤ [220] inches.	1 hour
F. Required Action E.1 not met within the required Completion Time.	F.1	Close and maintain closed the makeup control valve and its associated isolation valve.	12 hours
	AND		
	F.2	Stop RCS heatup.	12 hours
G. PORV inoperable.	G.1	Restore PORV to OPERABLE status.	1 hour
H. Required Action G.1 not met within the required Completion Time.	H.1	Reduce makeup tank level to ≤ [70] inches.	12 hours
	AND		
	H.2	Deactivate low low makeup tank level interlock to the borated water storage tank suction valves.	12 hours

(continued)

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6

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. Pressurizer level > [220] inches. AND PORV inoperable. OR LTOP System inoperable for any reason other than Condition A through Condition H.	I.1 Restore LTOP System to OPERABLE status. OR I.2 Depressurize RCS and establish RCS vent of ≥ [0.75] square inch.	1 hour 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify a maximum of [one] makeup pump is capable of injecting into the RCS.	12 hours
SR 3.4.12.2	Verify HPI is deactivated.	12 hours
SR 3.4.12.3	Verify each CFT is isolated.	12 hours

(continued)

INSERT 3.4-25A

6

<INSERT 3.4-25A>

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	CTS
3.4B-10	SR 3.4.11.1	Verify pressurizer level does not represent a water solid condition.	30 minutes during RCS heatup and cooldown AND 12 hours	NA
3.4B-10	SR 3.4.11.2	Verify HPI is deactivated.	12 hours	NA
3.4B-10	SR 3.4.11.3	Verify each pressurized CFT is isolated.	12 hours	NA
	SR 3.4.11.4	----------------NOTE-------------------- Verification of locked, sealed, or otherwise secured open vent path(s) only required to be performed every 31 days. -- Verify OPERABLE pressure relief capability.	12 hours	NA
3.4B-11 3.4B-09	SR 3.4.11.5	Perform CHANNEL CALIBRATION of ERV opening circuitry.	18 months	T4.1-2, #17

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6

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.12.4	Verify pressurizer level is ≤ [220] inches.	30 minutes during RCS heatup and cooldown AND 12 hours
SR 3.4.12.5	Verify PORV block valve is open.	12 hours
SR 3.4.12.6	--------------------NOTE-------------------- Only required when complying with LCO 3.4.12.b. --- Verify RCS vent ≥ [0.75] square inch is open.	12 hours for unlocked open vent valve(s) AND 31 days for locked open vent valve(s)
SR 3.4.12.7	Perform CHANNEL FUNCTIONAL TEST for PORV.	Within [12] hours after decreasing RCS temperature to ≤ [283]°F AND 31 days thereafter

(continued)

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6

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.4.12.8 Perform CHANNEL CALIBRATION for PORV.	[18] months

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE; 3.1.6.3.a

b. 1 gpm unidentified LEAKAGE; 3.1.6.2

c. 10 gpm identified LEAKAGE; and 3.1.6.1 edit

~~d. 1 gpm total primary to secondary LEAKAGE through all steam generators (SGs); and~~ 1

~~e.~~ d. ~~[720]~~ 150 gallons per day primary to secondary LEAKAGE through any one SG. 3.1.6.3.b

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. RCS primary to secondary LEAKAGE not within limits ~~for reasons other than pressure boundary LEAKAGE~~.	A.1 Reduce LEAKAGE to within limits.	4 hours	3.1.6.3.b 1
B. RCS identified or unidentified LEAKAGE not within limits.	B.1 Reduce LEAKAGE to within limits.	18 hours	3.1.6.1 3.1.6.2
C ~~B~~. Required Action and associated Completion Time of Condition A or B not met. OR Pressure boundary LEAKAGE exists.	C ~~B~~.1 Be in MODE 3. AND C ~~B~~.2 Be in MODE 5.	6 hours 36 hours	3.1.6.1 3.1.6.2 3.1.6.3.a/b 3.1.6.1 3.1.6.2 3.1.6.3.a/b

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.4.13.1	after establishment -----NOTE----- Not required to be performed ~~in MODE 3 or 4~~ until 12 hours of steady state operation at or near operating pressure. ----- Verify RCS operational LEAKAGE is within limits by performance of an ~~Perform~~ RCS water inventory balance.	~~-----NOTE----- Only required to be performed during steady state operation -----~~ 72 hours	NA 11 Tbl 4.1-2 #6.a 27
SR 3.4.13.2	Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program.	In accordance with the Steam Generator Tube Surveillance Program	4.18.5.b

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.14 Leakage from each RCS PIV shall be within limits. 3.1.6.9 - TBl 3.5.1-1 - #1a & 1.b - N/A

APPLICABILITY: MODES 1, 2, and 3,
MODE 4, except valves in the decay heat removal (DHR) flow path when in, or during the transition to or from, the DHR mode of operation. N/A

3.4B-17

ACTIONS

-----NOTES----- N/A

1. Separate Condition entry is allowed for each flow path.
2. Enter applicable Conditions and Required Actions for systems made inoperable by an inoperable PIV. 3.1.6.9 -

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more flow paths with leakage from one or more RCS PIVs not within limit.	-----NOTE----- ~~Each valve used to satisfy Required Action A.1 and Required Action A.2 must have been verified to meet SR 3.4.14.1 and be on the RCS pressure boundary [or the high pressure portion of the system].~~ -----	(continued)

3.1.6.9 -

13 -

CTS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. (continued)	A.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed ~~manual,~~ deactivated automatic, ~~or~~ check valve. valve and one OPERABLE	4 hours	3.1.6.9 3.1.6.9 Note* 13
	~~AND~~		
	~~A.2 Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.~~	~~72 hours~~	13
	~~or~~		
	~~A.2 Restore RCS PIV to within limits.~~	~~72 hours~~	
C ~~B~~. Required Action and associated Completion Time for Condition A not met.	C ~~B~~.1 Be in MODE 3.	6 hours	3.1.6.9 + Tbl 3.5.1-1 Notes 1+5
	AND		
	C ~~B~~.2 Be in MODE 5.	36 hours	13
B ~~C~~. Decay Heat Removal (DHR) System autoclosure interlock function inoperable. a	B ~~C~~.1 Isolate the affected penetration by use of one closed manual or deactivated automatic valve.	4 hours	a Table 3.5.1-1 Notes 1+5

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	CTS
SR 3.4.14.1	-------------------NOTES------------------ ~~1.~~ Not required to be performed in MODES 3 and 4. ~~2. Not required to be performed on the RCS PIVs located in the DHR flow path when in the DHR mode of operation.~~ ~~3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided.~~ ---		NA 33 14
	Verify leakage from each RCS ~~PIV is~~ pressure isolation check valve, or pair of check valves, as applicable, is less than or equal to an equivalent ~~to ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm~~ the Allowable Leakage Limit identified below at ~~an RCS~~ a differential test pressure ≥ ~~[2215] psia and ≤ [2255] psia.~~ 150 psid.	In accordance with the Inservice Testing Program ~~or [18] months~~ AND Once prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months ~~[AND]~~ (continued)	Table 3.1.6.9 Notes (a.4) & (b) Table 4.1-2 #6b, Note 1 edit Table 4.1-2 #6b, Note 1 14

Pressure Isolation Check Valve(s)	Allowable Leakage Limit
DH-14A	≤ 5 gpm
DH-13A and DH-17	≤ 5 gpm total
DH-14B	≤ 5 gpm
DH-13B and DH-18	≤ 5 gpm total

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
~~SR 3.4.14.1 (continued)~~		~~Within 24 hours following valve actuation due to automatic or manual action or flow through the valve~~	14
SR 3.4.14.2	~~-----NOTE----- Not required to be met when the DHR System autoclosure interlock is disabled in accordance with LCO 3.4.12.~~ Verify DHR System autoclosure interlock prevents the valves from being opened with a simulated or actual high RCS pressure signal ~~≥ [425] psig~~.	[18] 2 months	15 Table 4.1-1, #30 Table 4.1-2, #11 24
SR 3.4.14.3	~~-----NOTE----- Not required to be met when the DHR System autoclosure interlock is disabled in accordance with LCO 3.4.12.~~ Verify DHR System autoclosure interlock causes the valves to close automatically with a simulated or actual high RCS pressure signal ~~≥ [600] psig~~ a. ≤ 340 psig for one valve, and b. ≤ 400 psig for the other valve	[18] 2 months	15 3.5.1.7 Tbl 4.1-1, #30 Tbl 4.1-2, #11

<INSERT 3.4-33A> 16

Tbl 4.1-1, #30
Tbl 4.1-2, #11

<INSERT 3.4-33A>

			CTS
SR 3.4.14.4	Verify DHR System autoclosure interlock prevents the valves from being opened with a simulated or actual Core Flood Tank isolation valve "not closed" signal.	18 months	NA T4.1-1, #30 T4.1-2, #11
SR 3.4.14.5	Verify DHR System autoclosure interlock causes the valves to close automatically with a simulated or actual Core Flood Tank isolation valve "not closed" signal.	18 months	NA T4.1-1, #30 T4.1-2, #11

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE: 3.1.6.7

reactor building

a. One containment sump monitor; and 30

reactor building

b. One containment atmosphere radioactivity monitor (gaseous or particulate).

APPLICABILITY: MODES 1, 2, 3, and 4. 3.1.6.7

ACTIONS 28

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. Required containment (reactor building) sump monitor inoperable.	------------NOTE------------ LCO 3.0.4 is not applicable. ----------------------------		NA 17
<INSERT 3.4-34A>	A.1 Perform SR 3.4.13.1.	Once per 24 hours	NA
	AND		
reactor building	A.2 Restore required containment sump monitor to OPERABLE status.	30 days	NA 30
B. Required containment (reactor building) atmosphere radioactivity monitor inoperable.	~~------------NOTE------------ LCO 3.0.4 is not applicable. ----------------------------~~		28 NA
	B.1.1 Analyze grab samples of the containment (reactor building) atmosphere.	Once per 24 hours	3.1.6.7 30
	OR		

(continued)

<INSERT 3.4-34A>

------------ NOTE --------------
Not required until 12 hours after establishment of steady state operation at or near operating pressure.

CTS

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	CTS
B. (continued) <INSERT 3.4-35A>	B.1.2	Perform SR 3.4.13.1.	Once per 24 hours	17 NA
	AND			
	B.2	Restore required ~~containment~~ reactor building atmosphere radioactivity monitor to OPERABLE status.	30 days	3.1.6.7 30
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	6 hours	3.1.6.7
	AND			
	C.2	Be in MODE 5.	36 hours	3.1.6.7
D. Both required monitors inoperable.	D.1	Enter LCO 3.0.3.	Immediately	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	CTS
SR 3.4.15.1	Perform CHANNEL CHECK of required ~~containment~~ reactor building atmosphere radioactivity monitor.	12 hours	Table 4.1-1 #28a 30
SR 3.4.15.2	Perform CHANNEL FUNCTIONAL TEST of required ~~containment~~ reactor building atmosphere radioactivity monitor.	92 days	Table 4.1-1 #28a

(continued)

<INSERT 3.4-35A>

------------- NOTE ---------------
Not required until 12 hours after establishment of steady state operation at or near operating pressure.

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.15.4	Perform CHANNEL CALIBRATION of required ~~containment~~ reactor building sump monitor.	~~[~~18~~]~~ months
SR 3.4.15.3	Perform CHANNEL CALIBRATION of required ~~containment~~ reactor building atmosphere radioactivity monitor.	~~[~~18~~]~~ months

Table 4.1-1 #45

30

Table 4.1-1 #28a

5
CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.~~16~~ 12 RCS Specific Activity

LCO 3.4.~~16~~ 12 The specific activity of the reactor coolant shall be: ~~within limits.~~ 3.1.4.1.b

a. ≤ 3.5 µCi/gm DOSE EQUIVALENT I-131; and 3.1.4.1.a

b. ≤ 72/Ē µCi/gm total. 19

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F. 3.1.4.1 Tbl 4.1-3 Note 7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. ~~DOSE EQUIVALENT I-131 > 1.0 µCi/gm.~~ Specific activity not within limits.	~~-----NOTE----- LCO 3.0.4 is not applicable.~~		
	~~A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1~~	~~Once per 4 hours~~	19
	~~AND~~		
	~~A.2~~ A.1 Restore ~~DOSE EQUIVALENT I-131~~ specific activity to within limit. (5)	~~48~~ 24 hours	3.1.4.1.c
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours	3.1.4.1.c
~~OR~~			
~~DOSE EQUIVALENT I-131 in unacceptable region of Figure 3.4.16-1.~~			19

(continued)

 5

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~C. Gross specific activity of the coolant not within limit.~~	~~C.1 Perform SR 3.4.16.2~~ ~~AND~~ ~~C.2 Be in MODE 3 with T_{avg} < 500°F.~~	~~4 hours~~ ~~6 hours~~

19

3.4B-16

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.4.~~16~~ 12.1	Verify reactor coolant gross specific activity ≤ ~~100~~ 72/Ē μCi/gm.	7 days	Table 4.1-3 #1.b
SR 3.4.~~16~~ 12.2	--------------------NOTE-------------------- Only required to be performed in MODE 1. -- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ ~~1.0~~ 3.5 μCi/gm.	14 days ~~AND~~ ~~Between 2 and 6 hours after THERMAL POWER change of ≥ 15% RTP within a 1 hour period~~	NA Tbl 4.1-3, #1.c 29

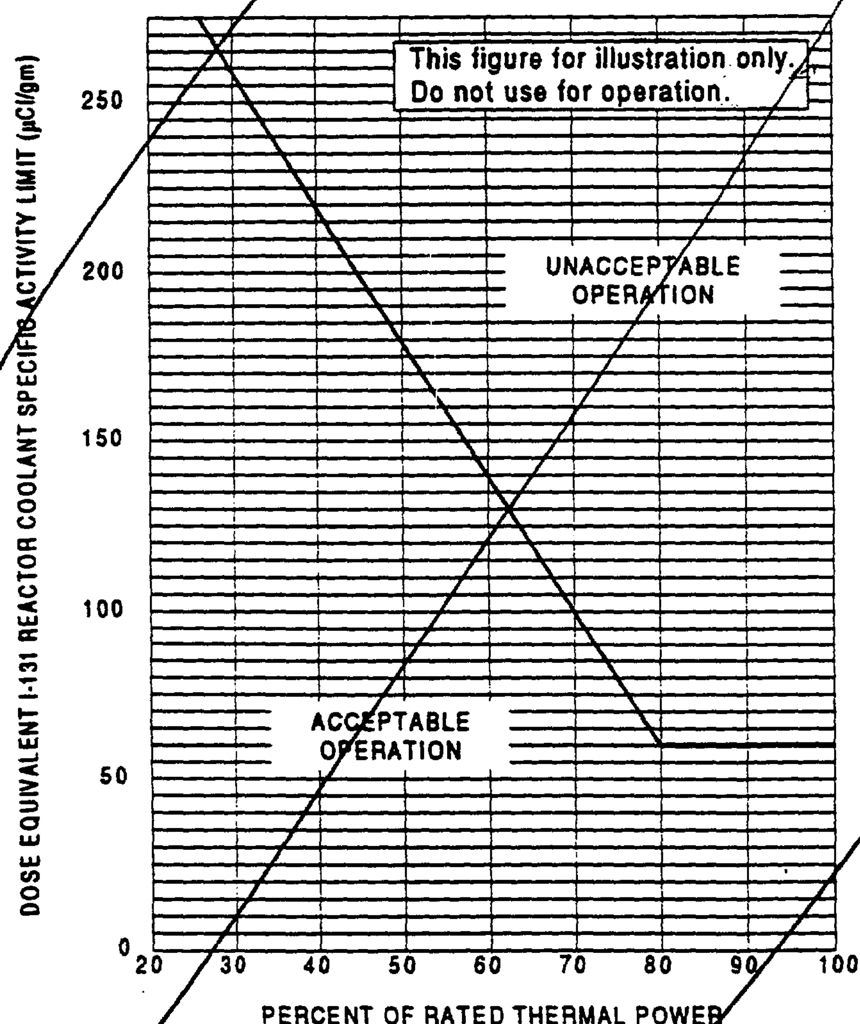
(continued)

H5

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.4.12.3	------------------NOTE------------------ Not required to be performed until 31 days after a minimum of 2 EFPD and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. --- Determine Ē.	 184 days	NA T4.1-3 #1.g



19

Figure 3.4.16-1 (page 1 of 1)
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity Limit Versus Percent of RATED THERMAL POWER With Reactor Coolant Specific Activity >1.0 µCi/gm DOSE EQUIVALENT I-131

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

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

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters, and their controls~~, and emergency power supplies~~. 8 Pressurizer safety valves ~~and pressurizer power operated relief valves (PORVs)~~ are addressed by LCO 3.4.10, "Pressurizer Safety Valves~~,~~" ~~and LCO 3.4.11, "Pressurizer Power Operated Relief Valve (PORV)," respectively~~. 5

The maximum water level limit has been established to ensure that a liquid to vapor interface exists to permit RCS pressure control during normal operation and proper pressure response for ~~anticipated design basis transients~~ or abnormalities. The water level limit thus serves two purposes:

a. Provides ~~Pressure~~ pressure control during normal operation ~~maintains subcooled reactor coolant in the loops and thus is in the preferred state for heat transport~~; and

b. Prevents the peak RCS pressure from exceeding the safety limit of 2750 psig during an abnormality ~~By restricting the level to a maximum, expected transient reactor coolant volume increases (pressurizer insurge) will not cause excessive level changes that could result in degraded ability for pressure control.~~

The maximum water level limit thus permits pressure control equipment to function as designed. The limit preserves the steam space during normal operation, ~~thus~~ so that both sprays and heaters can operate to maintain the design operating pressure. The level limit also prevents filling the pressurizer (water solid) ~~for anticipated design basis transients~~ during abnormalities, thus ensuring that pressure relief devices (~~PORVs~~ electromatic relief valve (ERV) or code safety valves) can control pressure by steam — edit

(continued)

BASES

BACKGROUND (continued)

(an abnormality)

relief rather than water relief. If the level limits were exceeded prior to a transient that creates a large pressurizer insurge volume leading to water relief, the maximum RCS pressure might exceed the design Safety Limit (SL) of 2750 psig or damage may occur to the PORVs or pressurizer code safety valves.

edit

10 -

The minimum water level limit has been established to ensure that the water level is above the minimum detectable level.

The pressurizer heaters are used to maintain a pressure in the RCS so reactor coolant in the loops is subcooled and thus in the preferred state for heat transport to the steam generators (SGs). This function must be maintained with a loss of offsite power. Consequently, the emphasis of this LCO is to ensure that the ~~essential power supplies and the associated~~ heaters are adequate to maintain pressure for RCS loop subcooling with an extended loss of offsite power.

Engineered Safeguards (ES) bus powered

8

A minimum required available capacity of {126} kW ensures that the RCS pressure can be maintained. Unless adequate heater capacity is available, reactor coolant subcooling ~~cannot be maintained indefinitely.~~ Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to loss of single phase natural circulation and decreased capability to remove core decay heat.

may not be maintained (although the pressure control provided by the high head high pressure injection pumps is an alternate method of maintaining subcooling).

10 -

APPLICABLE SAFETY ANALYSES

In MODES 1 and 2, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. No safety analyses are performed in lower MODES. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present.

Safety analyses presented in the FSAR do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

edit

The maximum level limit is ~~of prime interest for the loss of main feedwater (LOMFW) event. Conservative safety analyses assumptions for this event indicate that it produces the largest increase of pressurizer level caused by a moderate frequency event. Thus this event has been selected to~~

10 -

provided to prevent the peak RCS pressure from exceeding the safety limit of 2750 psig in the event of a rod withdrawal accident or a startup accident.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~establish the pressurizer water level limit.~~ Assuming proper response ~~action~~ by ~~emergency~~ systems, the level limit prevents water relief through the pressurizer safety valves. ~~Since prevention of water relief is a goal for abnormal transient operation, rather than an SL,~~ the value for pressurizer level ~~is nominal and is not adjusted for~~ instrument error.

reactor protection

INSERT B3.4-41A

is the safety analysis value. Therefore the implementing procedures must contain allowance for

10 edit

~~Evaluations performed for the design basis large break loss of coolant accident (LOCA), which assumed a higher maximum level than assumed for the LOMFW event, have been made. The higher pressurizer level assumed for the LOCA is the basis for the volume of reactor coolant released to the containment. The containment analysis performed using the mass and energy release demonstrated that the maximum resulting containment pressure was within design limits.~~

10

The requirement for emergency power supplies is based on ~~NUREG-0737~~ NUREG-0578 (Ref. 1), item 2.1.1. The intent is to ~~allow~~ maintain~~ing~~ the reactor coolant in a subcooled condition with natural circulation at hot, high pressure conditions for an ~~undefined, but~~ extended time period after a loss of offsite power. While loss of offsite power is an initial condition or coincident event assumed in many accident analyses, maintaining hot, high pressure conditions over an extended time period is not evaluated as part of SAR accident analyses.

edit

edit

In MODES 1 and 2,

In MODE 3 and MODE 4 above the LTOP enable temperature, the maximum pressurizer water limit satisfies Criterion 4 of 10 CFR 50.36,

~~The~~ the maximum pressurizer water level limit satisfies Criterion 2 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 2). Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-~~0737~~ 0578 (Ref. 1), is the reason for providing an LCO. Therefore, the pressurizer heaters satisfy Criterion 4 of 10 CFR 50.36 (Ref. 2).

35

3.4B-01

LCO

The LCO requirement for the pressurizer to be OPERABLE with a water level ~~≤ [290] inches~~ ≥ 45 inches and ≤ 320 inches ensures that a steam bubble exists. Limiting the maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

21

INSERT B3.4-41B

(continued)

<INSERT B3.4-41A>

If the level limits were exceeded prior to an abnormality that creates a large pressurizer insurge volume leading to water relief, the maximum RCS pressure might exceed the design SL of 2750 psig or damage may occur to the ERV or pressurizer code safety valves.

<INSERT B3.4-41B>

prior to criticality and that the indication of the level is above the minimum detectable level.

To be considered OPERABLE, the required heaters must be

BASES

LCO (continued)

The LCO requires a minimum of ~~[126]~~ kW of pressurizer heaters OPERABLE ~~and capable of being~~ powered from an ~~emergency power supply~~. ES bus. ~~As such, the LCO addresses both the heaters and the power supplies.~~ ~~The minimum heater capacity required is sufficient to maintain the system near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. The exact design value of [126] kW is derived from the use of nine heaters rated at 14 kW each.~~ The amount needed to maintain pressure is dependent on the insulation losses, which can vary due to tightness of fit and condition.

This provides assurance that sufficient heater capacity is available to provide RCS pressure control during a loss of off-site power.

8

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus Applicability has been designated for MODES 1 and 2. The Applicability is also provided for MODE 3 and, for pressurizer water level, for MODE 4 with RCS temperature ~~≥ [275]~~ > 262 °F. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbations, such as reactor coolant pump startup. The temperature of ~~[275]~~ 262 °F has been designated as the cutoff for applicability because LCO 3.4.~~12~~ 11, "Low Temperature Overpressure Protection (LTOP) ~~System~~," provides a requirement for pressurizer level, at or below ~~[275]°F~~ 262 °F. The LCO does not apply to MODE 5 with loops filled because LCO 3.4.~~12~~ 11 applies and provides adequate overpressure protection. The LCO does not apply to MODES 5 and 6 with partial loop operation.

This parameter value does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

21 edit
21
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In MODES 1, 2, and 3, there is the need to maintain the availability of pressurizer heaters capable of being powered from an emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. The Applicability is modified by a Note stating that the OPERABILITY requirements on pressurizer heaters do not apply in MODE 4. For MODE 4, 5, or 6, ~~it is not necessary~~ the need to control pressure (by heaters) to ensure loop subcooling for heat transfer is significantly reduced when the Decay Heat Removal

(continued)

BASES

APPLICABILITY (continued)

System is in service, and therefore the LCO is not applicable.

ACTIONS

A.1

With pressurizer water level ~~in excess of~~ outside the ~~maximum~~ limits action must be taken to restore pressurizer operation to within the bounds assumed in the analysis. This is done by restoring the pressurizer water level to within the limits. The 1 hour Completion Time is considered to be a reasonable time for ~~draining excess liquid~~ adjusting pressurizer level. 21

B.1 and B.2

If the water level cannot be restored, reducing core power constrains heat input effects that drive pressurizer insurge that could result from an anticipated transient. By shutting down the reactor and reducing reactor coolant temperature to at least MODE 3, the potential thermal energy of the reactor coolant mass for LOCA mass and energy releases is reduced. edit

Six hours is a reasonable time based upon operating experience to reach MODE 3 from full power in an orderly manner and without challenging ~~plant~~ unit systems ~~and operators~~. Further pressure and temperature reduction to MODE 4 with RCS temperature ≤ ~~[275]~~ 262 °F places the ~~plant~~ unit into a MODE where the LCO is not applicable. The ~~12~~ 24 hour Completion Time to reach the nonapplicable MODE is reasonable based upon operating experience. edit 36

unit

ANO-249

C.1

If ~~the [emergency] power supplies to the heaters are not capable of providing [126] kW, or~~ the required pressurizer heaters are inoperable, restoration is required in 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand caused by loss of offsite power will not occur in this period. Pressure control may be maintained during this time using ~~normal station~~ non-ES bus powered heaters. 8

(continued)

BASES

ACTIONS (continued)

D.1 and D.2

are not met

The Required Action and associated

If ~~pressurizer heater capability cannot be restored within~~ ~~the allowed~~ Completion Time ~~of Required Action C.1~~, the unit ~~plant~~ must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit ~~plant~~ must be brought to MODE 3 within 6 hours and to MODE 4 within the following 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit ~~plant~~ systems. Similarly, the Completion Time of 12 hours to reach MODE 4 is reasonable based on operating experience to achieve power reduction from full power conditions in an orderly manner and without challenging unit ~~plant~~ systems.

edit

edit

edit

edit

edit

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that ~~during steady state operation,~~ pressurizer water level is maintained below the ~~nominal~~ upper limit to provide a minimum space for a steam bubble. ~~The Surveillance is performed by observing the indicated level.~~ The values specified for pressurizer level do not contain an allowance for instrument error. Therefore additional allowances for instrument uncertainties must be provided in the implementing procedures. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess the level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level ~~indications~~.

edit

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edit

SR 3.4.9.2

The SR requires ~~the power supplies are capable of producing the minimum power and the associated~~ sufficient pressurizer heaters ~~are~~ verified to be ~~at their design rating~~ capable of providing the required capacity. (This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance.) The Frequency of [18] months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

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(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~SR 3.4.9.3~~

~~This SR is not applicable if the heaters are permanently powered by 1E power supplies.~~

~~This Surveillance demonstrates that the heaters can be manually transferred to, and energized by, emergency power supplies. The Frequency of [18] months is based on a typical fuel cycle and is consistent with similar verifications of emergency power.~~

8

REFERENCES

1. NUREG-~~0737, November 1980.~~ 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," July 1979.
2. 10 CFR 50.36.

8 edit

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The purpose of the two spring loaded pressurizer safety valves is to provide RCS overpressure protection. (Ref. 1) Operating in conjunction with the Reactor Protection System (RPS), two valves are used to ensure that the Safety Limit (SL) of 2750 psig is not exceeded for analyzed transients during operation in MODES 1 and 2. One ~~Two~~ safety valves ~~are used~~ is required for MODE 3 and portions of MODE 4. For the remainder of MODE 4, MODE 5, and MODE 6 with the reactor head on, overpressure protection is provided by operating procedures and LCO 3.4.~~12~~ 11, "Low Temperature Overpressure Protection (LTOP) System." ~~For these conditions, the American Society of Mechanical Engineers (ASME) requirements are satisfied with one safety valve.~~ edit 3 5 18

The self actuated pressurizer safety valves are designed in accordance with the requirements set forth in the ASME Boiler and Pressure Vessel Code, Section III (Ref. ~~1~~ 2). The required lift pressure is 2500 psig ~~± 1%~~ +1%, -3%. The safety valves discharge steam from the pressurizer to a quench tank located in the ~~containment~~ reactor building. The discharge flow is indicated by acoustic flow monitoring devices, by an increase in temperature downstream of the safety valves, and by an increase in the quench tank temperature, pressure, and level. edit 2 2 9 edit

The upper and lower as-left pressure limits are based on the ± 1% tolerance requirement for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the ASME pressure limit could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

~~All accident analyses in the FSAR that require safety valve actuation assume operation of both pressurizer safety valves to limit increasing reactor coolant pressure.~~ The overpressure protection analysis (Ref. 3) is also based on operation of both safety valves and assumes that the valves open at the high range of the setting (2500 psig system design pressure plus 1%). <INSERT B 3.4-47A> These valves must accommodate pressurizer insurges that could occur during a startup, rod withdrawal, or ejected rod, ~~loss of main feedwater, or main feedwater line break accident.~~ event. The startup accident establishes the minimum safety valve capacity. The startup accident is assumed to occur at ~~< 15%~~ low power. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this Specification is required to ensure that the accident analysis and design basis calculations remain valid.

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3
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edit

In MODES 1 and 2, Pressurizer safety valves satisfy Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 5).

In MODE 3 and MODE 4 above the LTOP enable temperature, the pressurizer safety valves satisfy Criterion 4 of 10CFR50.36.

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LCO

The two pressurizer safety valves are set to open at the RCS design pressure (2500 psig) and within the ~~ASME~~ specified tolerance to avoid exceeding the maximum RCS design pressure SL, to maintain accident analysis assumptions and to comply with ASME Code requirements. The upper and lower as-left pressure tolerance limits are based on the ± 1% tolerance requirements (Ref. 4) for lifting pressures above 1000 psig. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or both valves could result in exceeding the SL if a transient were to occur.

edit
2
2

<INSERT B 3.4-47B>

The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

<INSERT FROM APPLICABILITY BASES>

<INSERT B 3.4-47C>

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APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP ~~cut in~~ enable temperature, OPERABILITY of ~~two~~ pressurizer safety valves is required ~~because the combined~~ to ensure adequate relieving capacity is ~~required~~ available to keep reactor coolant pressure below 110% of its design value during

3

(continued)

<INSERT B3.4-47A>

One pressurizer code safety valve is capable of preventing overpressurization in MODE 3 and in MODE 4 with RCS temperature > 262°F since its relieving capacity is greater than that required by the sum of the available heat sources, i.e., pump energy, pressurizer heaters, and reactor decay heat (Refs. 1 and 4).

<INSERT B3.4-47B>

3.4B-23

The LCO is modified by four Notes. Note 1 states that in MODE 3 and MODE 4 with RCS temperature above 262°F, only one pressurizer safety valve is required to be OPERABLE. In this condition, one pressurizer safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than the sum of the available heat sources.

<INSERT B3.4-47C>

3.4B-23

Note 3 states that the LCO is not applicable in MODE 3, and in MODE 4 with RCS temperature > 262°F during hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III. During hydrostatic tests, the code safeties must be gagged to prevent them from relieving at the target test pressure. RCS pressure is carefully observed and compensatory measures are in place to provide assurance that the pressure is appropriately controlled during the performance of hydrostatic tests.

Note 4 states that the provisions of LCO 3.0.3 are not applicable in MODE 3, and in MODE 4 with RCS temperature > 262°F. In the event no code safety valve is OPERABLE in this MODE, the Required Actions ensure that the RCS is placed in a condition in which the ERV is capable of relieving any potential LTOP pressure transient.

The parameter value (262°F) does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

BASES

APPLICABILITY (continued)

certain accidents. ~~MODE 3 and portions of MODE 4 are conservatively included, although the listed accidents may not require both safety valves for protection.~~ 3

in MODE 5, nor in MODE 6 when the reactor vessel head is on

The LCO is not applicable in MODE 4 with ~~when any~~ RCS ~~cold leg~~ temperature ~~≤~~ ≤ ~~[283]~~ 262 °F ~~and MODE 5~~ because LTOP protection is provided. Overpressure protection is not required in MODE 6 with the reactor vessel head ~~detensioned~~ removed. 3 edit

<INSERT B3.4-48A>

MOVE TO LCO BASES

~~The~~ Note allows entry into MODE~~S~~ 3 ~~and 4~~ potentially into MODE 4 with RCS temperature > 262 F with the lift settings outside the ~~LCO~~ limits. This permits testing ~~and examination~~ of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The ~~[36]~~ hour exception is based on an 18 hour outage time for each of the two valves. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe. 4 3

ACTIONS

A.1

With one pressurizer safety valve inoperable in MODES 1 and 2, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS overpressure protection system. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the RCPB. 3

B.1 ~~and B.2~~

If the Required Action and associated ~~cannot be met within the required~~ Completion Time of Condition A are not met, or if both pressurizer safety valves are inoperable in MODES 1 and 2, the unit ~~plant~~ must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit ~~plant~~ must be brought to at least MODE 3 within 6 hours. ~~and to MODE 4 with any RCS cold leg temperature ≤ [283] °F within 12 hours.~~ The 6 hours allowed is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit ~~plant~~ systems. ~~Similarly~~, the ~~12~~ 18 hours allowed is 3

<INSERT from Pg B3.4-49>

<INSERT B3.4-48B>

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(continued)

<INSERT B3.4-48A>

The parameter value (262°F) does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

<INSERT B3.4-48B>

C.1

With the required pressurizer code safety valve inoperable, the RCS overpressure protection capability is significantly reduced and an overpressure event could challenge the integrity of the RCPB. Therefore, the unit must be placed in a condition in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 4 with RCS temperature at or below the LTOP

ANO-249 enable temperature within 18 hours.

BASES

ACTIONS

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

a low temperature within

reasonable, based on unit operating experience, to reach MODE 4 without challenging ~~plant~~ systems. With ~~any~~ RCS ~~cold leg~~ temperature at or below 262 ~~[283]~~ °F, overpressure protection is provided by LTOP. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by two pressurizer safety valves.

MOVE to B.1 Pg B3.4-48

edit 3

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 6 ~~2~~), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer (Ref. 7) safety valve setpoint is +1%, -3% ~~± [3]%~~ for OPERABILITY~~;~~ however, the valves are reset to ± 1% during the Surveillance to allow for drift.

2

REFERENCES

1. SAR, Section 4.2.4.
2. ASME, Boiler and Pressure Vessel Code, Section III, ~~Section XI.~~ Article 9, Summer 1968.
3. SAR, Section 4.3.8.
4. SAR, Section 4.3.11.4.
5. 10 CFR 50.36.
6. ASME, Boiler and Pressure Vessel Code, Section XI.
7. ASME/ANSI, Operations and Maintenance Codes (OM), Part 10, 1987, Part 10 Addenda, 1988, and Part 1, 1987.

edit 35 edit 2

3.4B-03

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Pressurizer Power Operated Relief Valve (PORV)

BASES

BACKGROUND

The pressurizer is equipped with three devices for pressure relief functions: two American Society of Mechanical Engineers (ASME) pressurizer safety valves that are safety grade components and one PORV that is not a safety grade device. The PORV is an electromatic pilot operated valve that is automatically opened at a specific set pressure when the pressurizer pressure increases and is automatically closed on decreasing pressure. The PORV may also be manually operated using controls installed in the control room.

An electric motor operated, normally open, block valve is installed between the pressurizer and the PORV. The function of the block valve is to isolate the PORV. Block valve closure is accomplished manually using controls in the control room and may be used to isolate a leaking PORV to permit continued power operation. Most importantly, the block valve is to be used to isolate a stuck open PORV to isolate the resulting small break loss of coolant accident (LOCA). Closure terminates the RCS depressurization and coolant inventory loss.

The PORV, its block valve, and their controls are powered from normal power supplies but are also capable of being powered from emergency supplies. Power supplies for the PORV are separate from those for the block valve. Power supply requirements are defined in NUREG-0737, Paragraph III, G.1 (Ref. 1).

The PORV setpoint is above the high pressure reactor trip setpoint and below the opening setpoint for the pressurizer safety valve as required by IE Bulletin 79-05B (Ref. 2). The purpose of the relationship of these setpoints is to limit the number of transient pressure increase challenges that might open the PORV, which, if opened, could fail in the open position. A pressure increase transient would cause a reactor trip, reducing core energy, and for many expected transients, prevent the pressure increase from reaching the PORV setpoint. The PORV setpoint thus limits the frequency of challenges from transients and limits the possibility of a small break LOCA from a failed open PORV.

(continued)

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BASES

BACKGROUND (continued)

Placing the setpoint below the pressurizer safety valve opening setpoint reduces the frequency of challenges to the safety valves, which, unlike the PORV, cannot be isolated if they were to fail open. The PORV setpoint is therefore important for limiting the possibility of a small break LOCA.

The primary purpose of this LCO is to ensure that the PORV, its setpoint, and the block valve are operating correctly so the potential for a small break LOCA through the PORV pathway is minimized, or if a small break LOCA were to occur through a failed open PORV, the block valve could be manually operated to isolate the path.

The PORV may be manually operated to depressurize the RCS as deemed necessary by the operator in response to normal or abnormal transients. The PORV may be used for depressurization when the pressurizer spray is not available; a condition that would be encountered during loss of offsite power. Steam generator tube rupture (SGTR) is one event that may require use of the PORV if the sprays are unavailable.

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

The PORV functions as an automatic overpressure device and limits challenges to the safety valves. Although the PORV acts as an overpressure device for operational purposes, safety analyses [do not take credit for PORV actuation, but] do take credit for the safety valves.

The PORV also provides low temperature overpressure protection (LTOP) during heatup and cooldown. LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," addresses this function.

APPLICABLE SAFETY ANALYSES

The PORV small break LOCA break size is bounded by the spectrum of piping breaks analyzed for plant licensing. Because the PORV small break LOCA is located at the top of the pressurizer, the RCS response characteristics are different from RCS loop piping breaks; analyses have been performed to investigate these characteristics.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The possibility of a small break LOCA through the PORV is reduced when the PORV flow path is OPERABLE and the PORV opening setpoint is established to be reasonably remote from expected transient challenges. The possibility is minimized if the flow path is isolated.

The PORV opening setpoint has been established in accordance with Reference 2. It has been set so expected RCS pressure increases from anticipated transients will not challenge the PORV, minimizing the possibility of a small break LOCA through the PORV.

Overpressure protection is provided by safety valves, and analyses do not take credit for the PORV opening for accident mitigation.

Operational analyses that support the emergency operating procedures utilize the PORV to depressurize the RCS for mitigation of SGTR when the pressurizer spray system is unavailable (loss of offsite power). FSAR safety analyses for SGTR have been performed assuming that offsite power is available and thus pressurizer sprays (or the PORV) are available.

The PORV and its block valve do not satisfy any specific Criterion of the NRC Policy Statement. This Specification was evaluated using insights gained from reviewing representative probabilistic risk assessments. The PORV and its block valve are deemed important to risk.

LCO

The LCO requires the PORV and its associated block valve to be OPERABLE. The block valve is required to be OPERABLE so it may be used to isolate the flow path if the PORV is not OPERABLE. If the block valve is not OPERABLE, the PORV may be used for temporary isolation.

APPLICABILITY

In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. A likely cause for PORV LOCA is a result of pressure increase transients that cause the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the

(continued)

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BASES

APPLICABILITY (continued)

RCS pressure to increase to the PORV opening setpoint. Pressure increase transients can occur any time the steam generators are used for heat removal. The most rapid increases will occur at higher operating power and pressure conditions of MODES 1 and 2.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the applicability is pertinent to MODES 1, 2, and 3. The LCO is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for LTOP in MODES 4, 5, and 6 with the reactor vessel head in place. LCO 3.4.12 addresses the PORV requirements in these MODES.

ACTIONS

A.1 and A.2

With the PORV inoperable, the PORV must be restored or the flow path isolated within 1 hour. The block valve should be closed and power must be removed from the block valve to reduce the potential for inadvertent PORV opening and depressurization.

B.1 and B.2

If the block valve is inoperable, it must be restored to OPERABLE status within 1 hour. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to close the block valve and remove power within 1 hour rendering the PORV isolated. The 1 hour Completion Times are consistent with an allowance of some time for correcting minor problems, restoring the valve to operation, and establishing correct valve positions and restricting the time without adequate protection against RCS depressurization.

(continued)

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BASES

ACTIONS (continued)

C.1 and C.2

If the Required Action and associated Completion Time cannot be met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The 6 hours allowed is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. Similarly, the 12 hours allowed is reasonable, based on operating experience, to reach MODE 4 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Block valve cycling verifies that it can be closed if needed. The basis for the Frequency of 92 days is ASME Code, Section XI (Ref. 3). Block valve cycling, as stated in the Note, is not required to be performed when it is closed for isolation; cycling could increase the hazard of an existing degraded flow path.

SR 3.4.11.2

PORV cycling demonstrates its function. The Frequency of 18 months is based on a typical refueling cycle and industry accepted practice.

SR 3.4.11.3

This Surveillance is not required for plants with permanent 1E power supplies to the valves.

This SR demonstrates that emergency power can be provided and is performed by transferring power from the normal supply to the emergency supply and cycling the valves. The Frequency of 18 months is based on a typical refueling cycle and industry accepted practice.

(continued)

BASES (continued)

REFERENCES

1. NUREG-0737, Paragraph III, G.1, November 1980.
2. NRC IE Bulletin 79-05B, April 21, 1979.
3. ASME, Boiler and Pressure Vessel Code, Section XI.

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.~~12~~ 11 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

~~[Reviewer's Note: For plants for which the NRC has approved LTOP setpoints based on non-10 CFR 50, Appendix G, methodology, as allowed in NRC Generic Letter 88-11, the following Bases must be revised accordingly.]~~ edit

prevent over edit

The LTOP ~~System~~ controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) requirements of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component ~~for providing~~ such protection. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," provides the allowable combinations for ~~operational~~ pressure and temperature during cooldown, shutdown, and heatup to keep from violating the Reference 1 limits. 6

as modified by approved exemptions.

requiring

The reactor vessel material is less tough at reduced temperatures than at normal operating temperature. Also, as vessel neutron irradiation accumulates, the material becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure must be maintained low when temperature is low and ~~must~~ may be increased only as temperature is increased. edit

Operational maneuvering during cooldown, heatup, or any anticipated operational occurrence must be controlled to not violate LCO 3.4.3. Exceeding these limits could lead to brittle fracture of the reactor vessel. LCO 3.4.3 presents requirements for administrative control of RCS pressure and temperature to prevent exceeding the P/T limits.

to be OPERABLE with the

electromatic

E

This LCO provides RCS overpressure protection in the applicable MODES by ensuring an adequate pressure relief capacity and a minimum coolant addition capability. The pressure relief capacity requires ~~either~~ the (power operated) relief valve (~~P~~ERV) lift setpoint ~~to be~~ reduced and pressurizer coolant level at or below a maximum limit, or the RCS depressurized and with an RCS vent of sufficient size to handle the limiting transient ~~during~~ LTOP. 6 edit

for the RCS pressure,

(continued)

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5

BASES

6

BACKGROUND (continued)

The LTOP approach to protecting the vessel by limiting coolant addition capability ~~allows a maximum of [one] makeup pump, and~~ requires deactivating HPI, and isolating the core flood tanks (CFTs).

Should ~~more than [one]~~ an HPI pump inject on an HPI actuation, the pressurizer level and E ~~P~~ORV or another RCS vent ~~cannot~~ may not prevent overpressurizing the RCS. ~~Even with only one HPI pump OPERABLE, the vent cannot prevent RCS overpressurization.~~

INSERT B 3.4-57A

~~The pressurizer level limit~~ provides a compressible vapor space or cushion (either steam or nitrogen) that can accommodate a coolant insurge and prevent a rapid pressure increase, allowing the operator time to stop the increase. The E ~~P~~ORV, with reduced lift setting, or the RCS vent is the overpressure protection device that acts as backup to the operator in terminating an increasing pressure event.

With HPI deactivated, the ability to provide RCS coolant addition is restricted. To ~~balance the possible need~~ allow for coolant addition, the LCO does not require the makeup ~~System~~ function to be deactivated. Due to the lower pressures associated with the LTOP MODES and the expected decay heat levels, the makeup ~~System~~ can provide flow ~~with the OPERABLE makeup pump~~ through the makeup control valve.

E ~~P~~ORV Requirements

As designed for the LTOP ~~System~~, ~~each PORV~~ the ERV is signaled to open if the RCS pressure ~~approaches~~ reaches a limit set in the LTOP actuation circuit. The LTOP actuation circuit monitors RCS pressure and determines when an overpressure condition is approached. When the monitored pressure meets or exceeds the setting, the E ~~P~~ORV is signaled to open. Maintaining the lowered setpoint ~~within the limits of the LCO~~ ensures the Reference 1 limits will be met in any event analyzed for LTOP.

~~When a PORV is opened in an increasing pressure transient, the release of coolant causes the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.~~

(continued)

<INSERT B3.4-57A>

As indicated in Reference 3, the deactivation of HPI injection capability, along with the LTOP alarms, provides sufficient basis for excluding the inadvertent actuation of HPI as a design basis event. Additionally, the CFT controls preclude the inadvertent mass input from the CFT. Finally, maintaining the pressurizer level to prevent operation in a water solid condition with the RCS pressure boundary intact

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BASES

BACKGROUND (continued)

RCS Vent Requirements

Once the RCS is depressurized, adequate pressure relief capability may be provided by a vent ~~exposed~~ path to the ~~containment~~ reactor building atmosphere ~~will maintain the RCS at ambient containment pressure in an RCS overpressure transient, if the relieving requirements of the maximum credible LTOP transient do not exceed the capabilities of the vent. Thus, the vent path must be~~ which is capable of relieving the flow of the limiting LTOP transient and maintaining pressure below P/T limits. The required vent capacity may be provided by one or more vent paths. Acceptable RCS vent paths include any of the following: edit 6

~~For an RCS vent to meet the flow capacity, it requires~~ E removing a pressurizer safety valve, locking the PORV in the open position and disabling its block valve in the open position, or similarly establishing a vent by ~~opening an RCS vent valve~~. INSERT B3.4-58A The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open. 6 6

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. ~~3~~ 5 4, 5, 6, and 7) demonstrate that the reactor vessel can be adequately protected against overpressurization transients during shutdown. ~~In MODES 1, 2, and 3, and in MODE 4 with RCS temperature exceeding [283]°F, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At nominally [283]°F and below, overpressure prevention falls to an OPERABLE PORV and a restricted coolant level in the pressurizer, or to a depressurized RCS and a sufficient size RCS vent. Each of these means has a limited overpressure relief capability.~~ INSERT B3.4-58B 6 6

The actual temperature at which the allowable pressure ~~in the P/T limit curve~~ falls below the pressurizer safety valve setpoint increases as vessel material ~~toughness~~ ductility decreases due to neutron embrittlement. INSERT B3.4-58C Each time the P/T limit curves are revised, the LTOP ~~System will be~~ is re-evaluated to ensure that its functional requirements can still be met. ~~with the PORV and pressurizer level method or the depressurized and vented RCS condition.~~ The ERV setpoint is revised as necessary. 6

~~Transients that are capable of overpressurizing the RCS have been identified and evaluated. These transients relate to either mass input or heat input: actuating the HPI System, discharging the CFTs, energizing the pressurizer heaters, failing the makeup control valve open, losing decay heat~~ INSERT B3.4-58D 6

(continued)

<INSERT B3.4-58A>

removing a steam generator (SG) primary manway, removing a SG primary hand hole cover, removing all control rod drive top closure assemblies (excluding reactor vessel level probe), or removing a pressurizer manway.

<INSERT B3.4-58B>

The pressure and temperature limits are derived from fracture mechanics analyses. Transients are then evaluated to determine a required ERV setpoint and other unit conditions that will ensure that the P/T limits are not exceeded.

Fracture mechanics analyses (using the safety margins of Reference 8) established the temperature of LTOP Applicability at 262°F. Above this temperature, the pressurizer safety valves provide the reactor vessel overpressure protection.

<INSERT B3.4-58C>

P/T limits are periodically determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3 discuss these examinations. For the current limits, vessel materials are assumed to have a neutron irradiation accumulation equivalent to 31 effective full power years (EFPYs) of operation.

<INSERT B3.4-58D>

at low temperature result in either excessive mass input or excessive heat input. Such transients include: HPI actuation, CFT discharge, energization of the pressurizer heaters, failing the makeup control valve open, loss of decay heat removal, starting a reactor coolant pump (RCP) with a large temperature mismatch between the primary and secondary coolant systems, and addition of nitrogen to the pressurizer. Without controls, HPI actuation and CFT discharge would be transients that result in exceeding P/T limits within the 10 minute period in which time no operator action can be assumed to take place. For the remaining events, operator action after that time precludes overpressurization.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

~~removal, starting a reactor coolant pump (RCP) with a large temperature mismatch between the primary and secondary coolant systems, and adding nitrogen to the pressurizer.~~

~~HPI actuation and CFT discharge are the transients that result in exceeding P/T limits within < 10 minutes, in which time no operator action is assumed to take place. In the rest, operator action after that time precludes overpressurization. The analyses demonstrate that the time allowed for operator action is adequate, or the events are self limiting and do not exceed P/T limits.~~ 6

INSERT B3.4-59A

~~The following are required during the LTOP MODES to ensure that transients do not occur, which either of the LTOP overpressure protection means cannot handle:~~

~~a. Deactivating all but [one] makeup pump;~~

~~b. Deactivating HPI; and~~

~~c. Immobilizing CFT discharge isolation valves in their closed positions.~~

~~The Reference 3 analyses demonstrate the PORV can maintain RCS pressure below limits when only one makeup pump is actuated. Consequently, the LCO allows only [one] makeup pump to be OPERABLE in the LTOP MODES.~~

~~Since the PORV cannot do this for one HPI pump and the RCS vent cannot do this for even one pump, the LCO also require: the HPI actuation circuits deactivated and the CFTs isolated.~~

~~The isolated CFTs must have their discharge valves closed and the valve power breakers fixed in their open positions. The analyses show the effect of CFT discharge is over a narrower RCS temperature range (175°F and below) than that of the LCO ([283]°F and below).~~

~~Fracture mechanics analyses established the temperature of LTOP Applicability at [283]°F. Above this temperature, the pressurizer safety valves provide the reactor vessel pressure protection. The vessel materials were assumed to have a neutron irradiation accumulation equal to 21 effective full power years (EFPYs) of operation.~~ 6

(continue

<INSERT B3.4-59A>

This specification prevents exceeding the P/T limits by: 1) limiting the capability for rapid mass input to the RCS; and 2) ensuring that adequate vent capability exists to accommodate inadvertent mass or energy addition to the RCS. Pressurizer level is also limited to ensure that increasing pressure during a transient will be slow enough to preclude exceeding pressure limits within the 10 minutes assumed to be required for operator action to mitigate the transient. Mass input into the system is limited by disabling HPI (with specific exceptions) and by deactivating pressurized CFT discharge isolation valves in the closed position with their power breakers open (with specific exceptions). The analyses demonstrate that HPI transients involving one HPI pump can be accommodated by the ERV without exceeding the maximum allowable pressure.

The ERV setpoint is determined by modeling LTOP performance assuming the most limiting LTOP transient of a makeup control valve failing open. Pressure overshoot beyond the setpoint resulting from signal processing and valve stroke times is considered. The resulting ERV setpoint ensures the reference 1 limits will not be exceeded.

Vent capability is required to ensure that the maximum allowable pressure is not exceeded in the event of full opening of the makeup control valve while one makeup pump is running. Acceptable vent paths have adequate capacity at a system pressure of 100 psig which is less than the maximum RCS pressure on the P/T limit curve in LCO 3.4.3.

AND-3.4.1

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~This LCO will deactivate the HPI actuation when the RCS temperature is ≤ [283]°F. The consequences of a small break LOCA in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 4 and 5), requirements by having a maximum of [one] makeup pump OPERABLE.~~ 6

~~Reference 3 contains the acceptance limits that satisfy the LTOP requirements. Any change to the RCS must be evaluated against these analyses to determine the impact of the change on the LTOP acceptance limits.~~

~~PORV Performance~~

~~The fracture mechanics analyses show that the vessel is protected when the PORV is set to open at ≤ [555] psig. The setpoint is derived by modeling the performance of the LTOP System, assuming the limiting allowed LTOP transient of uncontrolled HPI actuation of one pump. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoint at or below the derived limit ensures the Reference 1 limits will be met.~~

~~The PORV setpoint will be re-evaluated for compliance when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to embrittlement induced by neutron irradiation. Revised P/T limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3 discuss these examinations.~~ 6

The E PORV is ~~considered~~ an active component. Therefore, its failure represents the worst case ~~LTOP~~ single active failure of LTOP features. 6

~~Pressurizer Level Performance~~

~~Analyses of operator response time show that the pressurizer level must be maintained ≤ [220] inches to provide the 10 minute action time for correcting transients.~~ 6

(continued)

(5)

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BASES

APPLICABLE SAFETY ANALYSES

~~Pressurizer Level Performance (continued)~~

~~The pressurizer level limit will also be re-evaluated for compliance each time P/T limit curves are revised based on the results of the vessel material surveillance.~~

~~RCS Vent Performance~~

~~With the RCS depressurized, analyses show a vent of [0.75] square inches is capable of mitigating the transient resulting from full opening of the makeup control valve while the makeup pump is providing RCS makeup. The capacity of a vent this size is greater than the flow resulting from this credible transient at 100 psig back pressure, which is less than the maximum RCS pressure on the P/T limit curve in LCO 3.4.3.~~

~~The RCS vent size will also be re-evaluated for compliance each time P/T limit curves are revised based on the results of the vessel material surveillance.~~

(6)

The other vent ~~is~~ paths are passive and ~~is~~ not subject to active failure.

The LTOP ~~System~~ satisfies Criterion 2 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 9).

(35)

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LCO

The LCO requires an LTOP System OPERABLE with a limited coolant input capability and a pressure relief capability. To limit coolant input, the LCO requires ~~only [one] makeup pump OPERABLE,~~ the HPI deactivated, and the CFT discharge isolation valves closed and ~~immobilized.~~ deactivated. For pressure relief, ~~it requires either the pressurizer coolant at or below a maximum level and the PORV OPERABLE with a lift setting at the LTOP limit or the RCS depressurized and a vent established.~~

INSERT B3.4-61A

~~The pressurizer is OPERABLE with a coolant level ≤ [220] inches.~~

~~The PORV is OPERABLE when its block valve is open, its lift setpoint is set at ≤ [555] psig and testing has proven its ability to open at that setpoint, and motive power is available to the two valves and their control circuits.~~

(6)

(continued)

<INSERT B3.4-61A>

3.4B-05 the LCO requires the pressurizer coolant level to be below a level which represents a water solid condition, and the ERV OPERABLE with a lowered lift setting or the RCS depressurized and a vent established.

HPI deactivation requires that the motor operated valves be closed and the opening control circuits for the motor operators disabled. CFT isolation requires the CFT discharge valves to be closed and the circuit breakers for the motor operators to be opened.

The HPI deactivation and CFT isolation requirements are modified by five Notes. Note 1 indicates that the requirements are not applicable during ASME Section XI testing. This exception provides for required testing during these shutdown conditions rather than at power when the HPI and CFTs are required to be OPERABLE for the ECCS function. Note 2 indicates that the requirements are not applicable for the HPI deactivation during fill and vent of the RCS. The HPI pumps are used for this normal makeup function and must be available. Specific procedural controls are provided to prevent overpressurization during this activity. Note 3 indicates that the requirements are not applicable for the HPI deactivation during emergency RCS makeup. This exception is necessary to enhance the response capability to a loss of decay heat removal event without violating the TS (Ref. 10). Note 4 indicates that the requirements are not applicable for the HPI deactivation during valve maintenance. This exception allows maintenance to be performed during these shutdown conditions rather than at power when the HPI is required to be OPERABLE for the ECCS function. Note 5 states that CFT isolation is only required when CFT pressure is more than or equal to the maximum RCS pressure for the existing RCS temperature, as allowed in LCO 3.4.3. This is acceptable since the CFT can not be the source of an overpressurization event when its pressure is less than the allowable RCS pressure.

The pressurizer is considered to represent a water solid condition when coolant level is > 105 inches, when RCS pressure is > 100 psig, or > 150 inches, when RCS pressure is ≤ 100 psig. Although a vapor space still exists with pressurizer level above these values, from an analytical point of view, the unit is considered to be water solid. These parameter values contain allowances for instrument error.

The pressurizer level requirements are modified by two Notes. Note 1 indicates that the requirements are not applicable during operation allowed by the Emergency Operating Procedures (EOPs). This exception provides for use of the "feed and bleed" process when necessary as determined by the EOPs. Note 2 indicates that the requirements are not applicable during RCS hydrotesting. Specific procedural controls are provided to prevent overpressurization during this activity.

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INSERT B3.4-62A

BASES

LCO (continued)

~~For the depressurized RCS, an RCS vent is OPERABLE when open with an area of at least [0.75] square inches.~~ 6

APPLICABILITY

This LCO is applicable in MODE 4 with ~~when any~~ RCS ~~cold leg~~ temperature ~~is~~ ≤ ~~[283]~~ 262 °F, in MODE 5, and in MODE 6 when the reactor vessel head is on. The Applicability temperature of ~~[283]~~ 262 °F is established by fracture mechanics analyses. The pressurizer safety valves provide overpressure protection to meet LCO 3.4.3 P/T limits above ~~[283]~~ 262 °F. With the vessel head off, overpressurization is not possible.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the pressurizer safety valves OPERABLE to provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above ~~[283]~~ 262 °F.

The parameter value (262°F) does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

~~The Applicability is modified by a Note stating that CFT isolation is only required when the CFT pressure is more than or equal to the maximum RCS pressure for the existing RCS temperature, as allowed in LCO 3.4.3. This Note permits the CFT discharge valve surveillance performed only under these pressure and temperature conditions.~~ 6

ACTIONS

~~A.1 and B.1~~

~~With two or more makeup pumps capable of injecting into the RCS or if the HPI is activated, immediate actions are required to render the other pump(s) inoperable or to deactivate HPI. Emphasis is on immediate deactivation because inadvertent injection with [one] or more HPI pump OPERABLE is the event of greatest significance, since it causes the greatest pressure increase in the shortest time. Also, the vent cannot mitigate overpressurization from the injection of even one HPI pump.~~ 6

~~The immediate Completion Times reflect the urgency of quickly proceeding with the Required Actions.~~

~~Required Action A.1 is modified by a Note that permits two pumps capable of RCS injection for ≤ 15 minutes to allow for pump swaps.~~

(continued)

<INSERT B3.4-62A>

OPERABLE pressure relief capability may be provided by an OPERABLE ERV, or by depressurizing the RCS and providing an alternate RCS vent path. For the ERV to be considered OPERABLE, its block valve must be open, its lift setpoint must be set at ≤ 460 psig, testing must have proven its ability to open at that setpoint, and motive power must be available to the ERV and its control circuits. With the RCS depressurized, acceptable alternate vent paths include removing a pressurizer safety valve, locking the ERV in the open position and disabling its block valve in the open position, removing a SG primary manway, removing a SG primary hand hole cover, removing all control rod drive top closure assemblies (excluding reactor vessel level probe), or removing a pressurizer manway.

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BASES

ACTIONS (continued)

~~C.1, D.1, and D.2~~

~~An unisolated CFT requires isolation within 1 hour only when the CFT pressure is at or more than the maximum RCS pressure for the existing temperature allowed in LCO 3.4.3.~~

~~If isolation is needed and cannot be accomplished in 1 hour, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed in 12 hours. By increasing the RCS temperature to > 175°F, the CFT pressure of 600 psig cannot exceed the LTOP limits if both tanks are fully injected. Depressurizing the CFTs below the LTOP limit of [555] psig also prevents exceeding the LTOP limits in the same event.~~

~~The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that a limiting LTOP event is not likely in the allowed times.~~ 6

A.1, B.1, and B.2

~~E.1, F.1, and F.2~~

not within its required limits

With the pressurizer level ~~more than [220] inches~~, the time for operator action in a pressure increasing event is reduced. The postulated event most affected in the LTOP MODES is failure of the makeup control valve, which fills the pressurizer relatively rapidly. Restoration is requirec within 1 hour. 6

If restoration within 1 hour in either case cannot be accomplished, Required Actions B.1 ~~F.1~~ and B.2 ~~F.2~~ must be performe within 12 hours to close the makeup control valve and its isolation valve. These Required Actions limit the makeup capability, which is not required with a high pressurizer level, and permit cooldown and depressurization to continue Heatup must be stopped because heat addition decreases the reactor coolant density and increases the pressurizer level

The Completion Times again are based on operating experien that these activities can be accomplished in these time periods and ~~on engineering evaluations indicating~~ that a limiting LTOP transient is not likely in the allowed times edit

(continu

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BASES

ACTIONS (continued)

C.1 and D.1 ~~G.1, H.1, and H.2~~

With the ~~PORV~~ required ERV inoperable, overpressure relieving capability is lost, and restoration of the ~~P~~ERV within 1 hour is required. If that cannot be accomplished, the ability of the Makeup System to add water must be limited within the next 12 hours.

If restoration cannot be completed within 1 hour, Required Action ~~H.1 and Required Action H.2~~ D.1 must be performed to limit RCS water addition capability. Makeup is not deactivated to maintain the RCS coolant level. Required Action ~~H.1 and Required Action H.2~~ D.1 require~~s~~ reducing the makeup tank level to ~~[20]~~ 73 inches ~~and deactivating the low/low makeup tank level interlock to the borated water storage tank~~. This makes the available makeup water volume insufficient to exceed the LTOP limit by a makeup control valve full opening (Ref. 3).

This parameter value does contain allowances for instrument error. No additional allowances for instrument error are required in the implementing procedures

These Completion Times also consider these activities can be accomplished in these time periods. A limiting LTOP event is not likely in those times.

Some ~~P~~ERV testing or maintenance can only be performed at ~~plant~~ unit shutdown. Such activity is permitted if Required Action ~~H.1 and Required Action H.2 are~~ D.1 is taken to compensate for ~~PORV~~ required ERV unavailability.

~~I.1 and I.2~~ E.1

INSERT B3.4-64A

~~With the pressurizer level above [220] inches and the PORV inoperable or the LTOP System inoperable for any reason other than cited in Condition A through H, the system must be restored to OPERABLE status within 1 hour. When this is not possible, Required Action I.2 requires the RCS depressurized and vented within 12 hours from the time either Condition started.~~

~~One or more vents may be used. A vent size of ≥ [0.75] square inches is specified. This vent size assumes 100 psig backpressure. Because makeup may be required, the vent size accommodates inadvertent full makeup system operation. Such a vent keeps the pressure from full flow of [one] makeup pump with a wide open makeup control valve within the LCO limit.~~

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(continued)

<INSERT B3.4-64A>

With the LTOP requirements not met for any reason other than cited in Condition A through D, action must be initiated to restore compliance immediately. The immediate Completion Time reflects the urgency of quickly proceeding with the Required Actions.

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BASES

ACTIONS

I.1 and I.2 (continued)

The PORV has a larger area and may be used for venting by opening and locking it open.

This size RCS vent or the PORVs a vent cannot maintain RCS pressure below LTOP limits if the HPI and CFT systems are inadvertently actuated. Therefore, verification of the deactivation of two HPI pumps, HPI injection, and the CFTs must accompany the depressurizing and venting. Since these systems are required deactivated by the LCO, SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3 require verification of their deactivated status every 12 hours.

Again, the Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that a limiting LTOP transient is not likely in those times.

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INSERT B3.4-65A

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3 (11) (11)

Verifications must be performed that only [one] makeup pump is capable of injecting into the RCS, the HPI is deactivated, and the CFT discharge isolation valves are closed and immobilized. These Surveillances ensure the minimum coolant input capability will not create an RCS overpressure condition to challenge the LTOP System. The Surveillances are required at 12 hour intervals.

each pressurized

is isolated.

The 12 hour intervals are shown by operating practice to be sufficient to regularly assess conditions for potential degradation and verify operation within the safety analysis.

coolant input capability

6

SR 3.4.12.4

INSERT B3.4-65B

Verification of the pressurizer level at ≤ [220] inches by observing control room or other indications ensures a cushion of sufficient size is available to reduce the rate of pressure increase from potential transients.

The 30 minute Surveillance Frequency during heatup and cooldown must be performed for the LCO Applicability period when temperature changes can cause pressurizer level

(continued)

<INSERT B3.4-65A>

SR 3.4.11.1

3.4B-19

Verification of the pressurizer level at at ≤ 105 inches when RCS pressure is > 100 psig or ≤ 150 inches when RCS pressure is ≤ 100 psig, by observing control room or other indications ensures that the unit is not in a water solid condition and that a cushion of sufficient size is available to reduce the rate of pressure increase from potential transients (Ref. 3).

The 30 minute Surveillance Frequency during heatup and cooldown must be performed for the LCO Applicability period when temperature changes can cause pressurizer level variations. This Frequency may be discontinued when these evolutions are complete, as defined in unit procedures. Thereafter, the Surveillance is required at 12 hour intervals.

These Frequencies are shown by operating practice sufficient to regularly assess indications of potential degradation and verify operation within the safety analysis.

3.4B-10

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.~~12~~ 11 .4 (continued)

~~variations. This Frequency may be discontinued when the ends of these conditions are satisfied, as defined in plant procedures. Thereafter, the Surveillance is required at 12 hour intervals.~~

~~These Frequencies are shown by operating practice sufficient to regularly assess indications of potential degradation and verify operation within the safety analysis.~~

INSERT B3.4-66A

~~SR 3.4.12.5~~

~~Verification that the PORV block valve is open ensures a flow path to the PORV. This is required at 12 hour intervals.~~

~~The interval has been shown by operating practice sufficient to regularly assess conditions for potential degradation and verify operation is within the safety analysis.~~

~~SR 3.4.12.6~~

~~When stipulated by LCO 3.4.12.b, the RCS vent of at least [0.75] square inches must be verified open for relief protection.~~ For a vent ~~valve~~ path not locked open, the Frequency is every 12 hours. For a ~~valve~~ vent path locked open, the required Frequency is every 31 days.

6

path

~~Again,~~ ~~the~~ Frequency intervals ~~consider~~ are considered adequate based on operating practice ~~to determine adequacy~~ ~~to regularly assess conditions for potential degradation~~ of pressure relief capability and verify operation within the safety analysis.

~~A Note modifies the SR by requiring the Surveillance when complying with LCO 3.4.12.b.~~

~~SR 3.4.12.7~~

~~A CHANNEL FUNCTIONAL TEST is required within [12] hours after decreasing RCS temperature to ≤ [283]°F and every 31 days thereafter to ensure the setpoint is proper for~~

3.4B-11

(continued)

<INSERT B3.4-66A>

SR 3.4.11.4

OPERABLE pressure relief capability must be provided to prevent overpressurization due to inadvertent full makeup system operation. Such a vent keeps the pressure from full makeup flow within the LCO limit. OPERABLE pressure relief capability may be provided by an OPERABLE ERV, or by depressurizing the RCS and providing an alternate RCS vent path.

For the ERV to be considered OPERABLE, its block valve must be open, its lift setpoint must be set at ≤ 460 psig, testing must have proven its ability to open at that setpoint, and motive power must be available to the two valves and their control circuits. The parameter value of 460 psig does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

With the RCS depressurized, acceptable alternate vent paths include: a) removing a pressurizer safety valve; b) locking the ERV in the open position and disabling its block valve in the open position; c) removing a SG primary manway; c) removing a SG primary hand hole cover; d) removing all control rod drive top closure assemblies (excluding reactor vessel level probe); and e) removing a pressurizer manway.

3.4B-11

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3.4B-11

(5)

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.4.12.7 (continued)~~

~~using the PORV for LTOP. PORV actuation is not needed, as it could depressurize the RCS.~~

~~The [12] hour Frequency considers the unlikelihood of a low temperature overpressure event during the time. The 31 day Frequency is based on industry accepted practice and is acceptable by experience with equipment reliability.~~

SR 3.4.~~12.8~~ 11.5

The performance of a CHANNEL CALIBRATION is required every ~~[18]~~ months. The CHANNEL CALIBRATION for the LTOP ERV opening logic, including the ERV setpoint ensures that the ~~P~~ ERV will be actuated at the appropriate RCS pressure by verifying the accuracy of the instrument string. The calibration can only be performed in shutdown. (6)

(E)

The 18 month Frequency considers a typical refueling cycle and industry accepted practice.

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. ~~FSAR, Section 15.~~
4. ~~10 CFR 50.46.~~
5. ~~10 CFR 50, Appendix K.~~

INSERT B3.4-67A

<INSERT B3.4-67A>

3. ANO-1 LTOP Safety Evaluation Report (1CNA058302) dated May 5, 1983.

4. Response to NRC Request for Additional Information (1CAN117608) dated November 15, 1976.

5. Response to NRC Request for Additional Information (1CAN127602) dated December 3, 1976.

6. Response to NRC Request for Additional Information (1CAN037716) dated March 24, 1977.

7. ANO-1 License Amendment Request (1CAN119608), dated November 26, 1988, and Operating License Amendment 188, (1CNA039703) dated March 14, 1997.

8. ANO-1 Request for Exemption (1CAN119608), dated November 26, 1996, and Exemption from Requirements of 10 CFR 50.60, (1CNA039702) dated March 12, 1997.

9. 10 CFR 50.36.

10. ANO-1 License Amendment Request (1CAN059008), dated May 22, 1990, and Operating License Amendment 138, (1CNA119002) dated November 1, 1990.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

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

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

~~10 CFR 50, Appendix A~~ SAR, Section 1.4, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable ~~methods~~ criteria for selecting Leakage Detection Systems. Reference 3 provides a comparison of the ANO-1 RCS leak detection systems to Regulatory Guide 1.45 (Ref. 2). 32 31

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

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

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation~~, and the core from inadequate cooling, in addition to preventing the accident analysis radiation release assumptions from being exceeded~~. The consequences of violating this LCO include ~~the possibility~~ increasing the probability of a loss of coolant accident (LOCA). However, edit edit

(continued)

BASES

BACKGROUND (continued)

the ability to monitor leakage provides advance warning to permit ~~plant~~ unit shutdown before a LOCA occurs. This advantage has been shown by "leak before break" studies. edit

---

APPLICABLE SAFETY ANALYSES

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

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

The ~~F~~USAR (Ref. ~~3~~ 4) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to the condenser. The 1 gpm primary to secondary LEAKAGE is relatively inconsequential. edit

~~The SLB is more limiting for site radiation releases.~~ The safety analysis for the SLB accident assumes 1 gpm primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100. edit 31

<INSERT B3.4-69A>

In MODES 1 and 2, RCS operational LEAKAGE satisfies Criterion 2 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 6). 35

<INSERT B3.4-69B>

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued

(continued)

<INSERT B 3.4-69A>

RCS leakage detection capabilities and methods are identified and discussed in SAR Section 4.2.3.8 (Ref. 5) and in the Bases for LCO 3.4.15, "RCS Leakage Detection Instrumentation."

<INSERT B 3.4-69A>

In MODES 3 and 4, RCS Operational Leakage satisfies Criterion 4 of 10 CFR 50.36.

Controlled reactor coolant pump (RCP) seal water leakoff (bleed off) is a normal function and is not considered as LEAKAGE.

BASES

LCO (continued)

degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

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

reactor building

edit

31

c. Identified LEAKAGE

unidentified

26

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of ~~identified~~ LEAKAGE and is well within the capability of the RCS makeup system. Identified LEAKAGE includes LEAKAGE to the ~~containment~~ from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

and LEAKAGE through a SG to the secondary system

edit
edit

d. ~~Primary to Secondary LEAKAGE through All Steam Generators (SGs)~~

~~Total primary to secondary LEAKAGE amounting to 1 gpm through all SGs produces acceptable offsite doses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.~~

1

~~e.~~ d. Primary to Secondary LEAKAGE through Any One SG

(0.104 gpm)

150

The ~~[720]~~ gallon per day limit on one SG ~~allocates the total 1 gpm allowed primary to secondary LEAKAGE equally between the two generators.~~

1

<INSERT B 3.4-70A>

(continue

<INSERT B 3.4-70A>

is intended to assure timely shutdown of the plant for appropriate corrective action before rupture of the steam generator tube(s) occurs under normal operating or postulated accident conditions. These limits also serve to provide added assurance that the dosage contribution from tube leakage will be limited to a small fraction of 10 CFR 100 (Ref. 7) limits for a design basis steam generator tube rupture or main steam line break. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

BASES (continued)

3.4B-21

APPLICABILITY

In MODES 1, 2, 3, and 4, the LEAKAGE limits are required because ~~the potential for RCPB LEAKAGE is greatest~~, ~~when~~ the RCS is pressurized and. edit

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

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through ~~each individual PIV~~ RCS pressure isolation valves (PIVs) and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV in series does not result in RCS LEAKAGE when the other is leaktight. If both valves leak and result in a loss of coolant mass from the RCS, the loss must be included in the allowable identified LEAKAGE. edit edit

ACTIONS

A.1

If ~~unidentified LEAKAGE, identified LEAKAGE, or~~ primary to secondary LEAKAGE ~~are~~ is in excess of the LCO limits, the LEAKAGE must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and ~~either identify unidentified LEAKAGE or~~ reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the primary to secondary RCPB. 1

INSERT B3.4-71A

C.1 and C.2

If any pressure boundary LEAKAGE exists or if the Required Action and associated Completion Time of Condition A or B are not met, ~~unidentified, identified, or primary to secondary LEAKAGE cannot be reduced to within limits within 4 hours,~~ the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary. edit

The Completion Times allowed are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging ~~plant~~ unit systems. In MODE 5, the pressure stresses acting on the RCPB are much lower and further deterioration is much less likely.

(continued)

<INSERT B3.4-71A>

B.1

If unidentified LEAKAGE, or identified LEAKAGE, or both, are in excess of the LCO limits, the LEAKAGE must be reduced to within limits within 18 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE within the LCO limits ensures that the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and may ~~can only~~ be positively identified by inspection. Total ~~Unidentified LEAKAGE and identified~~ LEAKAGE is ~~are~~ determined by performance of an RCS water inventory balance. ~~Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems~~. edit 31

<INSERT B3.4-72A>
<INSERT B3.4-72B>

The RCS water inventory balance must be performed with the reactor at steady state operating conditions ~~and~~ at or near operating pressure. Therefore, a Note is added allowing that this SR is not required to be performed ~~in MODES 3 and 4~~ until 12 hours ~~of~~ after establishing steady state operation at or near operating pressure (i.e., at or near 2155 psig). ~~have been established.~~ 11

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

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the ~~automatic~~ systems that monitor the reactor building ~~containment~~ atmosphere radioactivity and the reactor building ~~containment~~ sump level. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation." 31 edit

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. ~~The Note states that the SR is required to be performed in steady state operation.~~ 11

SR 3.4.13.2

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

(continued)

<INSERT B3.4-72A>

(stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows)

<INSERT B3.4-72B>

The 12 hour allowance provides sufficient time to collect and process all necessary data after stable unit conditions are established.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.2 (continued)

Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.

REFERENCES

1. SAR, Section 1.4 ~~10 CFR 50, Appendix A,~~ GDC 30.
2. Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, May 1973.
3. Information Submittal - Comparison of ANO-1 RCS Leak Detection Systems to Regulatory Guide 1.45 (ICAN 108607), dated October 14, 1986.
4. ~~3~~. ~~F~~SAR, Chapter ~~[15]~~ 14.
5. SAR, Section 4.2.3.8.
6. 10 CFR 50.36.
7. 10 CFR 100.

32 edit edit edit

edit

35

edit

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

pressure isolation valves (PIVs) are identified in Reference 1 (12)

BASES

BACKGROUND

~~10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3), define~~ RCS ~~PIVs~~ as any two normally closed valves in series within the RCS pressure boundary that separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

INSERT B3.4-74A (12)

The PIV leakage limit applies to each individual ~~valve~~. Leakage through both series PIVs in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 3.4.13.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leakage rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight.

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

The ~~basis for this LCO is the~~ 1975 NRC "Reactor Safety Study" (Ref. ~~4~~ 2) ~~that~~ identified potential intersystem LOCAs as a significant contributor to the risk of core melt. edit

(continued)

<INSERT B3.4-74A>

isolation check valve which is closest to the reactor vessel in the decay heat system injection lines and to each parallel pair of check valves which protect an individual low pressure injection line (Ref. 1).

In 1981, PIV requirements were issued as an order for modification of the ANO-1 Operating License (Ref. 1).

(12)

BASES

BACKGROUND (continued)

A subsequent study (Ref. ~~3~~ 3) evaluated various PIV configurations to determine the probability of intersystem LOCAs.

edit

edit

PIVs are provided to isolate the RCS from the ~~following typically connected systems:~~ low pressure portion of the

~~a.~~ Decay Heat Removal (DHR) System~~s~~.

(12)

~~b. Emergency Core Cooling System (ECCS); and~~

~~c. Makeup and Purification System.~~

~~The PIVs are listed in [FSAR section] Reference 6.~~

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of ~~a low pressure system~~ the DHR System and the loss of the integrity of a fission product barrier.

APPLICABLE SAFETY ANALYSES

INSERT B3.4-75A

Reference ~~4~~ 2 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the DHR System outside of ~~containment~~ the reactor building. The accident is the result of a postulated failure of the PIVs, which are part of the reactor coolant pressure boundary (RCPB), and the subsequent pressurization of the DHR System. ~~downstream of the PIVs from the RCS. Because the low pressure portion of the DHR System is typically designed for 600 psig,~~ overpressurization failure of the DHR low pressure line would result in a LOCA outside ~~containment~~ the reactor building and subsequent risk of core melt.

edit

edit

edit

Reference ~~5~~ 3 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

edit

RCS PIV leakage satisfies Criterion ~~2~~ 4 of the ~~NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 4).

(35)

(continued)

<INSERT B3.4-75A>

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event.

BASES (continued)

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The LCO PIV leakage limit is 0.5 gpm ~~per nominal inch of valve size with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.~~ 14

Reference ~~7~~ 5 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to account for the maximum pressure differential by assuming leakage is directly proportional to the square root of the pressure differential ~~to the one half power~~. edit

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the DHR flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the DHR mode of operation.

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

3.4B-17

ACTIONS

The ACTIONS are modified by two Notes. Note 1 is added to provide clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable.

(continued)

BASES

ACTIONS (continued)

Required Action

The ~~leakage~~ may have ~~affected system operability, or isolation of a leaking flow path with an alternate valve may have~~ degraded the ability of the interconnected system to perform its safety function.

12

A.1 and A.2

leaking

The flow path must be isolated by two valves. ~~Required Actions A.1 and A.2 are modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be on the RCS pressure boundary [or the high pressure portion of the system].~~

INSERT B3.4-77A

Required Action A.1 requires that the isolation ~~with one valve~~ must be performed within 4 hours. Four hours provides time ~~to reduce leakage in excess of the allowable limit and~~ to isolate the affected system ~~if leakage cannot be reduced. The 4 hours allows the actions~~ and restricts the operation with leaking isolation valves.

13

~~Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation or restoring one leaking PIV. The 72 hour time after exceeding the limit considers the time required to complete the Action and the low probability of a second valve failing during this time period.~~

~~or~~

~~The 72 hour time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This timeframe considers the time required to complete this Action and the low probability of a second valve failing during this period. (Reviewer Note: Two options are provided for Required Action A.2. The second option (72 hour restoration) is appropriate if isolation of a second valve would place the unit in an unanalyzed condition).~~

MOVE UP B.1 FROM NEXT PG

C C

~~B~~.1 and ~~B~~.2

and associated Completion Times are not met

13

edit

If ~~leakage cannot be reduced, [the system isolated,] or other~~ Required Actions ~~accomplished~~, the ~~plant~~ must be brought to a MODE in which the requirement does not apply.

unit

(continued)

<INSERT B3.4-77A>

When using this automatic MOV for isolation, deactivation makes the low pressure injection subsystem of one train of the ECCS inoperable since the MOV must automatically open to provide the LPI ECCS function. The ECCS Specification will effectively limit continued operation.

BASES

ACTIONS

C C

~~B~~.1 and ~~B~~.2 (continued)

unit

To achieve this status, the ~~plant~~ must be brought to MODE 3 within 6 hours and to MODE 5 within 36 hours. This Required Action may reduce the leakage and also reduces the potential for a LOCA outside the ~~containment~~. The allowed Completion Times are reasonable based on operating experience, to reach the required ~~plant~~ conditions from full power conditions in an orderly manner and without challenging ~~plant~~ systems.

reactor building

unit

13 edit edit edit edit

MOVE B.1 TO PREV PAGE ...... B ~~C~~.1

13

The inoperability of the DHR autoclosure interlock renders the DHR suction isolation valves incapable of isolating in response to a high pressure condition and preventing inadvertent opening of the valves at RCS pressures in excess of the DHR systems design pressure. If the DHR autoclosure interlock is inoperable, operation may continue as long as the DHR suction penetration is closed by at least one closed manual or deactivated automatic valve within 4 hours. This action accomplishes the purpose of the autoclosure function.

required and

edit

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

pressure check

Performance of leakage testing on ~~each~~ RCS ~~PIV or~~ isolation valve(s) ~~used to satisfy Required Action A.1 or A.2~~ is required to verify that leakage is below the specified limit and to identify ~~each~~ leaking valve(s). 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.

14

INSERT B 3.4-78B

INSERT B3.4-78A

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

13 edit

Testing is to be performed ~~every [18] months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The [18 month]~~ Frequency ~~is~~ consistent with

on a

edit

(continued)

<INSERT B3.4-78A>

isolation check valve which is closest to the reactor vessel in the DHR System injection lines (DH-14A and DH-14B) and to each parallel pair of check valves which protect an individual low pressure injection line (total for DH-13A and DH-17, and total for DH-13B and DH-18).

<INSERT B3.4-78B>

Reference 5 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to account for the maximum pressure differential by assuming leakage is directly proportional to the square root of the pressure differential.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

and

10 CFR 50.55a(g) (Ref. ~~8~~ 6) as contained in the Inservice Testing Program, ~~is within frequency~~ allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. ~~9~~ 5), ~~and~~ is based on the need to perform such surveillances under conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the ~~plant~~ unit at power.

This Frequency

edit

edit

~~In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.~~

14

The leakage ~~limit~~ Surveillance is to be ~~met~~ performed at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

14

INSERT B3.4-79A

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complimentary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. ~~In addition, this Surveillance is not required to be performed on the DHR System when the DHR System is aligned to the RCS in the decay heat removal mode of operation. PIVs contained in the DHR flow path must be leakage rate tested after DHR is secured and stable unit conditions and the necessary differential pressures are established.~~

33

~~Reviewer Note: The "24 hour..." Frequency of performance for Surveillance Requirement 3.4.14.1 is not required for B&W Owner's Group plants licensed prior to 1980. These plants were licensed prior to the NRC establishing formal Technical Specification controls for pressure isolation valves. Subsequently, these earlier plants had their~~

14

(continued)

<INSERT B3.4-79A>

To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

~~licenses modified by NRC Order to require certain PIV testing Frequencies (excluding the "24 hour..." Frequency) be included in that plant's Technical Specifications. Based upon the information available to the Staff at the time, the content of those Orders was considered acceptable. Since 1980, the NRC Staff has determined an additional PIV leakage rate determination is required within 24 hours following actuation of the valve and flow through the valve. This is necessary in order to ensure the PIV's ability to support the integrity of the reactor coolant pressure boundary. The Revised Standard Technical Specifications include the "24 hours..." Frequency to reflect current NRC Staff position on the need to include this test requirement within Technical Specifications.~~ 14

SR 3.4.14.2 ~~and~~, SR 3.4.14.3, SR 3.4.14.4, and SR 3.4.14.5 16

Verifying that the DHR autoclosure interlocks are OPERABLE ensures that RCS pressure will not pressurize the DHR system ~~beyond 125% of its design pressure of [600] psig~~. The interlocks ~~setpoint~~ that prevents the valves from being opened ~~is set so the actual RCS pressure must be < [425] psig to open the valves. This setpoint ensures the DHR design pressure will not be exceeded and the DHR relief valves will not lift.~~ The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance was performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. 24 16 on edit

over

and that close the valves are designed to to protect the DHR system from gross overpressurization. Although the specified values include certain process measurement uncertainties, additional allowances for instrument uncertainty are contained in the implementing procedures.

~~These SRs are modified by Notes allowing the DHR autoclosure function to be disabled when using the DHR System suction relief valve for cold overpressure protection in accordance with LCO 3.4.12.~~ 15

3.4B-22

REFERENCES

1. ~~10 CFR 50.2.~~ "Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves," issued April 20, 1981.

~~2. 10 CFR 55a(c).~~

edit

(continued)

The Probability of Intersystem LOCA:
Impact Due to Leak Testing and Operational Changes,

BASES

REFERENCES (continued)

~~3. 10 CFR 50, Appendix A, Section V, GDC 55.~~ edit

Reactor Safety Study,

2 ~~4~~. NUREG-75/014, Appendix V, October 1975. edit

3 ~~5~~. NUREG-0677, ~~NRC~~, May 1980. edit

4 ~~6~~. ~~[Document containing list of PIVS.]~~ 10 CFR 50.36 35

5 ~~7~~. ASME, Boiler and Pressure Vessel Code, Section XI. edit

6 ~~8~~. 10 CFR 50.55a(g). edit

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

SAR, Section 1.4,

GDC 30 ~~of Appendix A to 10 CFR 50~~ (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable ~~methods~~ criteria for selecting leakage detection systems. 32 edit

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

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can readily be detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The ~~containment~~ reactor building sump used to collect unidentified LEAKAGE is instrumented to ~~alarm for~~ detect increases of ~~0.5 to~~ 1.0 gpm in the ~~normal flow~~ fill rates. This sensitivity is acceptable for detecting increases in unidentified LEAKAGE. edit edit 23

The reactor coolant contains radioactivity that, when released to the ~~containment~~ reactor building, can be detected by radiation monitoring instrumentation. ~~Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. Instrument sensitivities of 10^{-9} µCi/cc radioactivity for particulate monitoring and of 10^{-6} µCi/cc radioactivity for gaseous monitoring are practical for these leakage detection systems.~~ Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS LEAKAGE. 23 edit

~~An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can thus be used to monitor humidity levels of the containment atmosphere as an indicator of potential RCS LEAKAGE. A 1°F increase in dew~~ 23

(continued)

BASES

BACKGROUND (continued)

~~point is well within the sensitivity range of available instruments.~~

~~Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate by this means may be questionable and should be compared to observed increases in liquid flow into or from the containment sump [and condensate flow from air coolers]. Humidity level monitoring is considered most useful as an indirect alarm or indication to alert the operator to a potential problem. Humidity monitors are not required for this LCO.~~ 23

reactor building

Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS LEAKAGE into the containment. The relevance of temperature and pressure measurements are affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.

Reactor building unit reactor building Indications 23 edit edit

APPLICABLE SAFETY ANALYSES

INSERT B3.4-83A

The need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications from other systems is necessary. The system response times and sensitivities are described in the FSAR (Ref. 3). ~~Multiple instrument locations are utilized, if needed, to ensure the transport delay time of the leakage from its source to an instrument location yields an acceptable overall response time.~~ 23

edit edit

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

In MODES 1 and 2, RCS leakage detection instrumentation satisfies Criterion 1 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 4). 35

In MODES 3 and 4, RCS leakage detection instrumentation satisfies Criterion 4 of 10CFR50.36.

(continued)

<INSERT B3.4-83A>

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. Therefore, the

BASES (continued)

LCO

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

edit

The LCO requirements are satisfied when monitors of diverse measurement means are available. Thus, the ~~containment~~ reactor building sump monitor, in combination with a particulate or gaseous radioactivity monitor, provides an acceptable minimum. edit

---

APPLICABILITY

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

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

edit

28

ACTIONS

INSERT B3.4-84B from page B3.4-85

A.1 and A.2

With the required ~~containment~~ reactor building sump monitor inoperable, no other form of sampling can provide the equivalent information. edit

However, the ~~containment~~ reactor building atmosphere activity monitor will provide indications of changes in leakage. Together with the atmosphere monitor, the periodic surveillance for RCS inventory balance, SR 3.4.13.1, ~~water inventory balance, must be performed~~ performing at an increased frequency of 24 hours ~~to~~ provides information that is adequate to detect leakage. edit

edit

17

INSERT B3.4-84A

Restoration of the required sump monitor to OPERABLE status is required to regain the function in a Completion Time of 30 days after the monitor's failure. This time is

(continued)

<INSERT B3.4-84A>

A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows) at or near operating pressure. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable unit conditions are established.

BASES

ACTIONS

Move to INSERT as B3.4-84B

A.1 and A.2 (continued)

acceptable considering the ~~f~~requency and adequacy of the RCS water inventory balance required by Required Action A.1. edit

The ~~Required~~ Actions ~~A.1 and Required Action A.2~~ are modified by a Note indicating that the provisions of LCO 3.0.4 do not apply. As a result, a MODE change is allowed when the sump monitors ~~is~~ are inoperable. This allowance is provided because other instrumentation is available to monitor RCS LEAKAGE. 28

and required radiation

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

With the required gaseous or particulate ~~containment~~ reactor building atmosphere radioactivity monitoring instrumentation channel~~s~~ inoperable, alternative action is required. Either grab samples of the ~~containment~~ reactor building atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. With a sample obtained and analyzed or a water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of at least one of the radioactivity monitors. edit

edit

The 24 hour interval provides periodic information that is adequate to detect leakage. INSERT B3.4-85A The 30 day Completion Time recognizes at least one other form of leak detection is available. 17

~~Required Actions B.1.1, B.1.2, and B.2 are modified by a Note indicating that the provisions of LCO 3.0.4 do not apply. As a result, a MODE change is allowed when the containment atmosphere radioactivity monitor is inoperable. This allowance is provided because other instrumentation is available to monitor RCS LEAKAGE.~~ 28 edit

C.1 and C.2

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

(continued)

<INSERT B3.4-85A>

A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows) at or near operating pressure. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable unit conditions are established.

BASES

ACTIONS

C.1 and C.2 (continued)

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

D.1

indicated 23

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

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

reactor building

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required ~~containment~~ atmosphere radioactivity monitor. The check gives reasonable confidence that each channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions. edit

SR 3.4.15.2

reactor building

SR 3.4.15.2 requires the performance of a CHANNEL FUNCTIONAL TEST of the required ~~containment~~ atmosphere radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm function ~~setpoint~~ and relative accuracy of the instrument string. The Frequency of 92 days considers instrument reliability, and operating experience has shown it proper for detecting degradation. edit edit

SR 3.4.15.3 and SR 3.4.15.4

These SRs require the performance of a CHANNEL CALIBRATION for each of the required RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside the reactor building ~~containment~~. The Frequency of [18] months is a typical refueling cycle and considers channel reliability. edit

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.4.15.3 and SR 3.4.15.4 (continued) ~~Again~~ Additionally, operating experience has ~~proven~~ shown this Frequency is acceptable. edit
REFERENCES	1. ~~10 CFR 50, Appendix A, Section IV~~ SAR, Section 1.4, GDC 30. 32 2. Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, May 1973. edit 3. ~~F~~SAR, Section ~~[1.]~~ 4.2.3.8. edit 4. 10 CFR 50.36. 35

5

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.~~16~~ 12 RCS Specific Activity

BASES

BACKGROUND

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

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

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

~~The parametric evaluations showed the potential offsite dose levels for an SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits (Ref. 1). Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.~~ 19

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensure that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following an SGTR accident. ~~The SGTR safety analysis (Ref. 2) assumes the specific activity of the reactor coolant at the LCO limits and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm. The analysis also assumes a reactor trip and a turbine trip at the same time as the SGTR event.~~ 7

INSERT B3.4-88A

~~The analysis for the SGTR accident establishes the acceptance limits for RCS specific activity. Reference to~~

(continued)

<INSERT B3.4-88A>

The thyroid dose conversion factors used in the calculation of DOSE EQUIVALENT I-131 are identified in Section 1.1, "Definitions."

Rupture of a steam generator tube would allow primary coolant activity to enter the secondary coolant. The major portion of this activity is noble gases and would be released to the atmosphere from the condenser vacuum pump or a relief valve. Activity would continue to be released until the operator could reduce the primary system pressure below the setpoint of the secondary relief valves and could isolate the faulty steam generator. The worst credible set of circumstances is considered to be a double-ended break of a single steam generator tube, followed by isolation of the faulty steam generator within 34 minutes after the tube break. Assuming the full differential pressure across the steam generator, no more than one-quarter of the total primary coolant could be released to the secondary coolant in this period. The decay heat during this period of 1 hour for pressure reduction will generate steam in the secondary system representing less than 15 weight percent of the secondary system.

The parameters assumed in the dose analysis (Ref. 2) for the single steam generator tube failure included the following values:

1) total primary coolant volume (mass) = 5.2×10^5 lbs.

2) total secondary coolant volume (mass) = 2×10^6 lbs.

3) leakage rate from primary to secondary system = 1 gpm.

4) fission product decay heat energy for 1 hour = 1.56×10^8 BTU.

5) steam mass released to environs = 2.84×10^5 lbs.

6) primary coolant released to secondary (34 minutes) = 8.7×10^4 lbs.

7) minimum primary to secondary iodine equilibrium activity ratio = 20 to 1 (for 1 gpm leakage).

8) DOSE EQUIVALENT I-131 specific activity = 3.5 μCi/gm (Primary)

9) DOSE EQUIVALENT I-131 specific activity = 0.17 μCi/gm (Secondary).

10) total specific activity in primary = $72/\bar{E}$ μCi/gm.

11) X/Q = 7.0×10^{-4} sec/m^3 at limiting point beyond site boundary of 1046 meters for 30 m release height - equivalent to ground level release due to topography including building wake effect for 5 percentile meteorology.

12) total radioactivity in primary coolant released to secondary coolant released to environs.

13) ten percent of the combined radioiodine activity from primary activity in secondary coolant and secondary activity present in steam mass (released to environs) assumed released to environs.

<INSERT B3.4-88A> (continued)

The whole body dose resulting from immersion in the cloud containing the released activity would include both gamma and beta radiation. The gamma dose is dependent on the finite size and configuration of the cloud. However, the analysis employed the simple model of the semi-infinite cloud, which gives an upper limit to the potential gamma dose. The semi-infinite cloud model is applicable to the beta dose, because of the short range of beta radiation in air. The resulting whole body dose was determined to be less than 0.5 Rem for this accident.

The thyroid dose from the steam generator tube rupture accident has been analyzed assuming a tube rupture at full load and loss of offsite power at the time of the reactor trip, which results in steam release through the relief valves in the period before the faulty steam generator is isolated and primary system pressure is reduced. The limiting iodine activities for the primary and secondary systems are used in the initial conditions. One-tenth of the iodine contained in the liquid which is converted to steam and passed through the relief valves is assumed to reach the site boundary. The resulting thyroid dose from the combined primary and secondary iodine activity released to the environs was determined to be 1.5 Rem for this accident.

The limit for secondary iodine activity is consistent with the limits on primary system iodine activity and primary-to-secondary leakage of 1 gpm. If the activity should exceed the specified limits following a power transient, the major concern would be whether additional fuel defects had developed bringing the total to above expected levels. From the observed removal of excess activity by decay and cleanup, it should be apparent whether activity is returning to a level below the specification limit. Appropriate action to be taken to bring the activity within specification include one or more of the following: gradual decrease in power to a lower base power, increase in letdown flow rate, and venting of the makeup tank gases to the waste gas decay tanks.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~this analysis is used to assess changes to the facility that could affect RCS specific activity as they relate to the acceptance limits.~~ 7

~~The rise in pressure in the ruptured SG causes radioactively contaminated steam to discharge to the atmosphere through the atmospheric dump valves or the main steam safety valves. The atmospheric discharge stops when the turbine bypass to the condenser removes the excess energy to rapidly reduce the RCS pressure and close the valves. The unaffected SG removes core decay heat by venting steam until the cooldown ends.~~

The ~~safety~~ analysis shows the radiological consequences of an SGTR accident are within a small fraction of the Reference 1 dose guideline limits. ~~Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable Specification, for more than 48 hours.~~ 7 edit

~~The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of an SGTR accident occurring during the established 48 hour time limit. The occurrence of an SGTR accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.~~

RCS Specific Activity satisfies Criterion 2 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref 3). 35

LCO

The specific iodine activity total is limited to ~~1.0~~ ≤3.5 μCi/gm DOSE EQUIVALENT I-131, and the ~~gross~~ total specific activity in the primary coolant is limited to the number of μCi/gm equal to ~~100~~ 72 divided by E ~~(average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides)~~. The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the ~~Design Basis Accident (DBA)~~ SGTR will be a small fraction of the allowed thyroid dose. The limit on ~~gross~~ total specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the ~~DBA~~ SGTR will be a small fraction of the allowed whole body dose. edit edit edit edit

(continued)

BASES

LCO (continued)

The ~~SGTR accident~~ analysis ~~(Ref. 2)~~ shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits. (7)

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and ~~gross~~ total specific activity are necessary to contain the potential consequences of an SGTR to within the acceptable site boundary dose values. edit

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

ACTIONS

~~A Note to the ACTIONS excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is 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.~~ (19)

A.1 ~~and A.2~~

With the ~~DOSE EQUIVALENT I-131~~ specific activity of the reactor coolant greater than the LCO limit, (5) ~~samples at intervals of 4 hours must be taken to demonstrate the limits of Figure 3.4.16-1 are not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling must continue for trending.~~ (19)

(continued)

BASES

ACTIONS

A.1 ~~and A.2~~ (continued)

The ~~DOSE EQUIVALENT I-131~~ specific activity must be restored to limits within ~~48~~ 24 hours. The Completion Time of ~~48~~ 24 hours is required, ~~if the limit violation resulted from normal iodine spiking.~~ adequate to determine and implement appropriate actions to return specific activity to within limits. 19

B.1

If ~~a~~ the Required Action and associated Completion Time ~~of Condition A~~ are not met ~~or if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.16-1~~, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. ~~The Completion Time of 6 hours is required to get to MODE 3 below 500°F without challenging reactor emergency systems.~~ 19

~~C.1 and C.2~~

~~With the gross specific activity in excess of the allowed limit, an analysis must be performed within 4 hours to determine DOSE EQUIVALENT I-131. The Completion Time of 4 hours is required to obtain and analyze a sample.~~

Placing the unit in ~~The allowed Completion Time of 6 hours to reach~~ MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves, and prevents venting the SG to the environment in an SGTR event. The Completion Time of 6 hours is required to reach MODE 3 from full power conditions in an orderly manner and without challenging unit ~~reactor emergency~~ systems. edit edit

SURVEILLANCE REQUIREMENTS

SR 3.4.~~16~~12.1 5

SR 3.4.~~16~~.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once per 7 days. ~~While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken.~~ This Surveillance provides an indication of any increase in gross specific activity. edit 7

INSERT B3.4-91A

3.4B-16

(continued)

<INSERT B3.4-91A>

3.4B-16

The gross specific activity analysis consists of the quantitative measurement of the total activity of the primary coolant in units of microcuries per gram (μCi/gm). The total primary coolant activity is the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled and any identified beta emitters (i.e., tritium, SR89, SR90, etc.).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.~~16~~12.1 (continued) (5)

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with RCS average temperature at least 500°F. The 7 day Frequency ~~considers the unlikelihood~~ is based on the low probability of a gross fuel failure during that time period. edit (5)

SR 3.4.~~16~~12.2

This Surveillance is performed in MODE 1 only to ensure the iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level considering gross specific activity is monitored every 7 days. ~~The Frequency, between 2 and 6 hours after a power change of ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.~~ (29)

SR 3.4.~~16~~12.3 (5)

SR 3.4.~~16~~12.3 requires radiochemical analysis for Ē determination every 184 days ~~(6 months) with the plant operating in MODE 1 equilibrium conditions~~. The Ē determination directly relates to the LCO and is required to verify plant operation within the total specific ~~gross~~ activity LCO limit. ~~The analysis for Ē is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines.~~ The Frequency of 184 days recognizes Ē does not change rapidly. (7)

INSERT B3.4-92A

This SR ~~has been~~ is modified by a Note that requires ~~sampling~~ the determination to be performed 31 days after a minimum of 2 EFPD and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures the radioactive materials are at equilibrium so the analysis for Ē is representative and not skewed by a crud burst or other similar abnormal event.

(continued)

<INSERT B3.4-92A>

3.4B-15

The radiochemical analysis consists of the quantitative measurement of the activity for each radionuclide which is identified in the primary coolant 15 minutes after the primary system is sampled. The activities for the individual isotopes are used in the determination of E-bar. The gamma energy per disintegration for those radioisotopes determined to be present shall be as given in "Table of Isotopes" (1967) (Ref. 4) and beta energy per disintegration shall be as given in USNRDL-TR-802 (Part II) (Ref. 5) or other references using the equivalent values for the radioisotopes. Iodine isotopic activities are weighted to give DOSE EQUIVALENT I-131 activity.

BASES (continued)

REFERENCES

1. 10 CFR 100.11.
3. ~~2~~ ~~FSAR, Section [15.6.3]~~. 10 CFR 50.36. 35

4. "Table of Isotopes" (1967).
5. USNRDL-TR-802 (Part II). 7
2. ANO-1 Operating License Amendment 2, (1CNA 057502) dated May 9, 1975.

3.4B-20 3.4B-15

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Core Flood Tanks (CFTs)

LCO 3.5.1 Two CFTs shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System (RCS) pressure > 800 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CFT inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One CFT inoperable for reasons other than Condition A.	B.1 Restore CFT to OPERABLE status.	6 hours
C. Required Action and associated Completion Time of Condition A or B not met. <u>OR</u> Two CFTs inoperable.	C.1 Be in MODE 3. <u>AND</u> C.2 Reduce RCS pressure to $\leq$ 800 psig.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each CFT isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify borated water volume in each CFT is ≥ 970 ft^3 and ≤ 1110 ft^3.	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each CFT is $\geq$ 560 psig and $\leq$ 640 psig.	12 hours

SURVEILLANCE		FREQUENCY
SR 3.5.1.4	Verify boron concentration in each CFT is ≥ 2270 ppm.	31 days AND --------NOTE--------- Only required to be performed for affected CFT -------------------------- Once within 12 hours after each solution level increase of ≥ 0.2 feet that is not the result of addition from a borated water source of known concentration ≥ 2270 ppm
SR 3.5.1.5	Verify power is removed from each CFT isolation valve operator.	31 days

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS - Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System (RCS) temperature > 350°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable.	A.1 Restore train(s) to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Reduce RCS temperature to ≤ 350°F.	6 hours 12 hours
C. Less than 100% of the ECCS flow equivalent to a single OPERABLE train available.	C.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.2.2	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program

SURVEILLANCE		FREQUENCY
SR 3.5.2.3	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.5.2.4	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.5.2.5	Verify, by visual inspection, each ECCS train reactor building sump suction inlet is not restricted by debris and screens show no evidence of structural distress or abnormal corrosion.	18 months

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS – Shutdown

LCO 3.5.3 Two LPI trains shall be OPERABLE.

-----NOTE-----
An LPI train may be considered OPERABLE during alignment and when aligned for decay heat removal, if capable of being manually realigned to the LPI mode of operation.

APPLICABILITY: MODE 3 with Reactor Coolant System (RCS) temperature $\leq$ 350°F,
MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One LPI train inoperable.	A.1 Restore LPI train to OPERABLE status.	48 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 5.	24 hours
C. Two LPI trains inoperable.	C.1 Initiate action to restore one LPI train to OPERABLE status. <u>AND</u> C.2 -----NOTE----- Only required if one DHR train is OPERABLE. ---------- Be in MODE 5.	Immediately 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	-------------------------------NOTE------------------------------- An LPI train may be considered OPERABLE during alignment and operation for DHR, if capable of being manually realigned to the LPI mode of operation. -- For all equipment required to be OPERABLE, the following SRs are applicable: SR 3.5.2.1, SR 3.5.2.4, SR 3.5.2.2, SR 3.5.2.5. SR 3.5.2.3,	In accordance with applicable SRs

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Borated Water Storage Tank (BWST)

LCO 3.5.4 The BWST shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. BWST boron concentration not within limits. <u>OR</u> BWST water temperature not within limits.	A.1 Restore BWST to OPERABLE status.	8 hours
B. BWST inoperable for reasons other than Condition A.	B.1 Restore BWST to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	-----------------------------NOTE----------------------------- Only required to be performed when ambient air temperature is < 40°F or > 110°F. -- Verify BWST borated water temperature is ≥ 40°F and ≤ 110°F.	24 hours

SURVEILLANCE		FREQUENCY
SR 3.5.4.2	Verify BWST borated water level is ≥ 38.4 feet and ≤ 42 feet.	7 days
SR 3.5.4.3	Verify BWST boron concentration is ≥ 2270 ppm and ≤ 2670 ppm.	7 days

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Core Flood Tanks (CFTs)

BASES

BACKGROUND

The function of the ECCS CFTs is to supply water to the reactor vessel during the blowdown phase of a loss of coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA. Two CFTs are provided for these functions.

The blowdown phase of a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the reactor building atmosphere.

In the refill phase of a LOCA, which follows immediately, reactor coolant inventory has vacated the core through steam flashing and ejection through the break. The core is essentially in adiabatic heatup. The balance of inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of safety injection water.

The CFTs are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The CFTs are passive components, since no operator or control actions are required for them to perform their function. Each CFT is piped separately into the reactor vessel downcomer. The CFT injection lines are also utilized by the Low Pressure Injection (LPI) System. Each CFT is isolated from the RCS by a motor operated isolation valve and two check valves in series. The motor operated isolation valves are normally open, with power removed from the valve motor to prevent inadvertent closure prior to or during an accident.

The CFTs thus form a passive system for injection directly into the reactor vessel. Except for the core flood line break LOCA, a unique accident that also disables a portion of the injection system, both tanks are assumed to operate in the safety analyses for Design Basis Events. Because injection is directly into the reactor vessel downcomer, and because it is a passive system not subject to the single active failure criterion, all fluid injected is credited for core cooling.

The CFT gas/water volumes, gas pressure, and outlet pipe size are selected to provide core cooling for a large break LOCA prior to the injection of coolant by the LPI System.

APPLICABLE SAFETY ANALYSES.

The CFTs are credited in both the large and small break LOCA analyses at full power (Ref. 1). The CFT line break analysis credits only one CFT, since the tank with the broken line is assumed to empty out the break. These Design Basis Accident (DBA) analyses establish the acceptance limits for the CFTs. In performing the LOCA calculations, conservative assumptions are made concerning the availability of safety injection flow. In addition, a loss of offsite power is considered to ensure worst case conditions are postulated. In the early stages of a limiting large break LOCA with the loss of offsite power, the CFTs provide the sole source of makeup water to the RCS. This is because the LPI pumps and high pressure injection (HPI) pumps cannot deliver flow until the diesel generators (DGs) start and go through their timed loading sequence.

The limiting large break LOCA is a double ended guillotine cold leg break at the discharge of the reactor coolant pump.

During this event, the CFTs discharge to the RCS as soon as RCS pressure decreases below CFT pressure. No credit is taken for HPI for large break LOCAs. LPI is not assumed to occur until 35 seconds after the RCS pressure decreases to the ESAS actuation pressure. No operator action is assumed during the blowdown stage of a large break LOCA (Ref. 1).

The small break LOCA analysis also assumes a time delay after ESAS actuation before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated by the CFTs, with pumped flow then providing continued cooling. As break size decreases, the CFTs and HPI pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the CFTs continues to decrease until the tanks are not required and the HPI pumps become responsible for terminating the temperature increase.

This LCO helps to ensure that the following acceptance criteria for the ECCS established by 10 CFR 50.46 (Ref. 2) will be met following a LOCA:

a. Maximum fuel element cladding temperature of 2200°F;

b. Maximum cladding oxidation of ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction of ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

The limits for operation with a CFT that is inoperable for any reason other than the boron concentration not being within limits minimize the time that the unit is exposed to a LOCA event occurring along with failure of a CFT, which might result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be opened, or the proper water volume or nitrogen cover pressure cannot be restored, the full capability of one CFT is not available and prompt action is required to place the reactor in a MODE in which this capability is not required.

The minimum volume requirement for the CFTs ensures that both CFTs can provide adequate inventory to cover the core to the 3/4 point even assuming no liquid remains in the reactor vessel following a LOCA (Ref. 1). The downcomer then remains flooded until the HPI and LPI systems start to deliver flow for limiting large break LOCAs.

The maximum volume limit is based upon the need to maintain adequate gas volume to ensure proper injection and ensure the ability of the CFTs to fully discharge. The limiting safety analysis volume requirement is 1040 ± 70 ft^3. This volume corresponds to CFT levels of ≥ 11.95 ft and ≤ 14.00 ft. These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

The minimum nitrogen cover pressure requirement of 560 psig ensures that the contained gas volume will generate discharge flow rates during injection that satisfy the safety analysis. This parameter value does not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

The maximum nitrogen cover pressure limit of 640 psig will affect the amount and timing of CFT inventory discharged while the RCS depressurizes. Limiting the maximum pressure will therefore limit the CFT inventory lost through the break and assure that the CFT inventory injected into the RCS at the proper time is bounded by that predicted by the safety analysis. This parameter value does not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

The 2270 ppm limit for minimum boron concentration was established to ensure that, following a LOCA with a minimum CFT level, the reactor will remain adequately shutdown in the cold condition following mixing of the CFT and Reactor Coolant System (RCS) water volumes. This parameter value is considered to be a nominal value. No additional allowances for instrument uncertainty are required in the implementing procedures.

In MODE 1, the CFTs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3). In MODE 2 and MODE 3 with RCS pressure > 800 psig, the CFTs satisfy Criterion 4 of 10 CFR 50.36.

LCO

The LCO establishes the minimum conditions required to ensure that the CFTs are available to accomplish their core cooling safety function following a LOCA. Both CFTs are required to function in the event of a large break LOCA. If the entire contents of both tanks are not injected during the blowdown phase of a large break LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated. For a CFT to be considered OPERABLE, the isolation valve must be fully open with power removed, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure > 800 psig, the CFT OPERABILITY requirements are based on full power operation. Although cooling requirements may decrease as power decreases, the CFTs are still required to provide core cooling as long as elevated RCS pressures and temperatures exist.

In MODE 3 with RCS pressure ≤ 800 psig, and in MODES 4, 5, and 6, the CFT motor operated isolation valves may be closed to isolate the CFTs from the RCS. This allows RCS cooldown and depressurization without discharging the CFTs into the RCS or requiring depressurization of the CFTs.

In addition, LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)," requires that in MODE 4 when any RCS cold leg temperature is ≤ 262°F, MODE 5, and MODE 6 when the reactor vessel head is on, each CFT whose pressure is greater than or equal to the maximum RCS pressure for the existing RCS temperature allowed by the pressure and temperature limit curves provided in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," be isolated.

ACTIONS

<u>A.1</u>

If the boron concentration of one CFT is not within limits, the ability to maintain subcriticality may be reduced, but the effects of reduced boron concentration on core subcriticality during reflood are minor. Boiling of the ECCS water in the core during reflood concentrates the boron in the saturated liquid that remains in the core. In addition, the volume of the CFT is still available for injection. Since the boron requirements are based on the average boron concentration of the total volume of two CFTs, the consequences are less severe than they would be if the contents of a CFT were not available for injection. Thus, 72 hours is allowed to return the boron concentration to within limits.

B.1

If one CFT is inoperable for a reason other than boron concentration, it cannot be assumed that the CFT will perform its required function during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 6 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable CFT to OPERABLE status. The Completion Time minimizes the time the unit is potentially exposed to a LOCA in these conditions.

C.1 and C.2

If the Required Actions and associated Completion Times of Condition A or B are not met, or if both CFTs are inoperable, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and RCS pressure reduced to ≤ 800 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Verification every 12 hours that each CFT isolation valve is fully open ensures that the CFTs are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in accident analysis assumptions not being met. A 12 hour Frequency is considered reasonable in view of administrative controls that ensure that a mispositioned isolation valve is unlikely.

SR 3.5.1.2 and SR 3.5.1.3

Verification every 12 hours of each CFT's nitrogen cover pressure and the borated water volume is sufficient to ensure adequate injection during a LOCA. Due to the static nature of these parameters, a 12 hour Frequency usually allows the operator to identify changes before the limits are reached. Operating experience has shown that this Frequency is appropriate for early detection and correction of off normal trends.

SR 3.5.1.4

Surveillance once every 31 days is reasonable to verify that the CFT boron concentration is within the required limits, because the static nature of this parameter limits the ways in which the concentration can be changed. The Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage. Sampling of the affected CFT within 12 hours after a 0.2 ft volume increase will identify whether inleakage from the RCS has caused a reduction in boron concentration to below the required limit. The 0.2 ft increase represents approximately 102 gallons increase in volume. It is not necessary to verify boron concentration if the added water inventory is from a borated water source of known concentration ≥ 2270 ppm, such as the borated water storage tank (BWST), because the water is within CFT boron concentration requirements. Similarly, it would not be necessary to sample the CFT following inventory additions from the CFT makeup tank if sampling has determined that the added inventory had a boron concentration within the CFT requirements. This is consistent with the recommendations of NUREG-1366 (Ref. 4).

SR 3.5.1.5

Removing power from each CFT isolation valve operator ensures that an active failure could not result in the undetected closure of a CFT motor operated isolation valve coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that the power is removed.

REFERENCES

1. SAR, Section 6.1 and 14.2.
2. 10 CFR 50.46.
3. 10 CFR 50.36.
4. NUREG-1366, "Improvements to Technical Specifications Surveillance Requirements," December 1992.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS - Operating

BASES

BACKGROUND

The ANO-1 SAR states that the High Pressure Injection (HPI), Low Pressure Injection (LPI) and Core Flooding Systems are collectively designed as an Emergency Core Cooling System (ECCS) (Ref. 1). In this Technical Specification, the term ECCS refers to the components associated with the HPI and LPI systems. The core flood tanks are addressed by LCO 3.5.1, "Core Flood Tanks."

The function of the ECCS is to provide core cooling to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA);

b. Rod ejection accident;

c. Steam generator tube rupture (SGTR); and

d. Main steam line break (MSLB).

There are two phases of ECCS operation: injection and recirculation. In the injection phase, borated water from the borated water storage tank (BWST) is initially added to the Reactor Coolant System (RCS) via the cold legs and directly to the reactor vessel. After the BWST has been depleted, the recirculation phase is entered as the suction is transferred to the reactor building sump.

Two redundant, 100% capacity trains are provided. In MODES 1 and 2, and MODE 3 with RCS temperature > 350°F, each train consists of high pressure injection (HPI) and low pressure injection (LPI) subsystems. In MODES 1 and 2, and MODE 3 with RCS temperature > 350°F, both trains must be OPERABLE. This ensures that 100% of the core cooling requirements can be provided even in the event of a single active failure.

A suction header supplies water from the BWST or the reactor building sump to the ECCS pumps. HPI discharges into each of the four RCS cold legs between the reactor coolant pump and the reactor vessel. LPI discharges into each of the two core flood nozzles on the reactor vessel that discharge into the vessel downcomer area. Valves are set to balance the HPI flow to the RCS. This flow balance directs sufficient flow to the core to meet the analysis assumptions following a small break LOCA in one of the RCS cold legs.

The HPI pumps are capable of discharging to the RCS at an RCS pressure above the opening setpoint of the pressurizer safety valves. The LPI pumps are capable of discharging to the RCS at pressures below approximately 200 psia. When the BWST has been nearly emptied, the suction for the LPI pumps is manually transferred to the reactor building sump. The HPI pumps cannot take suction directly from the sump. If HPI is still needed, a cross connect from the discharge side of the LPI pump to the suction of the HPI pumps would be opened. This is known as "piggy backing" and enables continued HPI to the RCS, if needed, after the BWST is emptied.

In the long term cooling period, the flowpath inherent in the reactor vessel internals, commonly called the leakage gap flowpath, would be sufficient by itself to preclude boron precipitation (Ref. 2). Flow paths in the LPI System may be procedurally established to preclude the possibility of boric acid in the core region reaching an unacceptably high concentration. The desired flowpath establishes decay heat removal (DHR) in conjunction with LPI cooling. This requires conditions present which allow both DHR pumps to operate simultaneously. If DHR can not be established but hot leg level is above the bottom of the hot leg nozzle, an alternate flowpath is gravity draining from the decay heat suction piping through the idle DHR pump into the reactor building sump. If the first two methods are unsuccessful, the pressurizer auxiliary spray line is used. This provides reverse flow through the core using auxiliary spray into the pressurizer, out the pressurizer into the hot leg via the surge line then reactor vessel into the area above the core.

The HPI subsystem also functions to supply borated water to the reactor core following increased heat removal events, such as large MSLBs.

During a large break LOCA, RCS pressure will rapidly decrease. The ECCS is actuated upon receipt of an Engineered Safeguards Actuation System (ESAS) signal. If offsite power has not been lost, the safeguard loads start in sequence unless previously operating. If offsite power has been lost, the Engineered Safeguards (ES) buses shed normal operating loads and are connected to the diesel generators. Safeguard loads are then connected in sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the amount of time before pumped flow is available to the core following a LOCA.

The active ECCS components, along with the passive core flood tanks (CFTs) covered in LCO 3.5.1, "Core Flood Tanks (CFTs)," and the BWST covered in LCO 3.5.4, "Borated Water Storage Tank (BWST)," provide the cooling water necessary to meet 10 CFR 50.46 (Ref. 1 and 3).

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 3 and 4), will be met following a LOCA:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained.

Only the LPI subsystem is assumed to provide injection in the large break LOCA analysis at full power (Ref. 4). This analysis establishes a minimum required flow for the LPI subsystem, as well as the minimum required response time for subsystem actuation. The HPI subsystem is credited in the small break LOCA analysis. This analysis establishes the flow and discharge head requirements for the HPI pump. The SGTR and MSLB analyses also credit the HPI subsystem but are not limiting in HPI subsystem design.

The large break LOCA event assumes a loss of offsite power and a single failure (disabling one ECCS train). For analysis purposes, the loss of offsite power assumption may be conservatively inconsistent with the assumed operation of some equipment, such as reactor coolant pumps (Ref. 4). During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the reactor building. The nuclear reaction is terminated either by moderator voiding during large breaks or CONTROL ROD insertion for small breaks (Ref.4). Following depressurization, emergency cooling water is injected into the reactor vessel core flood nozzles, then flows into the downcomer, fills the lower plenum, and refloods the core.

The safety analyses show that an LPI train will deliver sufficient water to match decay heat boiloff rates for a large break LOCA. They also show that the HPI train will deliver sufficient water for a small break LOCA and provide sufficient boron to maintain the core subcritical.

In the large break LOCA analyses, LPI is not credited until at least 35 seconds after actuation of the ESAS signal. This is based on a loss of offsite power and the associated time delays in startup and loading of the diesel generator (DG). Further, LPI flow is not credited until RCS pressure drops below the pump's shutoff head. For a large break LOCA, HPI is not credited at all.

In the small break LOCA analysis, HPI is not credited until at least 35 seconds after actuation of the ESAS signal. This is based on a loss of offsite power and the associated time delays in startup and loading of the DG.

In MODE 1, the ECCS trains satisfy Criterion 3 of 10 CFR 50.36 (Ref. 5). In MODE 2 and MODE 3 with RCS temperature > 350°F, the ECCS trains satisfy Criterion 4 of 10 CFR 50.36.

LCO

In MODES 1 and 2 and MODE 3 with RCS temperature > 350°F, two independent (and redundant) ECCS trains are required to ensure that at least one is available, assuming a single failure in the other train.

In MODES 1 and 2 and MODE 3 with RCS temperature > 350°F, an ECCS train consists of an HPI subsystem and an LPI subsystem. Each train includes the piping, instruments, pumps, valves, heat exchangers, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an ESAS signal and the capability to manually transfer suction to the reactor building sump.

During an event requiring ECCS actuation, a flow path is provided to ensure an abundant supply of water from the BWST to the RCS via the HPI and LPI pumps and their respective discharge flow paths to each of the four cold leg injection nozzles and the reactor vessel. In the long term, this flow path may be manually transferred to take its supply from the reactor building sump and to supply borated water to the RCS via two paths (LPI and HPI piggy-back modes).

The flow path for each train must maintain its designed independence to ensure that no single failure can disable both ECCS trains.

APPLICABILITY

In MODES 1 and 2 and MODE 3 with RCS temperature > 350°F, the ECCS train OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The HPI pump performance requirements are based on a small break LOCA.

In MODE 3 with RCS temperature ≤ 350°F and in MODE 4, ECCS train OPERABILITY requirements are established by LCO 3.5.3, "ECCS - Shutdown." In MODE 3 with RCS temperature ≤ 350°F and in MODE 4, the probability of an event requiring ECCS actuation is significantly lessened. In this operating condition, the safety injection function is preserved through LCO 3.5.3 requirements for two OPERABLE LPI trains.

In MODES 5 and 6, unit conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation -

High Water Level," and LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With one or more trains inoperable, but at least 100% of the injection flow equivalent to a single OPERABLE ECCS train still available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on NRC recommendations (Ref. 6) that are based on a risk evaluation and is a reasonable time for repairs.

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two diverse components, each in a different train, necessarily result in a loss of function for the ECCS. This allows increased flexibility in unit operations under circumstances when diverse components in opposite trains are inoperable, i.e., an HPI subsystem in one train and an LPI subsystem in the opposite train.

An event accompanied by a loss of offsite power and the failure of a DG can disable one ECCS train until power is restored. A reliability analysis (Ref. 6) has shown the risk of having one full ECCS train inoperable to be sufficiently low to justify continued operation for 72 hours.

With one or more components inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be immediately entered.

B.1 and B.2

If the Required Action and associated Completion Time of Condition A are not met, or one or more components are inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and RCS temperature must be reduced to less than or equal to 350°F within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1

Condition A is applicable with one or more trains inoperable. The allowed Completion Time is based on the assumption that at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available. With less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the unit is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately

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SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an inoperable valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.2

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the ASME Code (Ref. 7). This testing confirms component OPERABILITY, trends performance, and detects incipient failures by indicating abnormal performance. SRs are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code.

SR 3.5.2.3

This SR demonstrates that each automatic ECCS valve actuates to the required position on an actual or simulated ESAS signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of the ESAS testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.4

The intent of this SR is to verify that the ECCS pumps are capable of automatically starting on an ESAS signal. Because of the system design configuration and the limitations imposed on pump operation during the unit conditions when this test would be conducted, this verification must be conducted through a series of sequential, overlapping or total steps in order to demonstrate functionality. SR 3.5.2.4 demonstrates that each ECCS pump would be capable of starting by verifying that its breaker closes on receipt of an actual or simulated ESAS signal. SR 3.5.2.4 works in conjunction with the Inservice Testing Program (SR 3.5.2.2) which periodically verifies the ability of the pumps to start and operate within limits, and the ESAS actuation logic testing which periodically verifies the ability of the ESAS to sense, process and generate an actuation signal.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

SR 3.5.2.5

Periodic inspections of the reactor building sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The 18 month Frequency is based on the need to perform this Surveillance during a unit outage. Operating experience has shown this Frequency to be acceptable to detect abnormal degradation.

REFERENCES

1. SAR, Section 6.
2. Letter from A. C. Thadani (NRC) to P. S. Walsh (BWOG) dated March 9, 1993.
3. 10 CFR 50.46.
4. SAR, Section 14.2.2.5.2.
5. 10 CFR 50.36.

6. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

7. ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS - Shutdown

BASES

BACKGROUND

The ANO-1 SAR states that the High Pressure Injection (HPI), Low Pressure Injection (LPI) and Core Flooding Systems are collectively designed as an Emergency Core Cooling System (ECCS) (Ref. 1). In this Technical Specification, the term ECCS refers to the components associated with the LPI system. The HPI system, in conjunction with the LPI system, is covered by LCO 3.5.2, "ECCS-Operating." The core flood tanks are addressed by LCO 3.5.1, "Core Flood Tanks (CFTs)."

In MODE 3 with RCS temperature ≤ 350°F and in MODE 4, the required trains consist of two redundant, 100% capacity low pressure injection (LPI) trains.

The LPI flow paths consist of piping, valves, heat exchangers, instruments, controls, and pumps, capable of taking suction from the borated water storage tank (BWST) and the capability to manually (locally or remotely) transfer suction to the reactor building sump such that water can be injected into the reactor vessel.

APPLICABLE SAFETY ANALYSES

The stable conditions associated with operation in MODE 3 with RCS temperature ≤ 350°F and in MODE 4, allow the operational requirements to be reduced.

In MODE 3 with RCS temperature ≤ 350°F and in MODE 4, the ECCS - Shutdown LCO satisfies Criterion 4 of 10 CFR 50.36.

LCO

In MODE 3 with RCS temperature ≤ 350°F and in MODE 4, two independent and redundant LPI trains are required to ensure sufficient LPI flow is available to the core. In MODE 3 with RCS temperature ≤ 350°F and in MODE 4, an LPI train includes the pump, heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST and the capability to manually (locally or remotely) transfer suction to the reactor building sump.

During an event requiring LPI, a flow path is required to provide water from the BWST, via the LPI pumps and their respective supply headers, to the reactor vessel. In the long term, this flow path may be switched to take its supply from the reactor building sump.

A valve that is locked, sealed, or otherwise secured in its ES position is OPERABLE.

This LCO is modified by a Note that allows a Decay Heat Removal (DHR) train to be considered OPERABLE during alignment, when aligned, or when operating for decay heat removal, if it is capable of being manually (locally or remotely) realigned to the LPI mode of operation and is not otherwise inoperable. This provision is necessary because of the dual requirements of the components that comprise the low pressure injection/decay heat removal system.

APPLICABILITY

In MODES 1 and 2 and MODE 3 with RCS temperature > 350°F, the OPERABILITY requirements for the ECCS are covered by LCO 3.5.2.

In MODE 3 with RCS temperature ≤ 350°F and in MODE 4, two OPERABLE LPI trains are acceptable on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

In MODES 5 and 6, unit conditions are such that the probability of an event requiring LPI injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation - High Water Level," and LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

If one LPI train is inoperable, the unit is not prepared to provide redundant, single failure proof LPI in response to events requiring ESAS. The 48 hour Completion Time to restore the LPI subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the unit in MODE 5, where an LPI train is not required.

B.1

When the Required Action and associated Completion Time of Condition A are not met, a controlled cooldown should be initiated. The allowed Completion Time of 24 hours is reasonable, based on operating experience, to reach MODE 5 from MODE 3 with RCS temperature ≤ 350°F in an orderly manner and without challenging unit systems.

C.1

If no LPI train is OPERABLE, the unit is not prepared to respond to an event requiring low pressure injection and may not be prepared to continue cooldown using the LPI pumps and DHR heat exchangers. The Completion Time of immediately, which would initiate action to restore at least one LPI train to OPERABLE status, ensures that prompt action is taken to restore the required LPI capacity. Normally, in MODE 4, reactor decay heat must be removed by an LPI train operating with suction from the RCS. If no LPI train is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generator(s). The alternate means of heat removal must continue until one of the inoperable LPI trains can be restored to operation so that continuation of decay heat removal (DHR) is provided.

With both DHR pumps and heat exchangers inoperable, it would be unwise to require the unit to go to MODE 5, where the only available heat removal system is the LPI trains operating in the DHR mode. Therefore, the appropriate action is to initiate measures to restore one ECCS LPI train and to continue the actions until the train is restored to OPERABLE status.

C.2

Required Action C.2 requires that the unit be placed in MODE 5 within 24 hours. This Required Action is modified by a Note that states that this Required Action is only required to be performed if one DHR train is OPERABLE. This Required Action provides for those circumstances where the LPI trains may be inoperable but are otherwise capable of providing the necessary decay heat removal. Under this circumstance, the prudent action is to remove the unit from the Applicability of the LCO and place the unit in a stable condition in MODE 5. The Completion Time of 24 hours is reasonable, based on operating experience, to reach MODE 5 in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The applicable Surveillance descriptions from Bases B 3.5.2 apply. This SR is modified by a Note that allows an LPI train to be considered OPERABLE during alignment and operation for DHR, if capable of being manually realigned (remote or local) to the LPI mode of operation and not otherwise inoperable. This allows operation in the DHR mode during MODE 4, if necessary.

REFERENCES

1. The applicable references from Bases B 3.5.2 apply.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Borated Water Storage Tank (BWST)

BASES

BACKGROUND

The BWST supports the ECCS and the Reactor Building Spray System by providing a source of borated water for ECCS and reactor building spray pump operation. In addition, the BWST supplies borated water to the refueling canal for refueling operations.

The BWST supplies two ECCS trains, each by a separate, redundant supply header. Each header also supplies one train of the Reactor Building Spray System. A motor operated isolation valve is provided in each header to allow the operator to isolate the BWST from the ECCS after the ECCS pump suction has been transferred to the reactor building sump following depletion of the BWST during a loss of coolant accident (LOCA). Use of a single BWST to supply both ECCS trains is acceptable because the BWST is a passive component, and passive failures are not assumed to occur coincidentally with the Design Basis Accident (DBA).

This LCO ensures that:

a. The BWST contains sufficient borated water to support the ECCS during the injection phase;

b. Sufficient water volume exists in the reactor building sump to support continued operation of the ECCS and reactor building spray pumps at the time of transfer to the recirculation mode of cooling; and

c. The reactor remains adequately shutdown following a LOCA.

Insufficient water inventory in the BWST could affect NPSH and result in insufficient cooling capability by the ECCS when the transfer to the recirculation mode occurs.

Improper boron concentrations could result in a reduction of adequate SDM or an excessive boric acid concentration in the core following a LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the reactor building.

APPLICABLE SAFETY ANALYSES

During accident conditions, the BWST provides a source of borated water to the high pressure injection (HPI), low pressure injection (LPI), and reactor building spray pumps. As such, it provides reactor building cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown. The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of Bases B 3.5.2, "ECCS - Operating," and B 3.6.5, "Reactor Building Spray and Cooling Systems."

The limits on level of ≥ 38.4 feet and ≤ 42 feet are the accident analysis assumed values. These levels correspond to volumes of approximately 375,096 gallons and 405,090 gallons, respectively. These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures. Sufficient deliverable volume must be available to provide the operator adequate time to prepare for switchover to reactor building sump recirculation.

A second factor that affects the minimum required BWST level is the ability to support continued ECCS pump operation after the manual transfer to recirculation occurs. When ECCS pump suction is transferred to the sump, there must be sufficient water in the sump to ensure adequate net positive suction head (NPSH) for the LPI and reactor building spray pumps. This NPSH calculation is described in the SAR (Ref. 1), and the amount of water that enters the sump from the BWST and other sources is one of the input assumptions. The calculation does not take credit for more than the minimum assumed level from the BWST.

The third factor is that the volume of water in the BWST must be within a range that will ensure the solution in the sump following a LOCA is within a specified pH range that will minimize the evolution of iodine and the effect of chloride and caustic stress corrosion cracking on the mechanical systems and components.

The fourth factor is that the volume of water in the BWST must be limited to ensure that the resulting post-LOCA maximum reactor building water level is less than that used for environmental qualification of safety related components in the reactor building.

The level limits refer to the safety analysis assumed level. A certain amount of water is unusable because of tank discharge line location and other physical characteristics, and the time assumed for the operator to accomplish swapover to the sump.

The 2270 ppm limit for minimum boron concentration was established to ensure that, following a LOCA with a minimum BWST level, the reactor will remain adequately shutdown in the cold condition following mixing of the BWST and Reactor Coolant System (RCS) water volumes.

The minimum and maximum concentration limits both ensure that the long term solution in the sump following a LOCA is within a specified pH range that will minimize the evolution of iodine and the effect of chloride and caustic stress corrosion cracking on the mechanical systems and components.

The 2670 ppm maximum limit for boron concentration in the BWST is also based on the potential for boron precipitation in the core during the long term cooling period following a LOCA. For a cold leg break, the core dissipates heat by pool nucleate boiling. Because of this boiling phenomenon in the core, the boric acid concentration will increase in this region. If allowed to proceed in this manner, a point may be reached where boron precipitation will occur in the core. B&W has evaluated the flowpath inherent in the reactor vessel internals, commonly called the leakage gap flowpath, and demonstrated that the flowpath would be sufficient by itself to preclude boron precipitation (Ref. 2). As a secondary measure, post LOCA emergency procedures direct the operator to establish dilution flow paths in the LPI System to prevent this condition by establishing a forced flow path through the core regardless of break location. These procedures are based on the minimum time in which precipitation could occur, assuming that maximum boron concentrations exist in the borated water sources used for injection following a LOCA.

The 40°F lower limit on the temperature of the solution in the BWST was established to ensure that the solution will not freeze. The 110°F upper limit on the temperature of the BWST contents is consistent with the maximum water temperature assumed in the safety analysis. These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

In MODE 1, the BWST satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3). In MODES 2, 3 and 4, the BWST satisfies Criterion 4 of 10 CFR 50.36.

LCO

The BWST exists to ensure that an adequate supply of borated water is available to cool and depressurize the reactor building in the event of a DBA; to cool and cover the core in the event of a LOCA, thereby ensuring the reactor remains adequately shutdown following a DBA; and to ensure an adequate level exists in the reactor building sump to support ECCS and reactor building spray pump operation in the recirculation mode. To be considered OPERABLE, the BWST must meet the limits for water volume, boron concentration, and temperature established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, the BWST OPERABILITY requirements are dictated by the ECCS and Reactor Building Spray System OPERABILITY requirements. Since both the ECCS and Reactor Building Spray System must be OPERABLE in MODES 1, 2, 3, and 4, the BWST must be OPERABLE to support their operation.

Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled," respectively. MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation - High Water Level," and LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation - Low Water Level."

ACTIONS

A.1

With either the BWST boron concentration or borated water temperature not within limits, the condition must be corrected within 8 hours. In this condition, neither the ECCS nor the Reactor Building Spray System may be able to perform its design functions. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the unit in a MODE in which these systems are not required. The 8 hour limit to restore the temperature or boron concentration to within limits was developed considering the time required to change boron concentration or temperature and assuming that the contents of the tank are still available for injection.

B.1

With the BWST inoperable for reasons other than Condition A (e.g., water volume), the BWST must be restored to OPERABLE status within 1 hour. In this condition, neither the ECCS nor the Reactor Building Spray System can perform its design functions. Therefore, prompt action must be taken to restore the BWST to OPERABLE status or to place the unit in a MODE in which the BWST is not required. The allowed Completion Time of 1 hour to restore the BWST to OPERABLE status is based on this condition simultaneously affecting multiple redundant trains.

C.1 and C.2

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

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

Verification every 24 hours that the BWST water temperature is within the specified temperature band ensures that the fluid will not freeze and that the fluid temperature will not be hotter than assumed in the safety analysis. These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures. The 24 hour Frequency is sufficient to identify a temperature change that would approach either temperature limit.

The SR is modified by a Note that requires the Surveillance to be performed only when ambient air temperatures are outside the operating temperature limits of the BWST. With ambient temperatures within this band, the BWST temperature should not exceed the limits.

SR 3.5.4.2

Verification every 7 days that the BWST level is ≥ 38.4 feet and ≤ 42 feet ensures that a sufficient initial supply is available for injection and to support continued ECCS pump operation on recirculation. These levels correspond to volumes of approximately 375,096 gallons and 405,090 gallons, respectively. These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures. Since the BWST level is normally stable, a 7 day Frequency has been shown to be appropriate through operating experience.

SR 3.5.4.3

Verification every 7 days that the boron concentration of the BWST fluid is ≥ 2270 ppm and ≤ 2670 ppm ensures that the reactor will remain adequately shutdown following a LOCA. These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures. Since the BWST level is normally stable, a 7 day sampling Frequency is appropriate and has been shown to be acceptable through operating experience.

REFERENCES

1. SAR, Section 6.1.

2. Letter from A. C. Thadani (NRC) to P. S. Walsh (BWOG) dated March 9, 1993.

3. 10 CFR 50.36.

ADMINISTRATIVE

A1 The designated change represents a non-technical, non-intent change to the Arkansas Nuclear One, Unit 1 (ANO-1) Current Technical Specifications (CTS) made to make the ANO-1 Improved Technical Specifications (ITS) consistent with the B&W Standard Technical Specification, NUREG-1430, Revision 1. This change does not alter the requirements of the CTS or NUREG-1430. Examples of this type of change include: wording preference; convention adoption; editorial, numbering and formatting changes; and hierarchy structure.

A2 The ANO-1 CTS Bases will be administratively deleted in their entirety in favor of the NUREG-1430 Bases. The CTS Bases will be reviewed for technical content that will be identified for retention in the ITS Bases.

A3 The CTS is marked to show the adoption of the second entry Condition for ITS 3.5.1, Condition C. Condition C is entered when the Required Actions and associated Completion Times of Condition A or B are not met or upon the declaration of two inoperable core flood tanks (CFTs). CTS 3.3.6 is not explicitly identified as "one CFT inoperable" but rather as "LCO not met." Therefore, providing an explicit ACTION rather than entering LCO 3.0.3 is consistent with CTS. Because this ITS Condition and its Required Actions are in accordance with current license requirements, the adoption of this additional entry condition is considered an administrative change.

A4 In several locations throughout the CTS, the applicability was established based on measurable parameters, such as reactor coolant system (RCS) temperature and pressure, with an additional qualifier that states "and irradiated fuel is in the core." This qualifier is shown as deleted because the definition of MODE (ITS Section 1.1) is premised on "fuel in the reactor vessel." This definition for MODE invalidates the need for the additional qualifier present in the CTS applicability statements.

A5 The CTS was marked to show the adoption of the NOTE for ITS LCO 3.5.3. The NOTE specifies that a decay heat removal (DHR) train may be considered operable during alignment and when aligned for decay heat removal, if it is capable of being manually realigned to the low pressure injection (LPI) mode of operation. The adoption of this NOTE in the CTS is considered an administrative change because it reflects necessary clarification of the OPERABILITY allowances for the LPI system when performing the decay heat removal function. Further, the NOTE is necessary because of ANO's adoption of the more restrictive Applicability requirements for the LPI system in ITS LCO 3.5.3 vice the CTS 3.3.1 requirements (ref. DOC M8). This change is consistent with TSTF-090.

A6 CTS 3.3.6 required actions for inoperability of the LPI system in MODE 3 with RCS temperature less than 350°F are segregated into ITS 3.5.3 Conditions A and B. Although ITS 3.5.3 Conditions A and B provide a different presentation of the CTS requirements, the cumulative time for restoration of one LPI train remains at 72 hours (i.e., 48 hours to restore the inoperable LPI train, and if not restored, 24 hours to be in MODE 5). Therefore, this is an administrative change because the ITS maintains the CTS action times but in a different manner of presentation.

A7 This portion of CTS 3.3.6 is not applicable to ITS 3.5.3 due to the differences in Applicability. CTS 3.3.6 was Applicable to operating conditions as well as shutdown conditions (i.e., MODES 1 through 4). ITS 3.5.3 is only Applicable in shutdown conditions (i.e., MODE 3 with RCS temperature less than or equal to 350°F and MODE 4). Therefore, those actions in CTS 3.3.6 which are associated with operating conditions which are irrelevant to ITS 3.5.3 are crossed out.

A8 CTS 3.3.3(A), (B) & (C) established the LCO requirements for the Core Flood Tanks (CFTs). Specific values were established in the LCO for tank inventory, boric acid concentration and nitrogen gas pressure. Compliance with the specifications on inventory, boric acid concentration and nitrogen gas pressure constitutes OPERABILITY of the CFTs. The values of the parameters are incorporated into ITS SR 3.5.1.2, SR 3.5.1.3, and SR 3.5.1.4. In addition, The CTS LCO established a requirement that the CFT electrically operated discharge valves be open and breakers open. These requirements are incorporated into ITS SR 3.5.1.1 and SR 3.5.1.5.

A9 CTS 3.3.1(G) established the LCO requirements for the Borated Water Storage Tank (BWST). Specific values were specified in the LCO for tank inventory, boric acid concentration and minimum temperature. Compliance with the specifications on inventory, boric acid concentration and minimum temperature constitutes OPERABILITY of the BWST. The values of these parameters are incorporated into ITS SR 3.5.4.1, SR 3.5.4.2 and SR 3.5.4.3.

A10 The CTS markup was annotated as adopting the Note from the SR 3.5.1.4 Frequency. Although not explicitly stated in Table 4.1-3, Item 3 of the CTS, the sampling requirement following CFT makeup has been interpreted as applying only to the CFT affected by the inventory addition. Thus, showing its adoption on the CTS markup is an administrative change.

A11 The CTS markup is annotated to show the adoption of Condition C for ITS

3.5.2-01 LCO 3.5.2. This entry condition addresses those situations in which at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is not available due to component inoperability that has resulted in one or more ECCS trains being declared inoperable. In this situation, the safety function provided by the ECCS is not capable of being met and the unit is operating outside of its accident analyses. Therefore, LCO 3.0.3 must be entered immediately. This change is classified as administrative because the intent of Condition C is comparable to the requirements of CTS 3.0.3, which would have been entered for this situation.

A12 Units expressed in CTS 3.3.3 were inconsistent in their application of allowances for measurement and instrumentation uncertainties. For example, the CFT required volume and pressure presented in the CTS contained instrumentation uncertainty allowances. The boron concentration presented in the CTS contained no allowances. Therefore, CTS 3.3.3(A) was modified to present the safety analysis values for the CFT tank volume and pressure. This change establishes consistency between parameters presented in the specification. This change is considered to be administrative in that the same instrumentation uncertainty allowances for these parameters will exist in the future.

A13 This page is not yet approved as provided in this package. Therefore, this markup is dependent on the expected NRC approval of the August 6, 1998, (Ref. 1CAN089801) license amendment request (LAR) related to the sodium hydroxide tank level.

TECHNICAL CHANGE -- MORE RESTRICTIVE

M1 Not used.

M2 ITS 3.5.1, Condition B will establish the Required Actions should a core flood tank (CFT) become inoperable for reasons other than boron concentration not being within its limits. Condition C will establish the Required Actions should the Required Actions and associated Completion Times of Condition A or B not be met. The Completion Time for Required Action B.1, which directs restoration of the CFT to an OPERABLE status, has been specified as being 6 hours. The Completion Time for Required Action C.1, which directs that the unit be placed in MODE 3, has also been specified as being 6 hours from entry into the Condition. In the ITS, this provides a cumulative time frame of 12 hours to be in MODE 3 (for inoperability circumstances other than boron concentration not being within its limits). While in CTS 3.3.6, the cumulative time frame for placing the unit in MODE 3 was 36 hours. The reduced time to place the unit in MODE 3 constitutes a more restrictive requirement.

In addition, the ITS 3.5.1 Completion Time for removing the unit from the Applicability of the LCO will be 12 hours following entry into Condition C. For comparable circumstances, the CTS would have allowed 72 hours to be in cold shutdown. Despite the differences in the final operating condition of the reactor, the ITS will require a faster rate of cooldown to satisfy its Required Action. This also constitutes an additional restriction on the unit.

The adoption of both of these additional restrictions is considered acceptable in light of the importance of the core flood tanks in mitigating the effects of large break LOCAs.

M3 ITS SR 3.5.1.4 Frequency for verification of Core Flood Tank (CFT) boron concentration requires that the CFT be sampled every 31 days which is consistent with sampling requirements per CTS Table 4.1-3, Item 3. In addition, the ITS and CTS require that the CFTs be sampled after inventory additions. The CTS requires sampling "after each makeup," but does not specify a time limit for the sampling. The ITS Frequency will be more restrictive than CTS requirements because sampling will be required "once within 12 hours after each solution level increase ..." This Completion Time is based on the need to clearly establish when the required sampling must be completed while taking into consideration the time necessary to recirculate the tank, obtain the sample and perform the analysis. (Also see DOC L3.) The change is consistent with the intent of NUREG-1430.

M4 ITS SR 3.5.1.1, SR 3.5.1.2, and SR 3.5.1.3 were annotated in the CTS markup as being adopted in the ITS. The SRs were not previously established in the CTS. These SRs ensure that the CFT is OPERABLE and available for injection consistent with the safety analysis. Further, the SRs provide for timely recognition of out-of-specification conditions. The adoption of the SRs constitute more restrictive requirements than those presently imposed by the CTS. The changes are consistent with NUREG-1430.

M5 ITS SR 3.5.1.5 was annotated in the CTS markup as being adopted in the ITS. This SR requires verification with a Frequency of 31 days that power is removed from each CFT isolation valve operator when RCS pressure is > 800 psig. The SR will provide assurance that the power is removed from the valve operator and that the valve is not susceptible to an active failure. Although the intent of this SR is satisfied through current administrative practices at ANO-1, its adoption in the ITS represents more restrictive requirements than those established by the CTS. This change is consistent with NUREG-1430.

M6 Not used.

M7 ITS SR 3.5.4.1, with its Note, and SR 3.5.4.2 are shown as being adopted. These SRs are adopted to ensure OPERABILITY of the BWST. SR 3.5.4.1 is modified by a Note that requires the surveillance to be performed only when ambient air temperature is below the minimum required temperature or exceeds the maximum temperature limit. The SR Frequencies are sufficient to determine an out-of-specification condition in an acceptable time frame based on operating experience. In addition, the CTS did not impose an upper limit on BWST temperature. The CTS does not contain these surveillances. Therefore, the adoption of these SRs in the ITS imposes additional requirements on the unit. The changes are consistent with NUREG-1430.

M8 CTS 3.3.1 establishes the Applicability for a variety of components, including the low pressure injection (LPI) pumps and the borated water storage tank (BWST), "whenever containment integrity is established as required by Specification 3.6.1." CTS 3.6.1 requires containment integrity whenever RCS pressure is ≥ 300 psig, RCS temperature is ≥ 200°F, and fuel is in the reactor. The ITS Applicability for the required LPI trains (ITS LCO 3.5.2 and LCO 3.5.3) and the BWST (ITS LCO 3.5.4) will be MODES 1, 2, 3 and 4. Thus, the LPI trains and BWST will be required when RCS temperature is ≥ 200°F without the RCS pressure qualifier. This results in the ITS imposing an Applicability that can occur at a significantly lower RCS temperature than that required by the combination of conditions expressed in CTS 3.6.1. This is an additional restriction on unit operation that is consistent with NUREG-1430.

M9 ITS LCO 3.5.4 Condition A is shown as being adopted on the CTS markup. LCO 3.5.4 Condition A established the Required Actions should the BWST boron concentration not be within limits or should the BWST water temperature not be within limits. The Completion Time for Required Action A.1 is 8 hours. Condition C established the Required Actions should the Required Actions and associated Completion Times of Conditions A or B not be met. Required Action C.1 directs that the unit be placed in MODE 3 within 6 hours. CTS 3.3.6 established the required actions should the requirements for BWST operability not be met. CTS 3.3.6 directed that "reactor shutdown shall be initiated and the reactor shall be in the hot shutdown condition (equivalent to ITS MODE 3) within 36 hours." This CTS completion time is significantly longer than the cumulative Completion Times of Required Action A.1 and Required Action C.1. Thus, the ITS will impose more restrictive requirements than those imposed by the CTS. The more restrictive requirements are acceptable based on the importance of the BWST and its support role for other ECCS subsystems. This is an additional restriction on unit operation that is consistent with NUREG-1430.

M10 ITS LCO 3.5.4 Condition B is shown as being adopted on the CTS markup. LCO 3.5.4 Condition B established the Required Actions should the BWST be inoperable for reasons other than Condition A (i.e., boron concentration not within limits or BWST water temperature not within limits). The Completion Time for Required Action B.1 is 1 hour. Condition C established the Required Actions should the Required Actions and associated Completion Times of Conditions A or B not be met. Required Action C.1 directs that the unit be placed in MODE 3 within 6 hours. CTS 3.3.6 established the required actions should the requirements for BWST OPERABILITY not be met. CTS 3.3.6 directed that "reactor shutdown shall be initiated and the reactor shall be in the hot shutdown condition (equivalent to ITS MODE 3) within 36 hours." This CTS completion time is significantly longer than the cumulative Completion Times of Required Action B.1 and Required Action C.1. Thus, the ITS will impose more restrictive requirements than those imposed by the CTS. The more restrictive requirements are acceptable based on the importance of the BWST and its support role for other ECCS subsystems. This change is consistent with NUREG-1430.

M11 ITS LCO 3.5.4 Condition C established the Required Actions should the Required Actions and associated Completion Times of Conditions A or B not be met. Required Action C.2 directs that the unit be placed in MODE 5 within 36 hours of entry into the Condition. CTS 3.3.6 established the required actions should the requirements for BWST OPERABILITY not be met. CTS 3.3.6 directed that "reactor shutdown shall be initiated and the reactor shall be in the hot shutdown condition (equivalent to ITS MODE 3) within 36 hours and, if not corrected, in cold shutdown condition (equivalent to MODE 5) within an additional 72 hours." This CTS completion time for transitioning the unit from MODE 3 to MODE 5 is significantly longer than the Completion Time of Required Action C.2. Thus, the ITS will impose more restrictive requirements than those imposed by the CTS. The more restrictive requirements are acceptable based on the importance of the BWST and its support role for other ECCS subsystems. This change is consistent with NUREG-1430.

M12 The CTS mark-up was annotated to indicate that ITS SR 3.5.2.1 has been adopted. SR 3.5.2.1 requires verification that each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position. SR 3.5.2.1 has a specified Frequency of 31 days. The CTS does not contain a similar surveillance requirement. Thus, the adoption of this SR results in the ITS imposing an additional restriction on the unit. The adoption of this SR is consistent with NUREG-1430.

M13 The CTS mark-up was annotated to indicate that ITS SR 3.5.2.5 has been adopted. ITS SR 3.5.2.5 requires verification that, by visual inspection, each ECCS train reactor building sump suction inlet is not restricted by debris and screens show no evidence of structural distress or abnormal corrosion. SR 3.5.2.5 has a specified Frequency of 18 months. The CTS does not contain a similar surveillance requirement. Thus, the adoption of this SR results in the ITS imposing an additional restriction on the unit. The adoption of this SR is consistent with NUREG-1430.

M14 The CTS mark-up was annotated to indicate that ITS SR 3.5.3.1 (and the associated Note) has been adopted. SR 3.5.3.1 requires that the equipment required to be OPERABLE by LCO 3.5.3 (i.e., two LPI trains) be verified OPERABLE by completion of SR 3.5.2.1, SR 3.5.2.2, SR 3.5.2.3, SR 3.5.2.4 and SR 3.5.2.5. These SRs demonstrate the OPERABILITY of the LPI trains. SR 3.5.3.1 has a specified Frequency that is in accordance with above referenced SRs. The basis for their individual SR Frequencies is given in the ITS Bases for those SRs in B 3.5.2. The adoption of this SR results in the ITS imposing an additional restriction on the unit as discussed by DOC M12 and DOC M13. The adoption of this SR is consistent with NUREG-1430.

M15 CTS 3.3.7(B) is shown as deleted. This Specification established an exception to the required actions given in CTS 3.3.6. Specifically, it allowed the unit to operate for up to 7 days with the CFT pressure or level instrument required by CTS 3.3.3(D) out of service provided the other channel (level or pressure) was still operable. This exception will not exist in the ITS. ITS 3.5.1 will require entry into Condition B should SR 3.5.1.2 or SR 3.5.1.3 not be capable of being satisfied due to the failure of their respective instrument channel. The Completion Time for Required Action B.1 is 6 hours. Thus, the ITS will be more restrictive than the CTS. This change is consistent with NUREG-1430.

M16 ITS 3.5.2 Condition B is entered when the Required Action and associated Completion Time of Condition A have not been met. Required Action B.2 specifies that the unit be placed in MODE 3 with RCS temperature ≤350°F with a Completion Time of 12 hours. CTS 3.3.6 directs that with the requirements for the specified ECCS components not met, a "reactor shutdown shall be initiated and the reactor shall be in hot shutdown condition (ITS MODE 3) within 36 hours and, if not corrected, in cold shutdown condition (ITS MODE 5) within an additional 72 hours." The adoption of Required Action B.2 in the ITS represents more restrictive requirements than those imposed by the CTS in that a cooldown to MODE 3 with RCS temperature less than or equal to 350°F would be required to take place in a significantly shorter time frame (i.e., at a faster rate) than that imposed by the CTS. The shortened time frame to be in MODE 3 with RCS temperature less than or equal to 350°F is acceptable based on the allowed Completion Time of Condition A and the length of time, based on operating experience, that it takes to reach the required unit conditions. The adoption of this Condition is consistent with NUREG-1430.

TECHNICAL CHANGE -- LESS RESTRICTIVE

L1 NUREG-1430 3.5.1, Condition A is shown as being adopted in the ITS. ITS 3.5.1 Condition A establishes the Required Actions and Completion Time should one CFT become inoperable due to its boron concentration not being within limits. The Required Action directs that the boron concentration be restored to within its limits within a Completion Time of 72 hours. CTS 3.3.6 specified the required actions should the defined OPERABILITY conditions for the CFTs (per CTS 3.3.3) not be met. CTS 3.3.6 established a requirement that a reactor shutdown be initiated and the unit be placed in hot shutdown (equivalent to ITS MODE 3) within 36 hours. The CTS does not differentiate between its required actions based on the cause of the inoperability of the CFT. The ITS will allow 72 hours for restoration of the boron concentration to within its limits. The increased restoration time is acceptable based on: 1) the otherwise OPERABLE condition of the CFT and its ability to otherwise fulfill its designated safety function, and 2) the low probability of an event requiring the injection of the contents of the CFTs. This change is consistent with NUREG-1430.

L2 NUREG-1430, LCO 3.5.1, Required Action C.2 directs that unit be placed in MODE 3 with RCS pressure less than or equal to 800 psig which removes the unit from the Applicability of LCO 3.5.1. The Completion Time for Required Action C.2 is 12 hours from time of entry into the Condition. CTS 3.3.6 would, for equivalent circumstances, direct that the unit be placed in cold shutdown within an additional 72 hours (following entry into MODE 3). The CTS has been marked to show ITS 3.5.1 Required Action C.2 and its associated Completion Time. The adoption of Required Action C.2 is consistent with NUREG-1430 and establishes Required Actions that remove the unit from the LCO Applicability. CTS 3.3.6 required actions would have directed the unit to an operating condition below the Applicability of the LCO. Because the requirement to place the unit in cold shutdown has been removed, the ITS Condition C Required Actions will be less restrictive than the CTS. This less restrictive requirement is acceptable because the ITS Required Actions function to remove the unit from an operating condition where the functional capability of the CFTs is required. This change is consistent with NUREG-1430.

L3 ITS SR 3.5.1.4 Frequency for verification of CFT boron concentration requires that the CFT be sampled every 31 days which is consistent with sampling requirements per CTS Table 4.1-3, item 3. In addition, the ITS and CTS require that the CFTs be sampled after inventory additions. The CTS requires sampling "after each makeup." The CTS does not establish any qualifiers on sampling Frequency based on the source of the makeup inventory. Therefore, the ITS Frequency will be less restrictive than current requirements because sampling will be required "once within 12 hours after each solution level increase of ≥ 0.2 feet that is not the result of addition from a source of known concentration ≥ 2270 ppm." The decreased sampling Frequency is acceptable because inventory makeup from sources that are of a known boron concentration will be capable of satisfying the boron concentration requirements of the CFTs. When inventory makeup is from a source for which the boron concentration is not established, sampling requirements are unchanged. This change is consistent with NUREG-1430. (Also see DOC M3).

L4 CTS 3.3.5 is marked as being less restrictive with respect to ITS LCO 3.5.2 and LCO 3.5.3 because this explicit requirement is not retained in the ITS. CTS 3.3.5 allowed maintenance to be performed during power operation on any component in the HPI or LPI systems provided that not more than one train was removed from service and that the maintenance would not make the train inoperable for more than 24 consecutive hours. Further, the Specification required that prior to initiating the maintenance, the redundant components were to be demonstrated to be OPERABLE within 24 hours prior to the maintenance. ITS LCO 3.5.2 and 3.5.3 will allow components to be out of service for a longer period of time. Specifically, 72 hours for an LPI or HPI train when in MODES 1 and 2 and MODE 3 with RCS temperature > 350°F. This is only allowed if at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available; thus, ensuring that the safety function is preserved. In MODE 3 with RCS temperature ≤ 350°F and in MODE 4, LCO 3.5.3 will allow one train of LPI to be out of service for up to 48 hours. Although these times are of longer duration, the act of maintenance on one train does not change the basis for believing that the redundant train is OPERABLE and capable of fulfilling its safety function. The ITS Completion Times are based on the capabilities provided by the OPERABLE train and the low probability of a design basis accident occurring during this time period. Lastly, this change establishes consistency with NUREG-1430.

L5 NUREG-1430 SR 3.5.4.3 specifies a Frequency of 7 days for verification of borated water storage tank (BWST) boron concentration. CTS Table 4.1-3 establishes a Frequency of "weekly and after each makeup." The ITS will adopt the 7 day Frequency established by NUREG-1430 SR 3.5.4.3. This is less restrictive in that inventory additions to the BWST will not immediately require sampling to verify boron concentration. This is acceptable because of: 1) the infrequent inventory additions to the BWST; 2) the generally small inventory addition and its small impact on the total BWST inventory concentration; and 3) the administrative controls used to govern inventory additions to the BWST. The adoption of this SR Frequency is consistent with NUREG-1430.

L6 ITS 3.5.3 Condition C is shown on the CTS markup as being adopted. Condition C is entered when both of the LPI trains are inoperable. ITS 3.5.3 Required Action C.1 directs that action be immediately initiated to restore at least one LPI train to an operable status. The current application of CTS requirements would direct entry into CTS 3.0.3 which states that "within 1 hour, action shall be initiated to place the unit in an operating condition (MODE) in which the Specification does not apply by placing it, as applicable, in: ... 3. At least cold shutdown (MODE 5) within the subsequent 24 hours." ITS Condition C will not direct that a reduction in operating temperature be taken until at least one LPI train has been restored to an OPERABLE status, thus ensuring that an effective method of decay heat removal will be available in the lower MODE. For those unforeseen circumstances that may result in an LPI train not being returned to service within a time period that would allow the required actions of CTS 3.0.3 to be satisfactorily completed, the imposition of this ITS Condition would be less restrictive based on the CTS requirement to place the unit in cold shutdown within 24 hours without regard for the ability to dissipate decay heat at this lower MODE. This change is acceptable because of the ITS direction that action be immediately initiated to restore a safety function (i.e., one train of LPI) while recognizing that it is an inappropriate action to direct that a unit without an OPERABLE decay heat removal system be directed to a MODE that relies on the DHR system as the mechanism for decay heat removal. The adoption of this Condition is consistent with NUREG-1430.

L7 ITS 3.5.2 Condition A is entered when one or more ECCS trains are inoperable and at least 100% of the ECCS flow equivalent to a single operable ECCS train is available. Required Action A.1 specifies that the ECCS train be restored to OPERABLE status with a Completion Time of 72 hours. ITS Condition B is entered when the Required Action and associated Completion Time of Condition A have not been met. Required Action B.1 specifies that the unit be placed in MODE 3 with a Completion Time of 6 hours. Cumulatively, under the ITS, the unit has 78 hours to be in MODE 3 (subcritical). CTS 3.3.6 requires that with the requirements for the specified ECCS components not met, a "reactor shutdown shall be initiated and the reactor shall be in hot shutdown condition within 36 hours." Thus, the adoption of the Completion Times in the ITS represent less restrictive requirements than those imposed by the CTS. The increase in the allowed restoration time is acceptable based on the preservation of the ECCS safety function provided by the redundant train and the verification that "at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train" is available. The adoption of these Completion Times is consistent with NUREG-1366, NUREG-1430 and NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975. (Also reference DOC L9).

L8 ITS 3.5.2 Condition B is entered when the Required Action and associated Completion Time of Condition A have not been met. Required Action B.2 specifies that the unit be placed in MODE 3 with RCS temperature less than or equal to 350°F within a Completion Time of 12 hours. CTS 3.3.6 directs that with the requirements for the specified ECCS components not met, a "reactor shutdown shall be initiated and the reactor shall be in hot shutdown condition (ITS MODE 3) within 36 hours and, if not corrected, in cold shutdown condition (ITS MODE 5) within an additional 72 hours." The adoption of Required Action B.2 in the ITS represents less restrictive requirements than those imposed by the CTS in that a cooldown to MODE 5 would no longer be required as a result of HPI subsystem inoperability. This is acceptable based on the preservation of the ECCS safety function in MODE 3 with RCS temperature less than or equal to 350°F through the requirements of ITS LCO 3.5.3. The combination of requirements in ITS 3.5.2 and ITS 3.5.3 result in no relaxation of the cooldown requirements for LPI subsystem inoperability. The adoption of this Required Action provides explicit guidance on exiting the LCO Applicability.

3.5.2-01 L9 Not used.

L10 CTS 3.3.1(I), CTS 3.3.2(B) and CTS 3.3.4(D) require that the engineered safety features valves for the high pressure injection (HPI) and low pressure injection (LPI) systems, and core flood tanks (CFTs) be OPERABLE or locked in the Engineered Safeguards (ES) position whenever the associated system or component is required to be OPERABLE. ITS LCOs 3.5.1, 3.5.2, and 3.5.3 will retain these requirements as a condition of system OPERABILITY. However, NUREG-1430 and the ITS allow the ES valves to be verified OPERABLE by actuation to the correct position or by being locked, sealed or otherwise secured in position. The expanded options for administratively controlling valve position will be adopted by the ITS. This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L11 CTS 3.2.1.3 is shown as deleted. This Specification established actions should the boric acid addition tank and associated piping, valves and pumps be inoperable for a period greater than 24 hours. These actions are unnecessary because they duplicate a requirement for an OPERABLE boric acid addition source from the borated water storage tank (BWST). The BWST will have appropriate controls established in ITS 3.5.4, "Borated Water Storage Tank (BWST)." In addition, this chemical addition source and its associated flow paths are not assumed to mitigate any design basis accident or transient as other systems and sources of borated water are assumed in the safety analysis. This change is consistent with NUREG-1430.

L12 CTS 3.3.7(B) is shown as deleted. This Specification established actions should one of the core flood tank (CFT) pressure or level instruments become inoperable. This Specification would have allowed continued operation of the unit for up to 7 days provided the other instrument channel (pressure or level) was operable. This Specification is an exception to CTS 3.3.6 and is deleted because it contradicts ITS

SR 3.0.1 requirements regarding the ability to satisfy an SR and demonstrate compliance with the LCO. In the ITS, failure to satisfy ITS SR 3.5.1.2 or SR 3.5.1.3 due to the absence of an instrumentation channel will result in entry into ITS 3.5.1 Condition B which has a 6 hour Completion Time. The absence of this Specification and its associated actions will result in less restrictive requirements because similar requirements will not exist in either the ITS or the TRM. The deletion of this Specification is consistent with NUREG-1430.

LESS RESTRICTIVE -- ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 This information has been moved to the Bases. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in the Bases provides adequate assurance that they will be maintained. The Bases will be controlled by the Bases Control Process in Chapter 5 of the proposed Technical Specifications. This change is consistent with NUREG-1430.

CTS Location	New Location
3.3.1(G)	Bases 3.5.4, Background
3.3.1(H)	Bases 3.5.2, LCO

LA2 Not used.

LA3 This information has been moved to the Technical Requirements Manual (TRM) or the Safety Analysis Report (SAR). This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The TRM and the SAR will be controlled by 10 CFR 50.59 and 10 CFR 50.71, as applicable. This change is consistent with NUREG-1430.

CTS Location	New Location
3.2.1	TRM
3.2.1.1	TRM
3.2.1.2	TRM
Figure 3.2-1	TRM
3.3.1(G)	SAR, Section 6.1.2.4.6
3.3.2(A)	SAR, Section 6.1.2.5
3.3.3(C)	SAR, Section 6.1.2.1.3
3.3.3(D)	TRM
Table 4.1-1, Item 25	TRM
Table 4.1-1, Item 36	TRM
4.5.1.1.1(a)	SAR, Table 6-5
4.5.1.1.1(b)	SAR, Table 6-5
4.5.1.1.2(a) & (a)(1)	SAR, Table 6-5
4.5.1.1.2(b)	SAR, Table 6-5
4.5.1.1.3	SAR, Table 6-5
4.5.1.2.1	SAR, Table 6-5
4.5.1.2.2(a)	SAR, Table 6-5
4.5.1.2.2(b)	SAR, Table 6-5

3.2 MAKEUP AND CHEMICAL ADDITION SYSTEMS

Applicability

Applies to the operational status of the makeup and the chemical addition systems.

Objective

To provide for adequate boration under all operating conditions to assure ability to bring the reactor to a cold shutdown condition.

A1

Specification

3.2.1 The reactor shall not be heated or maintained above 200°F unless the following conditions are met:

3.2.1.1 Two makeup pumps are operable except as specified in Specification 3.3.

3.2.1.2 A source of concentrated boric acid solution in addition to that in the borated water storage tank is available and operable. This requirement is fulfilled by the boric acid addition tank and one associated boric acid pump being operable. This tank shall contain at least the equivalent of the boric acid volume and concentration requirements of Figure 3.2-1 as boric acid solution with a temperature of at least 10°F above the crystallization temperature. System piping and valves necessary to establish a flow path from the tank to the makeup system shall also be operable and shall have a temperature of at least 10°F above the crystallization temperature for the concentration in the tank.

LA3
TRM

3.2.1.3 The boric acid addition tank and associated piping, valves and both pumps may be out of service for a maximum of 24 hours. After the 24 hour period, if the system is not returned to service and operable, the reactor shall be brought to the hot shutdown condition within an additional 12 hours.

L11

Bases

The makeup system and the chemical addition system provide control of the reactor coolant system boron concentration. (1) This is normally accomplished by using any of the three makeup pumps in series with a boric acid pump associated with the boric acid addition tank. The alternate method of boration will be the use of the makeup pumps taking suction directly from the borated water storage tank. (2)

The quantity of boric acid in storage from either of the two above mentioned sources is sufficient to borate the reactor coolant system to a 1% subcritical margin in the cold condition (200°F) at the worst time in core life with a stuck control rod assembly and after xenon decay.

A2

Minimum volumes (including a 10% safety factor) as specified by Figure 3.2-1 for the boric acid addition tank or an operable borated water storage tank (3) will each satisfy this requirement. The specification assures that adequate supplies are available whenever the reactor is heated above 200°F so that a single failure will not prevent boration to a cold condition. The minimum volumes of boric acid solution given include the boron necessary to account for xenon decay.

The principal method of adding boron to the primary system is to pump the concentrated boric acid solution (8700 ppm boron, minimum) into the makeup tank using the 25 gpm boric acid pumps.

A2

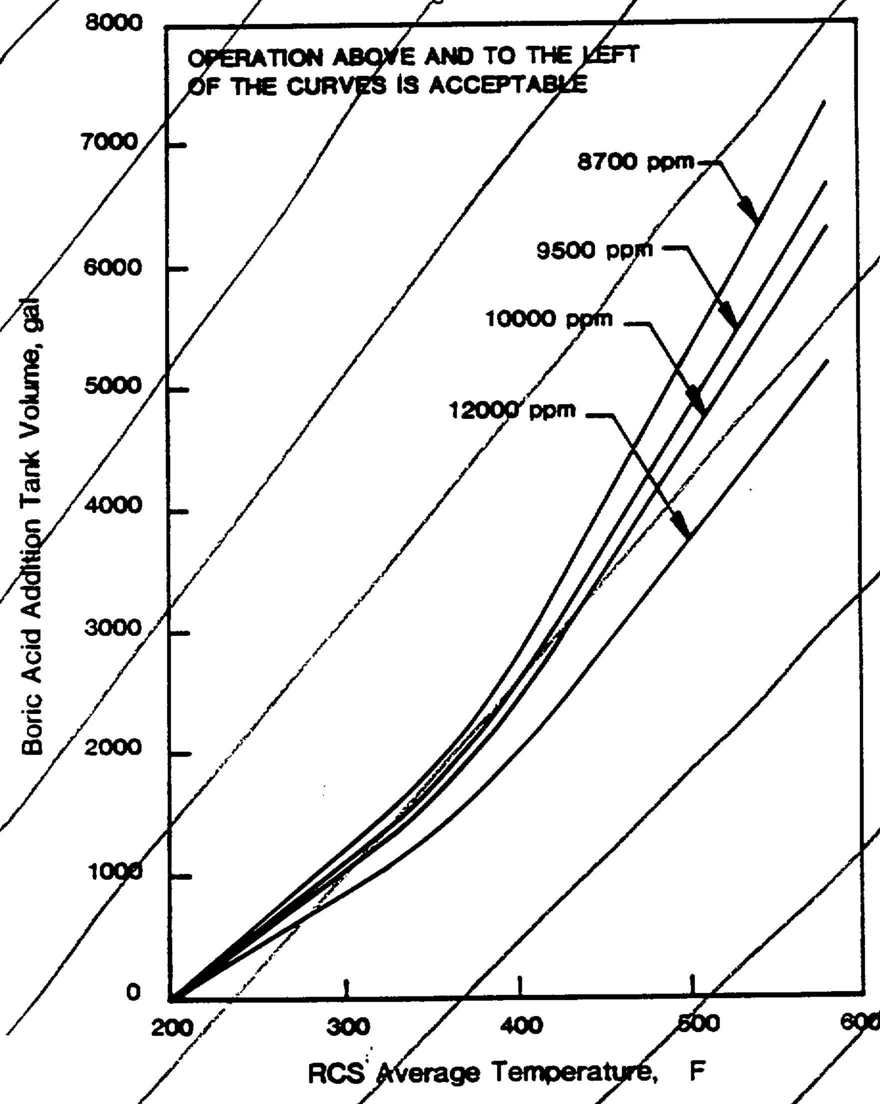
The alternate method of addition is to inject boric acid from the borated water storage tank using the makeup pumps.

Concentration of boron in the boric acid addition tank may be higher than the concentration which would crystallize at ambient conditions. For this reason and to assure a flow of boric acid is available when needed this tank and its associated piping will be kept 10°F above the crystallization temperature for the concentration present. Once in the makeup system, the concentrate is sufficiently well mixed and diluted so that normal system temperatures assure boric acid solubility.

REFERENCES

1. FSAR, Section 9.1; 9.2
2. FSAR, Figure 6-2
3. SAR, Section 3.1

Boric Acid Addition Tank Volume and
Concentration Vs RCS Average
Temperature - ANO-1, Cycle 8
Figure 3.2-1



Temp., °F	Required Volume, gal			
	8700 ppm	9500 ppm	10000 ppm	12000 ppm
579	7308	6657	6306	5200
532	6126	5580	5289	4355
500	5273	4802	4548	3749
400	2790	2543	2409	1984
300	1234	1129	1070	877
200	0	0	0	0

LA3
TRM

3.5.2
3.5.3
3.5.4

A1

3.3 EMERGENCY CORE COOLING ~~REACTOR BUILDING EMERGENCY COOLING AND REACTOR BUILDING SPRAY~~ SYSTEMS (ECCS)

~~Applicability~~

~~Applies to the emergency core cooling, reactor building emergency cooling and reactor building spray systems.~~

~~Objectivity~~

~~To define the conditions necessary to assure immediate availability of the emergency core cooling, reactor building emergency cooling and reactor building spray systems.~~

~~Specification~~ — A1

MODES 1, 2, 3, 4

3.5.2 Appl. (for LPI)
3.5.3 Appl.
3.5.4 Appl.
& ⟨LATER⟩(3.3D, 3.6, 3.7)

3.3.1 The following equipment shall be operable ~~whenever containment integrity is established as required by Specification 3.6.1~~: — M8 & LATER

⟨LATER⟩ (3.6)
~~(A) One reactor building spray pump and its associated spray nozzle header.~~
~~(B) One train of reactor building emergency cooling.~~ — LATER

⟨LATER⟩ (3.7)
~~(C) Two out of three service water pumps shall be operable, power from independent essential buses, to provide redundant and independent flow paths.~~ — LATER

3.5.2 LCO
3.5.3 LCO
(D) Two engineered safety feature actuated Low Pressure Injection (LPI) pumps shall be operable.

⟨LATER⟩ (3.7)
(E) Both low pressure injection coolers ~~and their cooling water supplies shall be operable.~~ — LATER

⟨LATER⟩ (3.3D)
~~(F) Two Borated Water Storage Tank (BWST) level instrument channel shall be operable.~~ — LATER

LA1 BASES

3.5.4 LCO
(G) The borated water storage tank shall contain a level of 40.2 ± 1.8 ft. ~~(387,400 ± 17,300 gallons)~~ of water having a concentration of 2470 ± 200 ppm boron at a temperature not less than 40F. ~~The manual valve on the discharge line from the borated water storage tank shall be locked open.~~ — A9, LA3 SAR

3.5.2 LCO
3.5.3 LCO
~~(H) The four reactor building emergency sump isolation valves to the LPI system shall be either manually or remote-manually operable.~~ — LA1 BASES

⟨Add SR 3.5.4.1 & Note⟩ — M7

⟨Add SR 3.5.4.2⟩ — M7

3.5.1
3.5.2
3.5.3
3.5.4

3.5.2 (for LPI), 3.5.3, 3.5.4 LCO & ⟨LATER⟩ (3.6, 3.7)

(I) The engineered safety features valves associated with each of the above systems shall be operable or locked, Sealed, or otherwise secured, in the ES position. L10 & LATER

A1

3.3.2 ~~In addition to 3.3.1 above~~ the following ECCS equipment shall be operable when the reactor coolant system is above 350F ~~and irradiated fuel is in the core:~~ A4 A1

3.5.2 Appl. (for HPI)

3.5.2 LCO

(A) Two ~~out of three~~ high pressure injection (makeup) pumps shall be ~~maintained~~ operable, ~~powered from independent essential buses, to provide redundant and independent flow paths.~~ LA3 SAR

(B) Engineered safety features valves associated with 3.3.2.a above shall be operable or locked, Sealed, or otherwise Secured, in the ES position. L10

A1

3.3.3 ~~In addition to 3.3.1 and 3.3.2 above, the following ECCS equipment shall be operable when the reactor coolant system is above 800 psig:~~

3.5.1 Appl.

3.5.1 LCO

(A) The two core flooding tanks shall each contain an indicated minimum of ~~13 ± 0.4 feet~~ (1040 ± ~~30~~ 70 ft³) of borated water at 600 ± ~~25~~ 40 psig. A12

(B) Core flooding tank boron concentration shall not be less than 2270 ppm boron. A8

(C) The electrically operated discharge valves from the core flood tanks shall be open. ~~and breakers locked open and tagged.~~ LA3 SAR

(D) ~~One of the two pressure instrument channels and one of the two level instrument channels per core flood tank shall be operable.~~ LA3 TRM

3.3.4 ~~The reactor shall not be made critical unless the following equipment in addition to 3.3.1, 3.3.2, and 3.3.3 above is operable.~~ LATER

⟨LATER⟩ (3.6)

~~(A) Two reactor building spray pumps and their associated spray nozzle headers and two trains of reactor building emergency cooling. The two reactor building spray pumps shall be powered from operable independent emergency buses and the two reactor building emergency cooling trains shall be powered from operable independent emergency buses.~~

~~(B) The sodium hydroxide tank shall contain a volume of ≥9,000 gallons of sodium hydroxide solution at a concentration >5.0 wt% and <16.5 wt%.~~

~~(C) All manual valves in the main discharge lines of the sodium hydroxide tanks shall be locked open.~~

⟨Add SR 3.5.1.5⟩ M5

⟨Add SR 3.5.1.1, SR 3.5.1.2, & SR 3.5.1.3⟩ M4

LAR

A13

3.5.1, 3.5.2, 3.5.3, 3.5.4 LCO

& <LATER> (3.6, 3.7)

(D) Engineered safety feature valves and interlocks associated with 3.3.1, 3.3.2, and 3.3.3 shall be operable or locked, sealed, or otherwise secured in the ES position. L10 & LATER

~~3.3.5 Maintenance shall be allowed during power operation on any component(s) in the high pressure injection, low pressure injection, service water reactor building spray and reactor building emergency cooling~~ L4 LATER

<LATER> (3.6, 3.7)

<Add 3.5.1 Condition A> L1

<Add 3.5.1 Condition C - Second entry condition> A3

See page 38-2 & 38-3

<LATER> (3.6)

systems which will not remove more than one train of each system from service. Maintenance shall not be performed on components which woul make the affected system train inoperable for more than 24 consecutiv hours. Prior to initiating maintenance on any component of a train i any system, the redundant component of that system shall be demonstra to be operable within 24 hours prior to the maintenance.

Later

See page 38-2

3.3.6 If the conditions of Specifications ~~3.3.1, 3.3.2~~, 3.3.3, ~~3.3.4 and 3.3.5~~ cannot be met except as noted in 3.3.7 below, reactor shutdown shall be initiated ~~and~~ the reactor shall be in ~~hot shutdown condition~~ MODE 3 within ~~36~~ 6 hours, and, if not corrected, in ~~cold shutdown condition~~ MODE 3 with RCS pressure ≤ 800 psig within ~~an additional 72~~ 12 hours. within 6 hours OR

M2 & LATER

L2

3.5.1 Cond. B

3.5.1 Cond. C

& <Later> (3.3D, 3.6, 3.7)

~~3.3.7 Exceptions to 3.3.6 shall be as follows:~~

<Later> (3.3D)

~~(A) If the conditions of Specification 3.3.1(F) cannot be met, reacto operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other BWST level instrument channel shall be operable.~~ Later

~~(B) If the conditions of Specification 3.3.3(D) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other CFT instrument channel (pressure o level) shall be operable.~~ [SIC] M15 L12

~~(C) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable but both reactor building spray systems are operable, restore the inoperable train of cooling to operable status within days or be in at least hot shutdown within the next 6 hours and i cold shutdown within the following 30 hours.~~

<Later> (3.6)

~~(D) If the conditions of Specification 3.3.4(A) cannot be met because two trains of the required reactor building emergency cooling are inoperable but both reactor building spray systems are operable, restore at least one train of cooling to operable status within 7 hours or be in at least hot shutdown within the next 6 hours and cold shutdown within the following 30 hours. Restore both above required cooling trains to operable status within 7 days of init loss or be in at least hot shutdown within the next 6 hours and i cold shutdown within the following 30 hours.~~ Later

3.5.2-01

<Add 3.5.2, Condition C> A11

& <LATER> (3.6, 3.7) ~~systems which will not remove more than one train of each system from service. Maintenance shall not be performed on components which would make the affected system train inoperable for more than 24 consecutive hours. Prior to initiating maintenance on any component of a train in any system, the redundant component of that system shall be demonstrated to be operable within 24 hours prior to the maintenance.~~ & LATER L4

<LATER> (3.6) Later

See page 38-1

3.5.2 RA A.1, B.1

3.5.2 RA B.2

& <Later> (3.3D, 3.6, 3.7)

3.3.6 If the conditions of Specifications 3.3.1, 3.3.2, 3.3.3, ~~3.3.4 and 3.3.5~~ cannot be met ~~except as noted in 3.3.7 below~~, reactor shutdown shall be initiated ~~and~~ within 72 hours OR the reactor shall be in ~~hot shutdown condition~~ MODE 3 within ~~36~~ 6 hours, and, if not corrected, in ~~cold shutdown condition~~ MODE 3 with RCS temperature ≤350°F within an ~~additional 72~~ 12 hours. L7 M16 L8 & LATER

3.3.7 Exceptions to 3.3.6 shall be as follows:

<Later> (3.3D) ~~(A) If the conditions of Specification 3.3.1(F) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other BWST level instrument channel shall be operable.~~ Later

See page 38-1

(B) If the conditions of Specification 3.3.3(D) cannot be met, reacto operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other CFT instrument channel (pressure level) shall be operable.

<Later> (3.6) ~~(C) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable but both reactor building spray systems are operable, restore the inoperable train of cooling to operable status within days or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.~~

~~(D) If the conditions of Specification 3.3.4(A) cannot be met because two trains of the required reactor building emergency cooling are inoperable but both reactor building spray systems are operable, restore at least one train of cooling to operable status within 7 hours or be in at least hot shutdown within the next 6 hours and cold shutdown within the following 30 hours. Restore both above required cooling trains to operable status within 7 days of initiation loss or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.~~ Later

<Add SR 3.5.3.1 with Note> M14

<Add 3.5.3 LCO Note> A5

<Add 3.5.3 Condition C> L6

& <LATER> (3.6, 3.7)

<LATER> (3.6)

~~systems which will not remove more than one train of each system from service. Maintenance shall not be performed on components which would make the affected system train inoperable for more than 24 consecutive hours. Prior to initiating maintenance on any component of a train in any system, the redundant component of that system shall be demonstrated to be operable within 24 hours prior to the maintenance.~~ & LATER L4 Later

See page 38-2

See page 38-1

3.5.3 Condition A

3.5.3 Condition B

3.3.6 If the conditions of Specifications 3.3.1, 3.3.2, 3.3.3, 3.3.4 and 3.3.5 cannot be met except as noted in 3.3.7 below, reactor shutdown shall be initiated and the reactor shall be in hot shutdown condition within 36 hours, and, if not corrected, in cold shutdown condition within an additional 72 hours. 24 within 48 hours MODE 5 A7 A6 & LATER

& <Later> (3.3D, 3.6, 3.7)

3.3.7 Exceptions to 3.3.6 shall be as follows:

<Later> (3.3D)

~~(A) If the conditions of Specification 3.3.1(F) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other BWST level instrument channel shall be operable.~~ Later

See page 38-1

(B) If the conditions of Specification 3.3.3(D) cannot be met, reacto operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other CFT instrument channel (pressure level) shall be operable.

<Later> (3.6)

~~(C) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable but both reactor building spray systems are operable, restore the inoperable train of cooling to operable status within days or be in at least hot shutdown within the next 6 hours and i cold shutdown within the following 30 hours.~~

~~(D) If the conditions of Specification 3.3.4(A) cannot be met because two trains of the required reactor building emergency cooling are inoperable but both reactor building spray systems are operable, restore at least one train of cooling to operable status within 7 hours or be in at least hot shutdown within the next 6 hours and cold shutdown within the following 30 hours. Restore both above required cooling trains to operable status within 7 days of initi loss or be in at least hot shutdown within the next 6 hours and i cold shutdown within the following 30 hours.~~ Later

<Add 3.5.4 Condition A> M9

<Add 3.5.4 Condition B> M10

See page 38-2, 38-3

systems which will not remove more than one train of each system from service. Maintenance shall not be performed on components which woul make the affected system train inoperable for more than 24 consecutiv hours. Prior to initiating maintenance on any component of a train i any system, the redundant component of that system shall be demonstra to be operable within 24 hours prior to the maintenance.

See page 38-3
See page 38-2
See page 38-1

3.5.4 RA C.1
3.5.4 RA C.2
& <Later> (3.3D, 3.6, 3.7)

3.3.6 If the conditions of Specifications 3.3.1, 3.3.2, 3.3.3, ~~3.3.4~~ and 3.3.5 cannot be met except as noted in 3.3.7 below, reactor shutdown shall be initiated and the reactor shall be in ~~hot shutdown condition~~ MODE 3 within ~~24~~ 6 hours, and, if not corrected, in ~~cold shutdown condition~~ MODE 5 within an additional ~~72~~ 30 hours.

Later (3.6) M9 M10 M11 & LATER

3.3.7 Exceptions to 3.3.6 shall be as follows:

<Later> (3.3D)

(A) If the conditions of Specification 3.3.1(F) cannot be met, reacto operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other BWST level instrument channel sh be operable.

— Later

(B) If the conditions of Specification 3.3.3(D) cannot be met, reacto operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other CFT instrument channel (pressure level) shall be operable.

See page 38-1

<Later> (3.6)

(C) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable but both reactor building spray systems are operable, restore the inoperable train of cooling to operable status within days or be in at least hot shutdown within the next 6 hours and i cold shutdown within the following 30 hours.

(D) If the conditions of Specification 3.3.4(A) cannot be met because two trains of the required reactor building emergency cooling are inoperable but both reactor building spray systems are operable, restore at least one train of cooling to operable status within 7 hours or be in at least hot shutdown within the next 6 hours and cold shutdown within the following 30 hours. Restore both above required cooling trains to operable status within 7 days of init loss or be in at least hot shutdown within the next 6 hours and i cold shutdown within the following 30 hours.

— Later

<LATER>
(3.6)

(E) ~~If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable and one reactor building spray system is inoperable, restore the inoperable spray system to operable status within 72 hours or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours. Restore the inoperable reactor building emergency cooling train to operable status within 7 days of initial loss or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.~~

LATER

Bases

A2

~~The requirements of Specification 3.3.1 assure that below 350°F, adequate long term core cooling is provided. Two low pressure injection pumps are specified. However, only one is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident.~~

~~The post-accident reactor building emergency cooling and long-term pressure reduction may be accomplished by two spray units or by a combination of one cooling train and one spray unit. Post-accident iodine removal may be accomplished by one of the two spray system strings. The specified requirements assure that the required post-accident components are available for both reactor building emergency cooling and iodine removal. Specification 3.3.1 assures that the required equipment is operable.~~

~~A train consists of two coolers and their associated fans which have sufficient capacity to meet post accident heat removal requirements. Conservatively each reactor building emergency cooling train consists of two fans powered from the same emergency bus and their associated coils, but other combinations may be justified by an engineering evaluation.~~

~~The borated water storage tank is used for three purposes:~~

~~(A) As a supply of borated water for accident conditions.~~

~~(B) As an alternate supply of borated water for reaching cold shutdown.(2)~~

~~(C) As a supply of borated water for flooding the fuel transfer canal during refueling operation.(3)~~

370,100 gallons of borated water are supplied for emergency core cooling and reactor building spray in the event of a loss-of-coolant accident. This amount fulfills requirements for emergency core cooling. Approximately 16,000 gallons of borated water are required to reach cold shutdown. The original nominal borated water storage tank capacity of 380,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature to prevent crystallization and local freezing of the boric acid. The minimum required BWST boron concentration of 2270 ppm assures that the core will be maintained at least 1 percent $\Delta k/k$ subcritical at 70°F without any control rods in the core.

A2

Specification 3.3.2 assures that above 350°F two high pressure injection pumps are also available to provide injection water as the energy of the reactor coolant system is increased.

Specification 3.3.3 assures that above 800 psig both core flooding tanks are operational. Since their design pressure is 600 ± 25 psig, they are not brought into the operational state until 800 psig to prevent spurious injection of borated water. Both core flooding tanks are specified as a single core flood tank has insufficient inventory to reflood the core.(1)

Specification 3.3.4 assures that prior to going critical the redundant train of reactor building emergency cooling and spray train are operable.

The spray system utilizes common suction lines with the low pressure injection system. If a single train of equipment is removed from either system, the other train must be assured to be operable in each system.

The volume specified by 3.3.4.B is the safety analysis volume and does not contain allowances for instrument uncertainty. 9,000 gallons corresponds to a level of approximately 26 feet at a temperature of 77°F and a NaOH concentration of 5.0 wt%. No maximum volume is specified as the value used as the maximum volume in the safety analysis bounds the physical size of the NaOH tank. Additional allowances for instrument uncertainties, as determined in Reference 6, are incorporated in the operating procedures associated with the level instrumentation used in the control room.

When the reactor is critical, maintenance is allowed per Specification 3.3.5. Operability of the specified components shall be based on the results of testing as required by Technical Specification 4.5. The maintenance period of up to 24 hours is acceptable if the operability of equipment redundant to that removed from service is demonstrated within 24 hours prior to removal. Exceptions to Specification 3.3.6 permit continued operation for seven days if one of two BWST level instrument channels is operable or if either the pressure or level instrument channel in the CFT instrument channel is operable.

In the event that the need for emergency core cooling should occur, functioning of one train (one high pressure injection pump, one low pressure injection pump, and both core flooding tanks) will protect the core and in the event of a main coolant loop severance, limit the peak clad temperature to less than 2200°F and the metal-water reaction to that representing less than 1 percent of the clad.

The service water system consists of two independent but interconnected, full capacity, 100% redundant systems, to ensure continuous heat removal.(4)

One service water pump is required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant accident.

REFERENCES

(1) FSAR, Section 14.2.5

(2) FSAR, Section 3.2

(3) FSAR, Section 9.5.2

(4) FSAR, Section 9.3.1

(5) FSAR, Section 6.3

(6) ANO Calculation 91-E-0019-01

A2

Amendment No. 10,148,
REVISED BY NRC LETTER DATED: September 13, 1991

Table 4.1-1 (Cont'd)

	Channel Description	Check	Test	Calibrate	Remarks	
<LATER> (3.3B)	20. Reactor Building Spray System System Logic Channels	NA	M(1)	NA	(1) Including RB spray pump, spray valves, and chem. add. valve logic channels.	LATER
	21. Reactor Building Spray System Analog Channels					
	a. Reactor Building Pressure Channels	NA	M	R		LATER
<LATER> (3.3D)	22. Pressurizer Temperature Channels	S	NA	R		
<LATER> (3.1)	23. Control Rod Absolute Position	S(1)	NA	R	(1) Compare with Relative Position Indicator.	LATER
	24. Control Rod Relative Position	S(1)	NA	R	(1) Check with Absolute Position Indicator	
	25. Core Flooding Tanks					LA3 TRM
	a. Pressure Channels	S	NA	R		
	b. Level Channels	S	NA	R		
<LATER> (3.3D)	26. Pressurizer Level Channels	S	NA	R		LATER
	27. Makeup Tank Level Channels	D	NA	R		
<LATER> (3.4B & 3.3D)	28. Radiation Monitoring Systems other than containment high range monitors (item 57)				(1) Check functioning of self-checking feature on each detector.	LATER
	a. Process Monitoring System	S	Q	R		
<LATER> (3.3D)	b. Area Monitoring System	S	M(1)	R		LATER
	c. Main Steam Line Radiation Monitors	S	M	R		

3.5

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
36. Boric Acid Addition Tank				LA3 TRM
a. Level Channel	NA	NA	R	
b. Temperature Channel	M	NA	R	
<LATER> (3.3 D) 37. Degraded Voltage Monitoring	W	R	R	LATER
38. Sodium Hydroxide Tank Level Indicator	NA	NA	R	R TRM
<LATER> (3.2) 39. Incore Neutron Detectors	M(1)	NA	NA	(1) Check Functioning LATER
<LATER> (3.3 D) 40. Emergency Plant Radiation Instruments	M(1)	NA	R	(1) Battery Check LATER
<LATER> (3.3A) 41. Reactor Trip Upon Turbine Trip Circuitry	M	PC	R	LATER
42. Deleted				A1

3.5

3.5.1
3.5.4

<Add SR 3.5.1.4 Frequency Note> A10

Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

	Item	Test	Frequency	
<LATER> (3.4B)	1. Reactor Coolant Samples	a. Gamma Isotopic Analysis	a. Bi-weekly (7)	LATER
		b. Gross Activity Determination	b. 3 times/week and at least every third day (1)(6)(7)	
		c. Gross Radioiodine Determination	c. Weekly (3)(6)(7)	
<LATER> (3.1)		d. Dissolved Gases	d. Weekly (7)	LATER
<LATER> (3.4A)		e. Chemistry (Cl, F, and O_2)	e. 3 times/week (8)	LATER
<LATER> (3.9)		f. Boron Concentration	f. 3 times/week	R TRM & LATER
<LATER> (3.4B)		g. Radiochemical Analysis for Ē Determination (2) (4)	g. Monthly (7)	LATER
SR 3.5.4.3	2. Borated Water Storage Tank Water Sample	Boron Concentration	7 days ~~Weekly and after each makeup~~	L5
SR 3.5.1.4	3. Core Flooding Tank Sample	Boron Concentration	31 days ~~Monthly~~ and within 12 hours after each makeup of ≥ 0.2 feet that is not the result of addition from a borated water source of known concentration ≥ 2270 ppm.	L3 M3
<LATER> (3.7)	4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly and after each makeup (9)	
	5. Secondary Coolant Samples	a. Gross Radioiodine Concentration	a. Weekly (5)(7)(10)	
		b. Isotopic Radioiodine Concentration (4)	b. Monthly (7)(10)	R TRM LATER
<LATER> (3.6)	6. Sodium Hydroxide Tank Sample	Sodium Hydroxide Concentration	Quarterly and after each makeup	LATER

<LATER> (3.4B)

Notes:

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of μCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled. Whenever the gross radioactivity concentration exceeds 10% of the limit specified in the Specification 3.1.4.1 or increases by 10 μCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established.

LATER

<Add SR 3.5.2.1> M12

<Add SR 3.5.2.5> M13

~~4.5 EMERGENCY CORE COOLING SYSTEM AND REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING~~

~~4.5.1 Emergency Core Cooling Systems~~

~~Applicability~~

~~Applies to periodic testing requirement for emergency core cooling systems.~~ A1

~~Objective~~

~~To verify that the emergency core cooling systems are operable.~~

~~Specification~~

~~4.5.1.1 System Tests~~

SR. 3.5.2.3
SR 3.5.2.4

4.5.1.1.1 High Pressure Injection System

(a) Once every 18 months, a system test shall be conducted to demonstrate that the system is operable. ~~A test signal will be applied to demonstrate actuation of the high pressure injection system for emergency core cooling operation.~~ LA3 SAR

~~(b) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly; all appropriate pump breakers shall have opened or closed and all valves shall have completed their travel.~~ LA3 SAR

SR 3.5.2.3
SR 3.5.2.4
& <Later> (3.7)

4.5.1.1.2 Low Pressure Injection System

(a) Once every 18 months, a system test shall be conducted to demonstrate that the system is operable. ~~The test shall be performed in accordance with the procedure summarized below:~~ & LATER

~~(1) A test signal will be applied to demonstrate actuation of the low pressure injection system for emergency core cooling operation.~~ LA3 SAR

<LATER> (3.7)

~~(2) Verification of the engineered safeguard function of the service water system which supplies cooling water to the decay heat removal coolers shall be made to demonstrate operability of the coolers.~~ LATER

& <LATER> (3.7)

~~(b) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly; all appropriate pump breakers shall have opened or closed, and all valves shall have completed their travel.~~ LA3 SAR & LATER

4.5.1.1.3 Core Flooding System LA3 SAR

(a) Once every 18 months, a system test shall be conducted to demonstrate proper operation of the system. During this test, verification shall be made that the check valves in the core flooding tank discharge lines operate properly.

(b) The test will be considered satisfactory if control board indication of core flood tank level verifies that all check valves have opened.

4.5.1.2 Component Tests A1

4.5.1.2.1 Pumps

SR 3.5.2.2

Approximately quarterly the high pressure and low pressure injection pumps shall be started and operated to verify proper operation. Acceptable performance will be indicated if the pump starts, operates for fifteen minutes, and the discharge pressure and flow are within ± 10% of the initial level of performance as determined using test flow paths. LA3 SAR

4.5.1.2.2 Valves - Power Operated

(a) At intervals not to exceed three months, each engineered safety feature valve in the emergency core cooling systems and each engineered safety feature valve associated with emergency core cooling in the service water system which are designed to open in the event of a LOCA shall be tested to verify operability. LA3 SAR LATER

<LATER> (3.7)

(b) The acceptable performance of each power operated valve will be that motion is indicated upon actuation by appropriate signals. LA3 SAR & LATER

& <LATER> (3.7)

Bases

The emergency core cooling systems are the principle reactor safety features in the event of a loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

The high pressure injection system under normal operating conditions has one pump operating. At least once per month, operation will be rotated to another high pressure injection pump. This will help verify that the high pressure injection pumps are operable.

The requirements of the service water system for cooling water are more severe during normal operation than under accident conditions. Rotation of the pump in operation on a monthly basis will verify that two pumps are operable.

The low pressure injection pumps are tested singularly for operability by opening the borated water storage tank outlet valves and the borated water storage tank recirc line. This allows water to be pumped from the borated water storage tank through each of the injection lines and back to the tank.

A2

~~With the reactor shutdown, the check valves in each core flooding line are checked for operability by reducing the reactor coolant system pressure until the indicated level in the core flood tanks verify that the check valves have opened.~~ A2

~~<u>REFERENCE</u>~~

~~FSAR Section 6~~

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"R" - Relocation of requirements:

Relocating requirements which do not meet the Technical Specification selection criteria to documents with an established control program allows the Technical Specifications to be reserved only for those conditions or limitations upon reactor operation which are necessary to adequately limit the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety, thereby focusing the scope of Technical Specifications.

Therefore, requirements which do not meet the Technical Specification selection criteria in 10 CFR 50.36 have been relocated to other controlled license basis documents. This regulation addresses the scope and purpose of Technical Specifications. In doing so, it establishes a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in Technical Specifications. These criteria are as follows:

Criterion 1: Installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable that is an initial condition of a design basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier.

Criterion 4: A structure, system or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

The application of these criteria is provided in the "Application of Selection Criteria to the ANO-1 Technical Specifications." Requirements which met the criteria have been included in the proposed improved Technical Specifications. Entergy Operations proposes to remove the requirements which do not meet the criteria from the Technical Specifications and relocate the requirements to a suitable owner controlled document. The requirements in the relocated Specifications will not be affected by this Technical Specification change. Entergy Operations will initially continue to perform the required operation and maintenance to assure that the requirements are satisfied. Relocating specific requirements for systems or variables will have no impact on the system's operability or the variable's maintenance, as applicable.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

License basis document control mechanisms, such as 10 CFR 50.59, 10 CFR 50.54(a)(3), and ITS Section 5, "Administrative Controls," will be utilized for the relocated Specifications as they will be placed in other controlled license basis documents. This would allow Entergy Operations to make changes to these requirements, without NRC approval, as allowed by the applicable regulatory requirements. These controls are considered adequate for assuring structures, systems and components in the relocated Specifications are maintained operable and variables in the relocated Specifications are maintained within limits.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

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

   The proposed change relocates requirements and surveillances for structures, systems, components or variables which did not meet the criteria for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the ANO-1 Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled license basis document and maintained pursuant to the applicable regulatory requirements. Therefore, this change does not involve a significant increase in the probability or consequences of an 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or change in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change 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?**

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the affected requirement will be relocated to an owner controlled license basis document for which future changes will be evaluated pursuant to the requirements of the applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"A" - Administrative changes to requirements:

Reformatting and rewording the remaining requirements in accordance with the style of the improved Babcock & Wilcox Standard Technical Specifications in NUREG-1430 will make the Technical Specifications more readily understandable to plant operators and other users. Application of the format and style will also assure consistency is achieved between specifications. As a result, the reformatting and rewording of the Technical Specifications has been performed to make them more readily understandable by plant operators and other users. During this reformatting and rewording process, no technical changes (either actual or interpretational) to the Technical Specifications were made unless they were identified and justified.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

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

   The proposed change involves reformatting and rewording of the existing Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change 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?**

   The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, there is no technical change to the requirements and therefore, there is no significant reduction in the margin of safety.

"LA" - Less restrictive, Administrative deletion of requirements:

Portions of some Specifications provide information that is descriptive in nature regarding the equipment, system(s), actions or surveillances. This information is proposed to be deleted from the specification and relocated to other license basis documents which are under licensee control. These documents include the TS Bases, Safety Analysis Report (SAR), Technical Requirements Manual, and Programs and Manuals identified in ITS Section 5, "Administrative Controls." The removal of descriptive information is permissible, because the documents containing the relocated information will be controlled through the applicable process provided by the regulatory requirements, e.g., 10 CFR 50.59, 10 CFR 50.54(a)(3), and ITS Section 5, "Administrative Controls." This will not impact the actual requirements but may provide some flexibility in how the requirement is conducted. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

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

   The proposed change relocates requirements from the Technical Specifications to other license basis documents which are under licensee control. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. Therefore, this change does not involve a significant increase in the probability or consequences of an 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change 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?**

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the Technical Specifications to other license basis documents, which are under licensee control, are the same as the existing Technical Specifications. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

"M" - More restrictive changes to requirements:

The ANO-1 Technical Specifications are proposed to be modified in some areas to impose more stringent requirements than previously identified. These more restrictive modifications are being imposed to be consistent with the improved Babcock & Wilcox Standard Technical Specifications. Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

The modification of the ANO-1 Technical Specifications and the changes made to achieve consistency within the specifications have been performed in a manner such that the most stringent requirements are imposed, except in cases which are individually evaluated.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

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

   The proposed change provides more stringent requirements for the ANO-1 Technical Specifications. These more stringent requirements are not assumed to be initiators of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes do not impact the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated for ANO-1.

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

The imposition of more stringent requirements prevents a reduction in the margin of plant safety by:

a) Increasing the analytical or safety limit,
b) Increasing the scope of the specification to include additional plant equipment,
c) Increasing the applicability of the specification,
d) Providing additional actions,
e) Decreasing restoration times,
f) Imposing new surveillances, or
g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

Entergy Operations has evaluated these proposed Technical Specification changes and has determined that they involve no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92(c) as indicated below:

NSHC 3.5 L1

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

The extension in the Completion Time for a Required Action that pertains to Core Flood Tank (CFT) inoperability (due to its boron concentration not being within limits) does not result in any hardware change nor does it alter the remaining functional capability of the other CFT. The extension in the Completion Time does not significantly increase the probability of occurrence of any analyzed event since the parameter (CFT boron concentration) is not associated with the initiation of any analyzed event. All initiation scenarios for analyzed events are unchanged as a result of this Completion Time extension. Furthermore, the proposed Completion Time extension is short. Therefore, the increase in probability of an accident during the period of CFT inoperability is not significant. Also, an extension of the Completion Time provides additional opportunity to restore compliance with the requirements while avoiding the increased potential for a transient during the shutdown process. An increase in the consequences of an evaluated accident will exist as a result of an inoperable CFT. However, the extension in the Completion Time for performance of a Required Action will not increase the consequences of an accident should one occur during the period of CFT inoperability. The Completion Time extension does not alter the assumed response of the remaining OPERABLE CFT, and other safety related structures, systems and components, to perform their specified mitigatory function. Therefore, this extension in Completion Time does not result in a significant increase in probability or consequences of a previously evaluated accident.

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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation or prompt and appropriate compensatory actions are taken. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

NSHC 3.5 L1 (continued)

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

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Completion Time has been determined appropriate based on a combination of the time required to perform the action, the relative importance of the function or parameter to be restored, and engineering judgment. Therefore, neither the reason for the inoperability nor the short extension of the Completion Time interval involves a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

<u>NSHC 3.5 L2</u>

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

The change in Required Actions for inoperable Core Flood Tanks does not result in any hardware changes. Further, the change also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment, and associated limits for the parameters, does not change (and therefore any initiation scenarios are not changed). The change limits the Required Actions to those necessary to exit the unit conditions under which the equipment is required to perform a safety function. Therefore, the change in Required Actions does not significantly increase the consequences of an accident because the change does not alter the assumed response of the equipment in performing its specified mitigation functions, or change the response of the core parameters to assumed scenarios, from that considered in the original safety analysis.

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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation or prompt and appropriate compensatory actions are taken. Thus, this change 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?

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Required Actions provide appropriate compensatory activity (i.e., exiting the Applicable conditions) based on the conditions under which the safety function is required. Therefore, the change in Required Actions does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

NSHC 3.5 L3

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

A less frequent performance of a Surveillance Requirement does not result in any hardware changes. The Frequency of performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed) and the proposed Frequency has been determined to be adequate to demonstrate the tank boron concentration is within the required parameter limits. Further, the Frequency of performance of a surveillance does not significantly increase the consequences of an accident because a change in Frequency does not change the assumed response of the equipment in performing its specified mitigation functions from that considered with the original Frequency. The core flood tank boron concentration change resulting from volume addition from a source of known concentration is a readily calculated quantity. Hence, a sample and analysis is not required to be assured of adequate boron concentration. Therefore, this change does not involve a significant increase in the probability or 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for all equipment considered in the safety analysis. Thus, this change 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?**

The proposed change does not impact maintaining acceptable boron concentration since addition from a source of known concentration results in a readily identifiable resulting concentration. Therefore, an change in the Surveillance Frequency does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

NSHC 3.5 L4

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

The extension in the Completion Time for a Required Action that pertains to inoperability of a high pressure injection (HPI) system or low pressure injection (LPI) system does not result in any hardware change nor does it alter the remaining functional capability of the remaining HPI or LPI train (which must posses at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train). All initiation scenarios for analyzed events are unchanged as a result of this Completion Time extension. Furthermore, the proposed Completion Time extension is short; therefore, the increase in probability of an accident during the period of HPI or LPI system inoperability is not significant. Also, an extension of the Completion Time provides additional opportunity to restore compliance with the requirements while avoiding the increased potential for a transient during the shutdown process. No significant increase in the consequences of an evaluated accident would exist as a result of an inoperable HPI or LPI system because of the capability of the remaining train. The extension in the Completion Time for performance of a Required Action will not increase the consequences of an accident should one occur during the period of HPI or LPI system inoperability. The Completion Time extension does not alter the assumed response of the remaining HPI train, LPI train, or other safety related structures, systems and components, to perform their specified mitigatory function. Therefore, this extension in Completion Time does not result in a significant increase in probability or consequences of a previously evaluated accident.

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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation or prompt and appropriate compensatory actions are taken. Thus, this change 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?

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Completion Time has been determined appropriate based on a combination of the time required to perform the action, the relative importance of the function or parameter to be restored, and engineering judgment. Therefore, neither the reason for the inoperability nor the short extension of the Completion Time interval involves a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

NSHC 3.5 L5

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

A less frequent performance of a Surveillance Requirement does not result in any hardware changes. The Frequency of performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed) and the proposed Frequency has been determined to be adequate to demonstrate reliable operation of the equipment. Further, the Frequency of performance of a surveillance does not significantly increase the consequences of an accident because a change in Frequency does not change the assumed response of the equipment in performing its specified mitigation functions from that considered with the original Frequency. The borated water storage tank boron concentration change resulting from volume addition is a readily calculated quantity since the volume addition is small. Hence, a sample and analysis is not required to be assured of adequate boron concentration. Therefore, this change does not involve a significant increase in the probability or 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for all equipment considered in the safety analysis. Thus, this change 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?**

Changes in the surveilled parameter have been determined to be relatively slow during the proposed intervals and the proposed Frequency has been determined to be sufficient to identify significant impact on compliance with the assumed conditions of the safety analysis. In addition, other indications continue to be available to indicate potential noncompliance. Therefore, an extended surveillance interval does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

NSHC 3.5 L6

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

The current Technical Specifications require the unit to be placed in cold shutdown if both low pressure injection (LPI) pumps or their associated flow paths are inoperable. However, the LPI components also provide the decay heat removal function. Therefore, if both LPI trains are inoperable, the unit may not be capable of continuing a normal cooldown. A change is proposed to not require the shutdown but rather require action be immediately initiated to restore an LPI train to OPERABLE status. This is consistent with the requirements for two inoperable decay heat removal loops. Inoperable LPI equipment is not considered as an initiator of any previously evaluated accident. Therefore, the change does not significantly increase the probability of an accident previously evaluated. The LPI train is considered in the mitigation of the consequences of previously evaluated accidents and the current license requires that the unit be shutdown if no LPI train is OPERABLE. However, the requirement to place the unit in cold shutdown requires the use of a decay heat removal train which is inoperable if both LPI trains are inoperable. Rather, the ITS proposes to require that action be immediately initiated to restore an LPI train to OPERABLE status. If a previously evaluated accident were to occur during this ITS time period prior to restoration, the event consequences would be equivalent to the consequences of the same event occurring during the CTS time frame allowed for a controlled shutdown with no LPI train OPERABLE. Therefore, the proposed change does not involve a significant increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation or prompt and appropriate compensatory actions are taken. Thus, this change 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?

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Required Action to initiate restoration of reliable safety related equipment has been determined appropriate based on a combination of the time required to perform the action, the relative importance of the function or parameter to be restored, and the potential impact of alternative required actions. Therefore, the proposed Required Action does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

NSHC 3.5 L7

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

An extension of the Completion Time for a Required Action does not result in any hardware changes. The Completion Time for performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment, or limit for the parameter, does not change (and therefore any initiation scenarios are not changed) and the proposed Completion Time extension is short (and therefore limits the impact on probability). Also, an extension of the Completion Time provides additional opportunity to restore compliance with the requirements and avoid the increased potential for a transient during the shutdown process. Further, the Completion Time for performance of Required Actions does not significantly increase the consequences of an accident because a change in the Completion Time does not change the assumed response of the equipment in performing its specified mitigation functions, or change the response of the core parameters to assumed scenarios, from that considered during the original Completion Time.

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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation or prompt and appropriate compensatory actions are taken. Thus, this change 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?

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Completion Time has been determined appropriate based on a combination of the time required to perform the action, the relative importance of the function or parameter to be restored, and engineering judgment. Therefore, the short extension of the Completion Time interval does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

NSHC 3.5 L8

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

The change in Required Actions for inoperable ECCS components does not result in any hardware changes. Further, the change also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment, and associated limits for the parameters, does not change (and therefore any initiation scenarios are not changed). The change limits the Required Actions to those necessary to exit the unit conditions under which the equipment is required to perform a safety function. Therefore, the change in Required Actions does not significantly increase the consequences of an accident because the change does not alter the assumed response of the equipment in performing its specified mitigation functions, or change the response of the core parameters to assumed scenarios, from that considered in the original safety analysis.

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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation or prompt and appropriate compensatory actions are taken. Thus, this change 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?

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Required Actions provide appropriate compensatory activity (i.e., exiting the Applicable conditions) based on the conditions under which the safety function is required. Therefore, the change in Required Actions does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.5.2-01

NSHC 3.5 L9

3.5DOC-L9 shown as not used.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

<u>**NHSC 3.5 L10**</u>

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

This change will introduce the option to lock, seal, or otherwise secure the engineered safeguards (ES) valves for the high pressure injection (HPI) and low pressure injection (LPI) systems and core flood tanks (CFTs) when OPERABILITY is required. Before this change, the only option was to lock the valves in the ES position. The method of verifying ES valve position is not an accident initiator and no hardware changes are proposed; therefore, the change does not significantly increase the probability of an accident. Expanding the methods available for verifying ES valve position does not significantly increase the consequences of a previously evaluated accident since the valves of interest are still placed in proper position for their safety function.

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 necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any 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 expanding the methods of securing the ES valves in their actuated position has minimal impact on the availability of the systems. Furthermore, valve position surveillance, regardless of method of verification, is considered sufficient to provide system availability in the event of an accident.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

NSHC 3.5 L11

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

The elimination of the CTS required actions for an inoperable chemical addition (boration) source does not result in any hardware or physical alteration of the unit. Further, the change does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment, and associated limits for the parameters, does not change (and therefore any initiation scenarios are not changed). The chemical addition system is not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient. Therefore, the change does not significantly increase the consequences of an accident because the change does not alter the assumed response of any equipment in performing its specified mitigation function from that considered for the original safety analysis.

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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation or prompt and appropriate compensatory actions are taken for those systems that function to mitigate design basis events and abnormalities. Thus, this change 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?**

The chemical addition system is not a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient. Prompt and appropriate ITS Required Actions have been determined for components and systems based on the safety analysis functions to be maintained. Therefore, the elimination of this CTS required action does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

<u>NSHC 3.5 L12</u>

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

The elimination of the CTS required actions for an inoperable core flood tank instrumentation channel does not result in any hardware change or physical alteration of the unit. Further, the change does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment, and associated limits for the parameters, does not change (and therefore any initiation scenarios are not changed). The core flood tank instrumentation does not constitute a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient. Therefore, the change does not significantly increase the consequences of an accident because the change does not alter the assumed response of any equipment in performing its specified mitigation function from that considered for the original safety analysis.

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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation or prompt and appropriate compensatory actions are taken for those systems that function to mitigate design basis events and abnormalities. Thus, this change 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?**

The core flood tank instrumentation does not constitute a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient. Prompt and appropriate ITS Required Actions have been determined for components and systems based on the safety analysis functions to be maintained. Therefore, the elimination of this CTS required action does not involve a significant reduction in the margin of safety.

1. ITS 3.5.1, Condition B, Completion Time has been changed from 1 hour to 6 hours. The 6 hour Completion Time of Condition B when combined with the Completion Times of Required Action C.1 are conservative with respect to the 36 hour completion time to be in hot shutdown (ITS MODE 3) as specified in CTS 3.3.6. The 6 hour Completion Time is reasonable based on: 1) the typical time needed to restore the OPERABILITY of a Core Flood Tank (CFT) if the inoperability were based on pressure, level or discharge valve position, and 2) the low probability of an event requiring the CFT to function during the 6 hour period. Further, the 6 hour Completion Time reduces the likelihood of unnecessary unit transients associated with reductions in power to comply with the short restoration periods provided for Condition B.

   The Bases have also been marked to reflect this change.

2. The Applicable Safety Analysis section of the NUREG-1430 Bases for 3.5.1 were modified to: 1) add additional details concerning the ANO CFT line break analysis, 2) include reference to the role the CFTs play in satisfying long term cooling requirements following a LOCA, and 3) delete the sentence that says the CFTs do not contribute to these long term cooling requirements. Although the CFTs play no active role in mitigating a LOCA after the blowdown phase, the borated water inventory provided by the CFTs is a contributor to, and is assumed in the accident analyses as part of , the required inventory in the reactor building that supports ECCS pump operation during the recirculation phase. This change is consistent with current license basis.

3. NUREG-1430 SR 3.5.1.4 Frequency has been modified to reflect unit specific system characteristics. The change deals with the Completion Time for performing the required sampling and the qualification of the source of the inventory increase in the core flood tank (CFT). The NUREG-1430 Completion Time of 6 hours was changed to 12 hours which reflects the time needed to recirculate the CFT following makeup, obtain the sample and then perform the sample analysis. Reference to the source of inventory was changed from the "borated water storage tank" to "a borated water source of known concentration ≥ 2270 ppm." Inventory makeup to the CFTs via the CFT Makeup Tank (T-110) can be sampled to demonstrate an acceptable boron concentration of the makeup water prior to its admittance into the CFT. A statement was added to the Bases that clarifies that a borated water source of known concentration is one that sampling has shown to have a boron concentration within CFT requirements. Non-sampled makeup or makeup from other non-verified sources will continue to require the initiation of sampling in accordance with the intent of SR 3.5.1.4 Frequency criteria. In addition, the characterization of the quantity of addition has been revised from "volume increase of ≥ [80 gallons] to level increase of ≥ 0.2 feet." This change is made for consistency with the instrumentation used by the operators to diagnose a level change in the CFT. A level change of 0.2 feet corresponds to a volume addition of approximately 102 gallons.

The Bases have also been marked to reflect this change. These changes are necessary to reflect unit specific design characteristics.

4. NUREG-1430 SR 3.5.1.5 was modified to remove reference to RCS pressure as a condition of Applicability for the SR. ANO-1 CTS 3.3.3.C requires that the CFT isolation valve breaker be open as a condition of OPERABILITY for the CFT. The removal of the pressure requirement in ITS SR 3.5.1.5 establishes that the SR is consistent with the Applicability of LCO 3.5.1 which imposes a more restrictive pressure requirement of 800 psig.

   The Bases were similarly marked. In addition, extraneous text referring to a NOTE that modified NUREG-1430 SR 3.5.1.5, that is not present in NUREG-1430, was removed. This change is entirely editorial.

5. ITS LCO 3.5.2 Applicability was modified to specify that it is applicable in MODES 1 and 2 and in MODE 3 with Reactor Coolant System (RCS) temperature greater than 350°F. This establishes an Applicability consistent with the high pressure injection (HPI) requirements of CTS 3.3.2. This change is consistent with current license basis.

   In addition, the Note modifying LCO 3.5.2 was deleted because high pressure injection (HPI) OPERABILITY is not required until RCS temperature exceeds 350°F. LTOP requirements (NUREG-1430 LCO 3.4.12) will be imposed when RCS temperature is less than 300°F. Therefore, there is no overlap of Applicability between these two Specifications and the Note is not required. This change is consistent with current license basis.

3.5.2-01

   The Bases have also been marked to reflect these changes.

6. The Bases discussion for ITS SR 3.5.2.3 and SR 3.5.2.4 (NUREG-1430 SR 3.5.2.5 and SR 3.5.2.6) was revised to separate the discussion for these two SRs. No substantive changes were made to the Bases discussion for ITS SR 3.5.2.3 (NUREG-1430 SR 3.5.2.5). However, the Bases discussion for ITS SR 3.5.2.4 was modified to reflect the CTS 4.5.1.1.1 and 4.5.1.1.2 methodology for this Surveillance. This change is necessary due to the system design configuration and the limitations imposed on pump operation during unit conditions when this Surveillance would be conducted. Specifically, the high pressure injection (HPI) pumps can not be started during unit outage conditions because they are not equipped with sufficient recirculation capability to perform this test, and they must remain secured and isolated from the RCS to prevent a possible inadvertent over-pressurization of the RCS while at this low temperature condition (LTOP). Therefore, this verification must be conducted through a series of sequential, overlapping, or total steps in order to demonstrate functionality. ESAS actuation logic testing verifies the ability of that system to generate an actuation signal. ITS SR 3.5.2.4 verifies the ability of the actuation signal to initiate closure of the breaker for each ECCS pump. ITS SR 3.5.2.2 verifies that the ECCS pumps will indeed start and operate within the limits established

by the Inservice Testing Program. In combination, these three tests verify the ability of the ESAS to actuate the ECCS pumps and their ability to perform as required. This change is consistent with current license basis as established by CTS 4.5.1.1.1 and CTS 4.5.1.1.2 and ANO-1 SAR Table 6-5.

7. ITS 3.5.1 Actions were modified to show the deletion of NUREG-1430 Condition D and the modification of NUREG-1430 Condition C to include inoperability of two core flood tanks (CFTs). NUREG-1430 Condition D required entry into LCO 3.0.3 upon inoperability of both CFTs. The requirements of LCO 3.0.3 would have provided roughly equivalent actions to the Required Actions established in Condition C. However, because of the RCS pressure based Applicability for LCO 3.5.1, LCO 3.0.3 would not have provided an explicit time frame for removing the unit from the Applicability of LCO 3.5.1 (i.e., LCO 3.0.3 does not provide direction for having RCS pressure below a particular value as a function of elapsed time from entry into the Condition). This change clarifies the Completion Time for removing the unit from the Applicability of the LCO. CTS 3.3.6 is not explicitly identified as "one CFT inoperable" but rather as "LCO not met." Therefore, providing an explicit ACTION rather than entering LCO 3.0.3 is consistent with CTS. This change is consistent with current license basis.

   The Bases were similarly marked.

8. NUREG-1430 SR 3.5.2.1, SR 3.5.2.3, SR 3.5.2.7 and SR 3.5.2.8 are shown as not adopted in the ITS. ANO-1 has no comparable requirement to these SRs. These changes are consistent with current license basis. Additional discussion follows:

   NUREG-1430 SR 3.5.2.1 - The ANO-1 ECCS trains contain no power operated valves that are required to remain de-energized or whose control circuits require key locked handswitches in order to prevent the disabling of both ECCS trains due to common mode failure. Information Notice 87-01, "RHR Valve Misalignment Causes Degradation of ECCS in PWRs" was reviewed for applicability. ANO-1 is not susceptible to the events described in the notice because of the physical independence of the ECCS trains and the absence of cross-tie valves between trains.

   NUREG-1430 SR 3.5.2.3 - Periodic performance of surveillances that require operation of the ECCS subsystems provides reasonable assurance that the ECCS piping remains full of water. In addition, procedural controls address filling and venting requirements for systems that are returned to service following maintenance activities. Therefore, there exists a high level of assurance that the majority of the ECCS piping is filled. Additionally, the physical design of the systems are such that this SR could not be applied to all portions of the piping because of the inability to perform venting operations to satisfy the SR due to the absence of vents, physical danger associated with the evolution, or due to localized radiation levels.

NUREG-1430 SR 3.5.2.7 - The ANO-1 HPI trains do not contain stop check valves whose purpose is to balance flow or prevent HPI pump runout. Throttle valves that are used for this purpose have welded stems to prevent movement; and thus, are not susceptible to repositioning. Therefore, this NUREG-1430 SR is not applicable to the unit.

NUREG-1430 SR 3.5.2.8 - The ANO-1 LPI trains do not contain automatic flow controlling throttle valves. Therefore, this NUREG-1430 SR has no applicability to the unit.

The Bases have also been marked to reflect this change.

9. NUREG-1430 SR 3.5.2.9 was renumbered as ITS SR 3.5.2.5 and modified to account for ANO-1 site specific characteristics. The ANO-1 containment is referred to as the reactor building. Reference to "suction inlet trash racks" was removed because these components are not present in the suction inlets. This change is consistent with the current license basis of the unit.

10. Insert B 3.5-14A was provided in the ITS Bases to cover the lack of Applicability of LCO 3.5.2 for MODE 3 with RCS temperature ≤ 350°F and in MODE 4. The NUREG-1430 Bases did not provide an Applicability discussion for these MODES. This insert was provided only for completeness.

11. ITS SR 3.5.1.2 was modified to specify only the volume requirement expressed in cubic feet ($ft.^3$) and not in gallons or the corresponding level indication in feet. CTS 3.3.3(A) and the safety analyses utilized a volume in cubic feet; therefore, the SR requirement expressed in cubic feet is a more appropriate requirement. This change is consistent with current license basis. In addition, statements in the Bases have been provided to explicitly establish that the safety analysis parameters presented in the ITS do not contain allowances for instrumentation uncertainty. This change is considered to be administrative in nature.

12. NUREG-1430 LCO 3.5.3 was modified to specify that two LPI trains shall be OPERABLE vice one ECCS train. This change is necessary because of the NUREG-1430 definition of an ECCS train, which states that it is composed of one HPI subsystem and one LPI subsystem. CTS 3.3.1 requirements specify that two LPI pumps shall be OPERABLE "when containment integrity is established by Specification 3.6.1." The requirements of CTS 3.6.1 have been correlated to ITS MODES 1, 2, 3 and 4. CTS 3.3.2 requirements specify that two HPI pumps shall be OPERABLE "when the reactor coolant system is above 350°F." Thus, the NUREG-1430 LCO 3.5.3 requirements are not consistent with the CTS requirements for OPERABLE LPI subsystems or HPI subsystems. Further, CTS 3.1.2.10 requires that "when the reactor coolant temperature is less than 300°F, the high pressure injection motor operated valves shall be closed with their opening control circuits for the motor

operators disabled." This requirement negates the OPERABILITY of the HPI system in MODE 4 which is defined as the RCS temperature range greater than 200°F but less than 280°F (ref. ITS Table. 1.1-1), and which coincides with the defined Applicability of LCO 3.5.3. Therefore, NUREG-1430 LCO 3.5.3 Applicability was modified to include MODE 3 with RCS temperature ≤ 350°F. This modification is necessary as described above and ensures continuity in the Applicability requirements for the LPI trains between ITS LCO 3.5.3 and LCO 3.5.2. These changes are consistent with current license basis.

In addition, the NUREG-1430 3.5.3 LCO NOTE regarding HPI was deleted because it is not pertinent to the revised LCO requirements that specify that LPI alone is the subject of the Specification. This change is consistent with current license basis.

NUREG-1430 3.5.3 LCO was also modified by TSTF-090, Rev. 1 which inserted a NOTE that states that a decay heat removal (DHR) train may be considered OPERABLE for the purposes of satisfying the LPI requirement during alignment and when aligned for DHR, if capable of being manually realigned to the ECCS mode of operation. This NOTE is necessary to preserve compliance with the LCO when the LPI train is performing its DHR function. The Note was derived from its original location in SR 3.5.3.1. The Bases annotation that the manual control can be accomplished either locally or remotely preserves current operational flexibility. This change is consistent with TSTF-090, Rev. 1, except that ANO has determined that deleting the Note from SR 3.5.3.1, per TSTF-90, Rev 1, could result in confusion with respect to the applicable SRs. For example, the train may be capable of satisfying the applicable SRs when aligned for LPI, but not when aligned for DHR. Upon realignment to LPI, all applicable SRs would again be satisfied. Retention of this Note resolves a potential conflict with SR 3.0.1.

NUREG-1430 LCO 3.5.3 Actions were significantly altered, while retaining the original intent of the Required Actions, in order to properly reflect the corrective actions should the LCO not be met. The individual ITS Conditions and their Bases will be discussed separately in the following paragraphs.

ITS Condition A

NUREG-1430 Condition B was designated as ITS Condition A. Condition A is entered with the declaration of one train of LPI being inoperable. ITS Required Action A.1 requires that the LPI train be restored to an OPERABLE status within a Completion Time of 48 hours. This Completion Time in conjunction with the Completion Time of ITS Required Action B.1 (24 hours) is in accordance with CTS 3.3.6 requirements for the restoration of OPERABILITY or completion of compensatory measures for the LPI systems. The 48 hour Completion Time is an acceptable allowance based on the fact that the redundant LPI train can still satisfy the required ECCS safety function for the specified LCO Applicability.

ITS Condition B

NUREG-1430 Condition C was designated as ITS Condition B. Condition B is entered when the Required Action and associated Completion Time of Condition A are not met. ITS Required Action B.1 requires that the unit be taken to MODE 5 within a Completion Time of 24 hours. This Completion Time in conjunction with the Completion Time of ITS Required Action A.1 (48 hours) is in accordance with CTS 3.3.6 requirements for the restoration of operability or completion of compensatory measures for the LPI systems. Further, the combination of ITS Conditions A and B preserves the philosophy of removing the unit from the MODES or other specified conditions for Applicability.

ITS Condition C

NUREG-1430 Condition A was designated as ITS Condition C. Condition C is entered when both of the required LPI trains are declared inoperable. ITS Required Action C.1 requires that action be immediately initiated to restore at least one of the two LPI trains to an OPERABLE status. This Required Action and its associated Completion Time are premised on the recognition that an ECCS safety function has been lost. Further, this Required Action and its associated Completion Time are structured such that no requirement for a reduction in RCS temperature exists (i.e., LCO 3.0.3 is not entered). If both LPI trains (and consequently both DHR trains) are inoperable, the corrective action is to restore at least one train to an OPERABLE status prior to cooling the unit down and into a MODE that requires operation of the DHR system. Required Action C.2 is inserted to provide a Required Action to place the unit in MODE 5 if an OPERABLE DHR train is available despite the inoperability of both of the LPI trains. This Required Action is conditional based on a NOTE that directs that this action is required only if one DHR train is OPERABLE. If the cause of the inoperability for both LPI trains also made the DHR trains inoperable, then no attempt to cool down the unit is required. Required Action C.2 is inserted to ensure that a cooldown to MODE 5 is initiated provided the required DHR capability exists. These changes are consistent with NUREG-1430 LCO 3.4.5 and LCO 3.4.6 Actions when a decay heat removal system is unavailable.

The Bases have also been marked to reflect these changes.

13. The CTS does not establish an upper limit on BWST temperature. An evaluation of the SAR small and large break LOCA analyses, reactor building (containment) design basis events, main steam line break analyses, steam generator tube rupture analyses, and other supporting calculations indicate that an upper limit of 110°F should be established. This temperature is above the anticipated maximum BWST temperature

attributed to the meteorological conditions expected at ANO. However, to ensure that the maximum limit is not exceeded, the SR 3.5.4.1 Note will require verification of the BWST temperature when the temperature exceeds 110°F.

The Bases have also been marked to reflect these changes.

14. Bases - The B 3.5.4 Background paragraph discussing the recirculation lines to the Borated Water Storage Tank (BWST) associated with the ECCS and containment spray pumps was deleted in its entirety because the ANO-1 unit design does not provide recirculation lines to the BWST for these components. This change is consistent with current license basis.

15. An editorial change to the Bases for the Applicability of LCO 3.5.1 was made that provides reference to the low temperature overpressure protection (LTOP) consideration extended to the CFTs. The inserted paragraph restates the Applicability of LCO 3.4.12, "Low Pressure Overpressure Protection (LTOP)." This change is provided for editorial clarification only.

16. NUREG-1430 Bases (Background and SR 3.5.1.5) text referring to an interlock associated with the CFT outlet valves and RCS pressure, and the text referring to ESAS actuated opening of the CFT outlet valves was deleted because these interlocks do not exist for the ANO-1 CFT outlet valves. These valves are operator controlled. In addition, text referring to the IEEE design requirements was removed because of the lack of applicability to ANO-1. This change is consistent with current license basis and with TSTF-316, Rev 1.

17. An editorial change to revise the Bases for ITS SR 3.5.2.2 was made. The change replaces wording in the Bases for this SR with wording that is consistent with the Bases for other SRs whose purpose is to verify pump performance. For example, the inserted wording is similar to the Bases wording for NUREG-1430 SR 3.7.5.2 which applies to inservice testing of the Emergency Feedwater Pumps.

18. ITS SR 3.5.1.4 was modified to specify only the lower boron concentration requirement in accordance with the requirements of CTS 3.3.3(B). The upper boron concentration limit will remain under administrative control. The Bases were similarly changed. This change is consistent with current license basis.

19. In multiple locations through the Bases for Section 3.5, the discussion on boron precipitation was corrected to reflect the ANO-1 credited mechanism for preventing boron precipitation in the core post LOCA. B&W has evaluated the flowpath inherent in the reactor vessel internals, commonly called the leakage gap flowpath, and demonstrated that the flowpath would be sufficient by itself to preclude boron precipitation. However, emergency procedures retain provisions for establishing flushing flow paths through the core, which would similarly prevent boron precipitation. These changes are consistent with current license basis.

20. NUREG Bases - The Criterion statement at the conclusion of the Applicable Safety Analysis section was modified at each occurrence to refer to 10 CFR 50.36 instead of the NRC Policy Statement. This is an editorial change associated with the implementation of the 10 CFR 50.36 rule changes after NUREG-1430, Revision 1 was issued.

    For ITS LCOs 3.5.1, 3.5.2, 3.5.3, and 3.5.4, the 10 CFR 50.36 Criterion satisfied by the respective ITS LCOs was modified to preserve consistency with the ANO-1 license basis. Specifically, ANO-1 safety analyses, upon which ITS LCOs 3.5.1, 3.5.2, 3.5.3, and 3.5.4 are based, were performed with the reactor in MODE 1 at RATED THERMAL POWER. The ITS Applicability for these Specifications will include other MODES and specified conditions. Thus, the Criterion statement was revised to specify that the LCO parameter satisfies Criterion 3 of 10 CFR 50.36 when in MODE 1. All other specified MODES of Applicability will satisfy Criterion 4 of 10 CFR 50.36. This change is consistent with current license basis and 10 CFR 50.36.

21. NUREG 3.5.2 Bases - Background discussion of the LPI system flowpaths for control of boron precipitation was revised to reflect the guidance currently provided in the implementing procedure. This change is consistent with the current license basis.

22. NUREG 3.5.4 Bases - Background discussion of the reactor shutdown state following a LOCA has been revised. In a Framatome Technologies Incorporated letter, FTI-99-1901, dated June 16, 1999, FTI sent the NRC a final report on PSC 1-95, Small Break LOCA Re-criticality. This report indicated that there was a possibility that re-criticality could occur due to the accumulation of de-borated water in the bottom of the steam generator. However, it was determined to be non-safety significant. This change is consistent with the current license basis.

23. NUREG-1430 3.5.4 Bases Applicable Safety Analysis is revised to delete a sentence incorrectly stating that large break LOCAs assume all control rods remain withdrawn in evaluating the core reactivity for the ensuing cold shutdown. The ANO-1 analyses for cold shutdown core reactivity following the limiting DBA LOCA assume some control rods are inserted.

    Also NUREG-1430 3.5.1 Bases Applicable Safety Analysis discussion of minimum boron concentration for the CFT is revised to match the Bases for the BWST (3.5.4 Bases). Both tanks require the same 2270 ppm boron concentration, and the Bases is revised to reflect a consistent description.

24. NUREG-1430 3.5.3 Bases Applicable Safety Analysis discussion included a basis for automatic instrumentation applicability, which is appropriately addressed in Bases for Section 3.3. It is therefore deleted from this Bases discussion.

3.5.2-01

25. Incorporated TSTF 325, Rev 0.

CTS

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Core Flood Tanks (CFTs)

LCO 3.5.1 Two CFTs shall be OPERABLE. 3.3.3(A)(B)(C) 3.3.4(D)

APPLICABILITY: MODES 1 and 2,
MODE 3 with Reactor Coolant System (RCS) pressure > ~~[750]~~ 800 psig. 3.3.3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One CFT inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours	N/A
B. One CFT inoperable for reasons other than Condition A.	B.1 Restore CFT to OPERABLE status.	~~2~~ 6 hours	3.3.6 1
C. Required Action and associated Completion Time of Condition A or B not met. OR	C.1 Be in MODE 3. AND C.2 Reduce RCS pressure to ≤ ~~[750]~~ 800 psig.	6 hours ~~[12]~~ 12 hours	3.3.6 N/A
~~D.~~ Two CFTs inoperable.	~~D.1 Enter LCO 3.0.3.~~	~~Immediately~~	7

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	CTS
SR 3.5.1.1	Verify each CFT isolation valve is fully open.	12 hours	N/A
SR 3.5.1.2	Verify borated water volume in each CFT is ≥ ~~[7555 gallons, [ ] ft]~~ 970 ft3 and ≤ ~~[8005 gallons, [ ] ft]~~ 1110 ft3.	12 hours	N/A 11
SR 3.5.1.3	Verify nitrogen cover pressure in each CFT is ≥ ~~[575]~~ 560 psig and ≤ ~~[625]~~ 640 psig.	12 hours	N/A
SR 3.5.1.4	Verify boron concentration in each CFT is ≥ 2270 ppm ~~and ≤ [3500] ppm~~.	31 days AND -----NOTE------ Only required to be performed for affected CFT --------------- Once within ~~[6]~~ 12 hours after each solution ~~volume~~ level increase of ≥ ~~[80 gallons]~~ 0.2 feet that is not the result of addition from ~~the~~ a borated water ~~storage tank~~ source of known concentration ≥ 2270 ppm.	18 Table 4.1-3 #3 N/A Table 4.1-3 #3 3

(continued)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.5.1.5	Verify power is removed from each CFT isolation valve operator ~~when RCS pressure is~~ ≥ [2000] ~~psig.~~	31 days	N/A

4

CTS

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS—Operating

3.3.1 (D)(E)
3.3.1 (H)(I)
3.3.2 (A)(D)
3.3.4.(D)

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

~~-----NOTE-----~~
~~Operation in MODE 3 with high pressure injection (HPI) de-activated in accordance with LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is allowed for up to [4] hours.~~ 5

APPLICABILITY: MODES 1, and 2, ~~and 3~~ 5 3.3.1 E 3.3.2
MODE 3 with Reactor Coolant System (RCS) temperature > 350 °F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One or more trains inoperable. ~~AND At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.~~	A.1 Restore train(s) to OPERABLE status.	72 hours	3.3.6 25
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours	3.3.6
	AND		
	B.2 Reduce RCS temperature to ≤ 350 °F ~~Be in MODE 4.~~	12 hours	5 3.3.6
C. Less than 100% of the ECCS flow equivalent to a single OPERABLE train available.	C.1 Enter LCO 3.0.3	Immediately	25 N/A

3.5.2-01

3.5.2-01

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	CTS
~~[SR 3.5.2.1~~	~~Verify the following valves are in the listed position with power to the valve operator removed.~~ ~~Valve Number / Position / Function~~ ~~[ ] [ ] [ ]~~ ~~[ ] [ ] [ ]~~ ~~. . .~~ ~~[ ] [ ] [ ]~~	~~12 hours]~~	8
SR 3.5.2.~~2~~ 1	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days	NA
~~[SR 3.5.2.3~~	~~Verify ECCS piping is full of water.~~	~~31 days]~~	8
SR 3.5.2.~~4~~ 2	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program	4.5.1.2.1
SR 3.5.2.~~5~~ 3	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~[~~18~~]~~ months	4.5.1.1.1(a) 4.5.1.1.2(a)
SR 3.5.2.~~6~~ 4	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	~~[~~18~~]~~ months	4.5.1.1.1(a) 4.5.1.1.2(a)

(continued)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
~~[SR 3.5.2.7~~	~~Verify the correct settings of stops for the following HPI stop check valves: a. [MUV-2]; b. [MUV-6], and c. [MUV-10].~~	~~[18] months]~~	8
~~[SR 3.5.2.8~~	~~Verify the flow controllers for the following LPI throttle valves operate properly: a. [DHV-110]; and b. [DHV-111].~~	~~[18] months]~~	8
SR 3.5.2.~~9~~ 5	Verify, by visual inspection, each ECCS train ~~containment~~ reactor building sump suction inlet is not restricted by debris and ~~suction inlet trash racks and~~ screens show no evidence of structural distress or abnormal corrosion.	~~[~~18~~]~~ months	N/A 9

CTS

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS—Shutdown

LCO 3.5.3 Two LPI ~~One ECCS~~ trains shall be OPERABLE. (12) 3.3.1 (D) 3.3.1 (E) 3.3.1 (H) 3.3.1 (I) 3.3.4 (D)

~~-----NOTE-----~~
~~High pressure injection (HPI) may be de-activated in accordance with LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."~~ (12) N/A

<INSERT 3.5-7A – NOTE> (12) 3.3.1

APPLICABILITY: MODE 3 with RCS temperature ≤ 350°F, MODE 4. (12)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ~~A.~~ Two LPI trains ~~Required ECCS decay heat removal (DHR) loop~~ inoperable.	C.1 ~~A.1~~ Initiate action to restore one LPI train ~~required ECCS DHR loop~~ to OPERABLE status. AND C.2 -----NOTE----- Only required if one DHR train is OPERABLE. ---------- Be in MODE 5.	Immediately 24 hours
A. ~~B.~~ One LPI train ~~Required ECCS HPI subsystem~~ inoperable.	A.1 ~~B.1~~ Restore LPI train ~~required ECCS HPI subsystem~~ to OPERABLE status.	48 ~~1~~ hours
B. ~~C.~~ Required Action and associated Completion Time of Condition A ~~B~~ not met.	B.1 ~~C.1~~ Be in MODE 5.	24 hours

N/A (12) 3.3.6 3.3.6

move

<INSERT 3.5-7A>

---NOTE---

An LPI train may be considered OPERABLE during alignment and when aligned for decay heat removal, if capable of being manually realigned to the LPI mode of operation.

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.5.3.1 -------------------NOTE------------------- An LPI ~~A DHR~~ train may be considered OPERABLE during alignment and operation for DHR, if capable of being manually realigned to the LPI ~~ECCS~~ mode of operation. --- For all equipment required to be OPERABLE, the following SRs are applicable: ~~[SR 3.5.2.1]~~ SR 3.5.2.~~2~~ 1, ~~[SR 3.5.2.3]~~ SR 3.5.2.~~4~~ 2, SR 3.5.2.~~5~~ 3, SR 3.5.2.~~6~~ 4, ~~[SR 3.5.2.7]~~ ~~[SR 3.5.2.8]~~ SR 3.5.2.~~9~~ 5.	In accordance with applicable SRs

N/A

12

N/A edit

CTS

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Borated Water Storage Tank (BWST)

LCO 3.5.4 The BWST shall be OPERABLE. 3.3.1(G) 3.3.1(I) 3.3.4(D)

APPLICABILITY: MODES 1, 2, 3, and 4. 3.3.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. BWST boron concentration not within limits. OR BWST water temperature not within limits.	A.1 Restore BWST to OPERABLE status.	8 hours	N/A
B. BWST inoperable for reasons other than Condition A.	B.1 Restore BWST to OPERABLE status.	1 hour	N/A
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours	3.3.6 3.3.6

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	CTS
SR 3.5.4.1	-------------------NOTE------------------- Only required to be performed when ambient air temperature is < ~~[~~40~~]~~°F or > ~~[100]~~ 110 °F. -- Verify BWST borated water temperature is ≥ ~~[~~40~~]~~°F and ≤ ~~[100]~~ 110 °F.	24 hours	N/A 13 N/A
SR 3.5.4.2	Verify BWST borated water ~~volume~~ level is ≥ ~~[415,200 gallons,~~ ~~[~~ 38.4 ~~]~~ ft and ≤ ~~449,000 gallons,~~ ~~[~~ 42 ~~]~~ ft].	7 days	N/A
SR 3.5.4.3	Verify BWST boron concentration is ≥ ~~[2270]~~ 2270 ppm and ≤ ~~[2450]~~ 2670 ppm.	7 days	Table 4.1-3 #2

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.1 Core Flood Tanks (CFTs)

BASES

BACKGROUND

The function of the ECCS CFTs is to supply water to the reactor vessel during the blowdown phase of a loss of coolant accident (LOCA), to provide inventory to help accomplish the refill phase that follows thereafter, and to provide Reactor Coolant System (RCS) makeup for a small break LOCA. Two CFTs are provided for these functions.

The blowdown phase cf a large break LOCA is the initial period of the transient during which the RCS departs from equilibrium conditions, and heat from fission product decay, hot internals, and the vessel continues to be transferred to the reactor coolant. The blowdown phase of the transient ends when the RCS pressure falls to a value approaching that of the ~~containment~~ reactor building atmosphere. edit

In the refill phase of a LOCA, which follows immediately, reactor coolant inventory has vacated the core through steam flashing and ejection through the break. The core is essentially in adiabatic heatup. The balance of inventory is then available to help fill voids in the lower plenum and reactor vessel downcomer so as to establish a recovery level at the bottom of the core and ongoing reflood of the core with the addition of safety injection water.

The CFTs are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The CFTs are passive components, since no operator or control actions are required for them to perform their function. Internal tank pressure is sufficient to discharge the contents of the CFTs to the RCS if RCS pressure decreases below the CFT pressure. edit
Each CFT is piped separately into the reactor vessel downcomer. The CFT injection lines are also utilized by the Low Pressure Injection (LPI) System. Each CFT is isolated from the RCS by a motor operated isolation valve and two check valves in series. edit

move

The motor operated isolation valves are normally open, with power removed from the valve motor to prevent inadvertent closure prior to or during an accident. Additionally, the valves are interlocked with RCS pressure to ensure that they will open automatically as RCS pressure is increased above 16

(continued)

BASES

BACKGROUND (continued)

~~CFT pressure and to prevent inadvertent closure prior to an accident. The valves also receive an Engineered Safety Feature Actuation System (ESFAS) signal to open. These features ensure that the valves meet the requirement of the Institute of Electrical and Electronic Engineers (IEEE) Standard 279-1971 for "operating bypasses" and that the CFTs will be available for injection without reliance on operator action.~~

16

The CFTs thus form a passive system for injection directly into the reactor vessel. Except for the core flood line break LOCA, a unique accident that also disables a portion of the injection system, both tanks are assumed to operate in the safety analyses for Design Basis Events. Because injection is directly into the reactor vessel downcomer, and because it is a passive system not subject to the single active failure criterion, all fluid injection ed is credited for core cooling.

edit

The CFT line break analysis credits only one CFT, since the tank with the broken line is assumed to empty out the break.

The CFT gas/water volumes, gas pressure, and outlet pipe size are selected to provide core cooling for a large break LOCA prior to the injection of coolant by the LPI System.

APPLICABLE SAFETY ANALYSES

The CFTs are ~~taken~~ credit ed ~~for~~ in both the large and small break LOCA analyses at full power (Ref. 1). These Design Basis Accident (DBA) analyses establish the acceptance limits for the CFTs. ~~Reference to the analyses for these DBAs is used to assess changes in the CFTs as they relate to the acceptance limits.~~ In performing the LOCA calculations, conservative assumptions are made concerning the availability of emergency Safety injection flow. ~~The assumption of the loss of offsite power is required by regulations.~~ In the early stages of a limiting large break LOCA with the loss of offsite power, the CFTs provide the sole source of makeup water to the RCS.

edit

2

edit

edit

In addition, a loss of offsite power is considered to ensure worst case conditions are postulated.

edit

This is because the LPI pumps and high pressure injection (HPI) pumps cannot deliver flow until the emergency diesel generators (EDGs) start, ~~come to rated speed,~~ and go through their timed loading sequence.

edit

The limiting large break LOCA is a double ended guillotine cold leg break at the discharge of the reactor coolant pump.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

During this event, the CFTs discharge to the RCS as soon as RCS pressure decreases below CFT pressure. ~~As a conservative estimate,~~ no credit is taken for HPI for large break LOCAs. LPI is not assumed to occur until 35 seconds after the RCS pressure decreases to the ESFAS actuation pressure. No operator action is assumed during the blowdown stage of a large break LOCA (Ref. 1). edit edit edit

The small break LOCA analysis also assumes a time delay after ESFAS actuation before pumped flow reaches the core. For the larger range of small breaks, the rate of blowdown is such that the increase in fuel clad temperature is terminated by the CFTs, with pumped flow then providing continued cooling. As break size decreases, the CFTs and HPI pumps both play a part in terminating the rise in clad temperature. As break size continues to decrease, the role of the CFTs continues to decrease until the tanks are not required and the HPI pumps become responsible for terminating the temperature increase. edit

This LCO helps to ensure that the following acceptance criteria for the ECCS established by 10 CFR 50.46 (Ref. 2) will be met following a LOCA:

a. Maximum fuel element cladding temperature of 2200°F;

b. Maximum cladding oxidation of ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction of ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; ~~and~~

d. Core maintained in a coolable geometry; and

<INSERT B3.5-3A>

~~Since the CFTs discharge during the blowdown phase of a LOCA, they do not contribute to the long term cooling requirements of 10 CFR 50.46.~~ 2

The limits for operation with a CFT that is inoperable for any reason other than the boron concentration not being within limits minimize the time that the ~~plant~~ unit is exposed to a LOCA event occurring along with failure of a CFT, which might result in unacceptable peak cladding temperatures. If edit

(continued)

<INSERT B3.5-3A>

e. Adequate long term core cooling capability is maintained.

BASES

APPLICABLE SAFETY ANALYSES (continued)

a closed isolation valve cannot be opened, or the proper water volume or nitrogen cover pressure cannot be restored, the full capability of one CFT is not available and prompt action is required to place the reactor in a MODE in which this capability is not required.

~~In addition to LOCA analyses, the CFTs have been assumed to operate to provide borated water for reactivity control for severe overcooling events such as a large steam line break (SLB).~~

~~The CFTs are part of the primary success path that functions or actuates to mitigate a DBA that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.~~

edit

The minimum volume requirement (Ref. 1) for the CFTs ensures that both CFTs can provide adequate inventory to ~~reflood~~ cover the core ~~and downcomer~~ following a LOCA for limiting large break LOCAs. The downcomer then remains flooded until the HPI and LPI systems start to deliver flow.

edit

to the 3/4 point even assuming no liquid remains in the reactor vessel

The maximum volume limit is based upon the need to maintain adequate gas volume to ensure proper injection and ensure the ability of the CFTs to fully discharge. ~~and limit the maximum amount of boron inventory in the CFTs. Values of [7555] gallons and [8005] gallons are specified. These values allow for instrument inaccuracies. Values of other parameters are treated similarly.~~

edit

11

<INSERT B 3.5-4A>

The minimum nitrogen cover pressure requirement of ~~[525]~~ 560 psig ensures that the contained gas volume will ~~generate~~ satisfy discharge flow rates during injection that ~~are consistent with those assumed in~~ the safety analysis.

<INSERT B3.5-4B>

The maximum nitrogen cover pressure limit of ~~[625]~~ 640 psig ~~ensures that the amount~~ will affect the amount and timing of CFT inventory ~~that is~~ discharged while the RCS depressurizes and ~~is therefore lost through the break, will not be larger than that predicted by the~~ safety analysis. ~~The maximum allowable boron concentration of [3500] ppm in the CFTs ensures that the sump pH will be maintained between 7.0 and 11.0 following a LOCA.~~

edit

2

edit

<INSERT B3.5-4C>

<INSERT B3.5-4D>

18

~~The minimum boron requirement of [2270] ppm is selected to ensure that the reactor will remain subcritical during the reflood stage of a large break LOCA. During a large break LOCA, all control rod assemblies are assumed not to insert~~

<INSERT B 3.5-4E>

23

(continued)

<INSERT B3.5-4A>

ANO-09

The limiting safety analysis volume requirement is 1040 ± 70 ft^3. This volume corresponds to CFT levels of ≥ 11.95 ft and ≤ 14.00 ft. These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

<INSERT B3.5-4B>

This parameter value does not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

<INSERT B3.5-4C>

Limiting the maximum pressure will therefore limit the CFT inventory lost through the break and assure that the CFT inventory injected into the RCS at the proper time is bounded by

<INSERT B3.5-4D>

This parameter value does not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

<INSERT B3.5-4E>

The 2270 ppm limit for minimum boron concentration was established to ensure that, following a LOCA with a minimum CFT level, the reactor will remain adequately shutdown in the cold condition following mixing of the CFT and Reactor Coolant System (RCS) water volumes.

This parameter value is considered to be a nominal value. No additional allowances for instrument uncertainty are required in the implementing procedures.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~into the core, and the initial reactor shutdown is accomplished by void formation during blowdown. Sufficient boron concentration must be maintained in the CFTs to prevent a return to criticality during reflood.~~ 23

~~The CFT isolation valves are not single failure proof; therefore, whenever these valves are open, power shall be removed from them. This precaution ensures that both CFTs are available during an accident. With power supplied to the valves, a single active failure could result in a valve closure, which would render one CFT unavailable for injection. Both CFTs are required to function in the event of a large break LOCA.~~ edit

In MODE 2 and MODE 3 with RCS pressure > 800 psig, the CFTs satisfy Criterion 4 of 10CFR 50.36.

In MODE 1, The CFTs satisfy Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 3). 20

LCO

The LCO establishes the minimum conditions required to ensure that the CFTs are available to accomplish their core cooling safety function following a LOCA. Both CFTs are required to function in the event of a large break LOCA. If the entire contents of both tanks are not injected during the blowdown phase of a large break LOCA, the ECCS acceptance criteria of 10 CFR 50.46 (Ref. 2) could be violated. For a CFT to be considered with ~~OPERABLE, the~~ isolation valve must be fully open, power removed above [2000] psig, and the limits established in the SRs for contained volume, boron concentration, and nitrogen cover pressure must be met. edit

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS pressure ≥ ~~750~~ 800 psig, the CFT OPERABILITY requirements are based on full power operation. Although cooling requirements may decrease as power decreases, the CFTs are still required to provide core cooling as long as elevated RCS pressures and temperatures exist. edit

~~This LCO is only applicable at pressures ≥ 750 psig. Below 750 psig, the rate of RCS blowdown is such that the safety injection pumps can provide adequate injection to ensure that peak clad temperature remains below the 10 CFR 50.46 (Ref. 2) limit of 2200°F.~~ edit

(continued)

BASES

APPLICABILITY (continued)

800

In MODE 3 with RCS pressure ≤ ~~750~~ psig, and in MODES 4, 5, and 6, the CFT motor operated isolation valves ~~are~~ may be closed to isolate the CFTs from the RCS. This allows RCS cooldown and depressurization without discharging the CFTs into the RCS or requiring depressurization of the CFTs. edit edit

<<INSERT B3.5-6A>> 15

ACTIONS

A.1

If the boron concentration of one CFT is not within limits, ~~it must be returned to within the limits within 72 hours. In this condition,~~ the ability to maintain subcriticality may be reduced, but the effects of reduced boron concentration on core subcriticality during reflood are minor. Boiling of the ECCS water in the core during reflood concentrates the boron in the saturated liquid that remains in the core. In addition, the volume of the CFT is still available for injection. Since the boron requirements are based on the average boron concentration of the total volume of two CFTs, the consequences are less severe than they would be if the contents of a CFT were not available for injection. Thus, 72 hours is allowed to return the boron concentration to within limits. edit

B.1

If one CFT is inoperable for a reason other than boron concentration, ~~the CFT must be returned to OPERABLE status within 1 hour. In this condition,~~ it cannot be assumed that the CFT will perform its required function during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the ~~1~~ 6 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable CFT to OPERABLE status. The Completion Time minimizes the time the ~~plant~~ unit is potentially exposed to a LOCA in these conditions. edit 1 edit

C.1 and C.2

If the Required Actions and ~~CFT cannot be returned to OPERABLE status within the~~ associated Completion ~~Time~~ Times of Condition A or B are not met, or if both CFTs are inoperable, the ~~plant~~ unit must be brought to a MODE in which the LCO does not apply. To achieve this edit edit 7

(continued)

<INSERT B3.5-6A>

In addition, LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)," requires that in MODE 4 when any RCS cold leg temperature is ≤ 262°F, MODE 5, and MODE 6 when the reactor vessel head is on, each CFT whose pressure is greater than or equal to the maximum RCS pressure for the existing RCS temperature allowed by the pressure and temperature limit curves provided in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," be isolated.

BASES

ACTIONS

C.1 and C.2 (continued)

status, the ~~plant~~ unit must be brought to at least MODE 3 within 6 hours and RCS pressure reduced to ≤ ~~750~~ 800 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required ~~plant~~ UNIT conditions from full power conditions in an orderly manner and without challenging ~~plant~~ unit systems.

edit
edit
edit
edit

~~D.1~~

~~If more than one CFT is inoperable, the unit is in a condition outside the accident analysis; therefore, LCO 3.0.3 must be entered immediately.~~

7

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Verification every 12 hours that each CFT isolation valve is fully open, ~~as indicated in the control room,~~ ensures that the CFTs are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in accident analysis assumptions not being met. A 12 hour Frequency is considered reasonable in view of administrative controls that ensure that a mispositioned isolation valve is unlikely.

edit

SR 3.5.1.2 and SR 3.5.1.3

Verification every 12 hours of each CFT's nitrogen cover pressure and the borated water volume is sufficient to ensure adequate injection during a LOCA. Due to the static ~~design of the CFTs~~ nature of these parameters, a 12 hour Frequency usually allows the operator to identify changes before the limits are reached. Operating experience has shown that this Frequency is appropriate for early detection and correction of off normal trends.

edit

(continued)

0.2 ft level

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.4

nature of this parameter

Surveillance once every 31 days is reasonable to verify that the CFT boron concentration is within the required limits, because the static ~~design of the CFT~~ limits the ways in which the concentration can be changed. The Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage. Sampling within ~~6~~ 12 hours after ~~an 80 gallon volume~~ increase will identify whether inleakage from the RCS has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the borated water storage tank (BWST), because the water ~~contained in the BWST~~ is within CFT boron concentration requirements. This is consistent with the recommendations of NUREG-1366 (Ref. ~~3~~ 4).

of the affected CFT

3

edit

edit

edit

The 0.2 ft increase represents approximately 102 gallons increase in volume.

a borated water source of known concentration ≥ 2270 ppm, such as

Similarly, it would not be necessary to sample the CFT following inventory additions from the CFT makeup tank if sampling has determined that the added inventory had a boron concentration within the CFT requirements.

SR 3.5.1.5

Removing power

~~Verification every 31 days that power is removed~~ from each CFT isolation valve operator ~~when the RCS pressure is ≥ [2000] psig~~ ensures that an active failure could not result in the undetected closure of a CFT motor operated isolation valve coincident with a LOCA. ~~If this closure were to occur and the postulated LOCA is a rupture of the redundant CFT inlet piping, CFT capability would be rendered inoperable. The rupture would render the tank with the open valve inoperable, and a closed valve on the other CFT would likewise render it inoperable. This would cause a loss of function for the CFTs.~~ Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that the power is removed.

4

edit

edit

~~This SR is modified by a Note that allows power to be supplied to the motor operated isolation valves when RCS pressure is ≤ [2000] psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns. Even with power supplied to the valves, inadvertent closure is prevented by the RCS pressure interlock associated with the valves. Should closure of a valve occur, in spite of the interlock, the ESFAS signal provided to the valves would open a closed valve in the event of a LOCA].~~

4

16

(continued)

BASES (continued)

REFERENCES

1. ~~F~~SAR, Section ~~[6.3]~~. 6.1 and 14.2 — edit
2. 10 CFR 50.46.
3. 10 CFR 50.36.
4. ~~3.~~ ~~Draft~~ NUREG-1366, "Improvements to Technical Specifications Surveillance Requirements," December 1992. ~~February 1990.~~ — edit

20

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.2 ECCS—Operating

BASES

edit

BACKGROUND

<INSERT B 3.5-10A>

The function of the ECCS is to provide core cooling to ensure that the reactor core is protected after any of the following accidents:

a. Loss of coolant accident (LOCA);

b. Rod ejection accident; ~~(REA),~~ edit

c. Steam generator tube rupture (SGTR); and

d. Main Steam line break ~~(SLB)~~ MSLB. edit

borated water from the borated water storage tank (BWST)

directly

There are two phases of ECCS operation: injection and recirculation. In the injection phase, ~~all injection~~ borated water from the borated water storage tank (BWST) is initially added to the Reactor Coolant System (RCS) via the cold legs and directly to the reactor vessel. After the ~~borated water storage tank (BWST)~~ BWST has been depleted, the ~~ECCS~~ recirculation phase is entered as the ~~ECCS~~ suction is transferred to the ~~containment~~ reactor building sump. edit edit edit edit edit

and

MODE 3 with RCS temperature > 350 °F,

Two redundant, 100% ~~capacity trains are~~ provided. In MODES 1, and 2, and ~~3~~, MODE 3 with RCS temperature > 350 °F, each train consists of high pressure injection (HPI) and low pressure injection (LPI) subsystems. In MODES 1, and 2, and ~~3~~, both trains must be OPERABLE. This ensures that 100% of the core cooling requirements can be provided even in the event of a single active failure. 5

reactor building

A suction header supplies water from the BWST or the ~~containment~~ reactor building sump to the ECCS pumps. Separate piping supplies each train. HPI discharges into each of the four RCS cold legs between the reactor coolant pump and the reactor vessel. LPI discharges into each of the two core flood nozzles on the reactor vessel that discharge into the vessel downcomer area. ~~Control~~ valves are set to balance the HPI flow to the RCS. This flow balance directs sufficient flow to the core to meet the analysis assumptions following a small break LOCA in one of the RCS cold legs ~~near an HPI nozzle~~. edit edit edit

The HPI pumps are capable of discharging to the RCS at an RCS pressure above the opening setpoint of the pressurizer

(continued)

<INSERT B3.5-10A>

The ANO-1 SAR states that the High Pressure Injection (HPI), Low Pressure Injection (LPI) and Core Flooding Systems are collectively designed as an Emergency Core Cooling System (ECCS) (Ref. 1). In this Technical Specification, the term ECCS refers to the components associated with the HPI and LPI systems. The core flood tanks are addressed by LCO 3.5.1, "Core Flood Tanks."

BASES

BACKGROUND (continued)

safety valves. The LPI pumps below ~~are capable of discharging~~ to the RCS at ~~an RCS~~ pressures ~~of~~ approximately 200 psia. When the BWST has been nearly emptied, the suction for the LPI pumps is manually transferred to the reactor building ~~containment~~ sump. The HPI pumps cannot take suction directly from the sump. If HPI is still needed, a cross connect from the discharge side of the LPI pump to the suction of the HPI pumps would be opened. This is known as "piggy backing" ~~HPI to LPI~~ and enables continued HPI to the RCS, if needed, ~~after~~ the BWST is emptied. edit edit edit 19

<INSERT B3.5-11A>

In the long term cooling period, ~~f~~Flow paths in the LPI System ~~are~~ may be procedurally established to preclude the possibility of boric acid in the core region reaching an unacceptably high concentration. ~~One flow path is from the hot leg through the decay heat suction line from the hot leg and then in a reverse direction through the containment sump outlet line into the sump. The other flow path is through the pressurizer auxiliary spray line from one LPI train into the pressurizer and through the hot leg into the top region of the core.~~ edit 21

<INSERT B3.5-11B>

The HPI subsystem also functions to supply borated water to the reactor core following increased heat removal events, such as large ~~SLBs~~ MSLBs. edit

~~During low temperature conditions in the RCS, limitations are placed on the maximum number of ECCS pumps that may be OPERABLE. Refer to the Bases for LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," for the basis of these requirements.~~ 5

During a large break LOCA, RCS pressure will rapidly decrease. ~~to ≤ 200 psia in ≤ 20 seconds~~. The ECCS is actuated upon receipt of an Engineered Safeguards ~~Safety Feature~~ Actuation System (ESAS) signal. ~~The actuation of safeguard loads is accomplished in a programmed time sequence.~~ If offsite power has not been lost, ~~is available~~, the safeguard loads start ~~immediately~~ Unless previously operating. ~~in the programmed~~ sequence. If offsite power has been lost, ~~is not available~~ the Engineered Safeguards ~~safety feature~~ (ES) buses shed normal operating loads and are connected to the diesel generators. Safeguard loads are then connected ~~actuated~~ in ~~the programmed time~~ sequence. The time delay associated with diesel starting, sequenced loading, and pump starting determines the amount of time ~~required~~ before pumped flow is available to the core following a LOCA. edit edit edit edit edit edit

(continued)

<INSERT B3.5-11A>

the flowpath inherent in the reactor vessel internals, commonly called the leakage gap flowpath, would be sufficient by itself to preclude boron precipitation (Ref. 2).

<INSERT B3.5-11B>

The desired flowpath establishes decay heat removal (DHR) in conjunction with LPI cooling. This requires conditions present which allow both DHR pumps to operate simultaneously. If DHR can not be established but hot leg level is above the bottom of the hot leg nozzle, an alternate flowpath is gravity draining from the decay heat suction piping through the idle DHR pump into the reactor building sump. If the first two methods are unsuccessful, the pressurizer auxiliary spray line is used. This provides reverse flow through the core using auxiliary spray into the pressurizer, out the pressurizer into the hot leg via the surge line then reactor vessel into the area above the core.

BASES

BACKGROUND (continued)

The active ECCS components, along with the passive core flood tanks (CFTs) ~~and the BWST~~ covered in LCO 3.5.1, "Core Flood Tanks (CFTs)," and the BWST covered in LCO 3.5.4, "Borated Water Storage Tank (BWST)," provide the cooling water necessary to meet 10 CFR 50.46 (Ref. 1 and 3). edit edit edit

APPLICABLE SAFETY ANALYSES

The LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 1, 3 and 4), will be met following a LOCA: edit

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react; edit

d. Core is maintained in a coolable geometry; and

e. Adequate long term core cooling capability is maintained. edit

~~The LCO also helps ensure that containment temperature limits are met.~~

Only the ~~Both HPI and~~ LPI subsystem~~s~~ is ~~are~~ assumed to provide injection ~~be OPERABLE~~ in the large break LOCA analysis at full power (Ref. 4). This analysis establishes a minimum required flow for the ~~HPI and~~ LPI subsystem ~~pumps~~, as well as the minimum required response time for ~~their~~ actuation. The HPI ~~pump~~ is credited in the small break LOCA analysis. This analysis establishes the flow and discharge head requirements ~~at the design point~~ for the HPI pump. The SGTR and MSLB ~~SLB~~ analyses also credit the HPI Subsystem ~~pump~~ but are not limiting in HPI Subsystem ~~their~~ design. edit edit edit

The large break LOCA event assumes ~~with~~ a loss of offsite power and a single failure (disabling one ECCS train). ~~establishes the OPERABILITY requirements for the ECCS.~~ <INSERT B3.5-12A> During the blowdown stage of a LOCA, the RCS depressurizes as primary coolant is ejected through the break into the reactor building ~~containment~~. The nuclear edit edit edit

(continued)

<INSERT B3.5-12A>

For analysis purposes, the loss of offsite power assumption may be conservatively inconsistent with the assumed operation of some equipment, such as reactor coolant pumps (Ref. 4).

BASES

APPLICABLE SAFETY ANALYSES (continued)

reaction is terminated either by moderator voiding during large breaks or CONTROL ROD ~~assembly~~ insertion for small breaks (Ref. 4). Following depressurization, emergency cooling water is injected into the reactor vessel core flood nozzles, then flows into the downcomer, fills the lower plenum, and refloods the core. edit edit

The ~~LCO ensures~~ safety analyses show that an ~~ECCS~~ LPI train will deliver sufficient water to match decay heat boiloff rates soon enough to minimize core uncovery for a large break LOCA. ~~It~~ They also ~~ensures~~ show that the HPI ~~pump~~ train will deliver sufficient water for a small break LOCA and provide sufficient boron to maintain the core subcritical. edit edit edit edit

In the large break LOCA analyses, ~~HPI and~~ LPI ~~are~~ is not credited until at least 35 seconds after actuation of the ESPAS signal. This is based on a loss of offsite power and the associated time delays in startup and loading of the emergency diesel generator (EDG). Further, LPI flow is not credited until RCS pressure drops below the pump's shutoff head. For a large break LOCA, HPI is not credited at all. edit edit edit edit edit

<Insert B3.5-13A>

In MODE 1, The ECCS trains satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10CFR50.36 (Ref. 5). 20

<Insert B3.5-13B>

LCO

In MODES 1 and 2, and ~~3~~ MODE 3 with RCS temperature > 350°F, two independent (and redundant) ECCS trains are required to ensure that at least one is available, assuming a single failure in the other train. ~~Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.~~ 5 edit

In MODES 1 and 2, and ~~3~~ MODE 3 with RCS temperature > 350°F, an ECCS train consists of an HPI subsystem and an LPI subsystem. Each train includes the pumps, valves, heat exchangers, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an ESPAS signal and the capability to manually transfer ~~ring~~ suction to the ~~containment~~ reactor building sump. 5 edit edit edit

During an event requiring ECCS actuation, a flow path is provided to ensure an abundant supply of water from the BWST to the RCS via the HPI and LPI pumps and their respective discharge flow paths to each of the four cold leg injection nozzles and the reactor vessel. In the long term, this flow

(continued)

<INSERT B3.5-13A>

In the small break LOCA analysis, HPI is not credited until at least 35 seconds after actuation of the ESAS signal. This is based on a loss of offsite power and the associated time delays in startup and loading of the DG.

<INSERT B3.5-13B>

In MODE 2 and MODE 3 with RCS temperature > 350°F, the ECCS trains satisfy Criterion 4 of 10CFR50.36.

BASES

LCO (continued)

path may be manually transferred to take its supply from the ~~containment~~ reactor building sump and to supply ~~its flow~~ borated water to the RCS via two paths~~, as described in the Background section.~~ (LPI and HPI piggy-back modes). The flow path ~~for each train must maintain its~~ designed independence to ensure that no single failure can disable both ECCS trains. edit edit edit

~~As indicated in the Note, operation in MODE 3 with ECCS trains de-activated pursuant to LCO 3.4.12 is necessary for plants with an LTOP System arming temperature at or near the MODE 3 boundary temperature of [350]°F. LCO 3.4.12 requires that certain components be de-activated at and below the LTOP System arming temperature. When this temperature is at or near the MODE 3 boundary temperature, time is needed to restore the systems to OPERABLE status.~~ 5

APPLICABILITY

In MODES 1~~,~~ and 2~~, and 3~~, MODE 3 with RCS temperature > 350 °F, the ECCS train OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. ~~The HPI pump performance is based on the small break LOCA, which establishes the pump performance curve and is less dependent on power.~~ The HPI pump performance requirements are based on a small break LOCA. ~~MODES 2 and 3 requirements are bounded by the MODE 1 analysis.~~ 5 edit edit 10

<INSERT B3.5-14A>

In MODES 5 and 6, ~~plant~~ unit conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation—High Water Level," and LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation—Low Water Level." edit

(continued)

<INSERT B3.5-14A>

In MODE 3 with RCS temperature ≤ 350°F and in MODE 4, ECCS train OPERABILITY requirements are established by LCO 3.5.3, "ECCS - Shutdown." In MODE 3 with RCS temperature ≤ 350°F and in MODE 4, the probability of an event requiring ECCS actuation is significantly lessened. In this operating condition, the safety injection function is preserved through LCO 3.5.3 requirements for two OPERABLE LPI trains.

BASES (continued)

ACTIONS

A.1

inoperable, but

With one or more trains ~~operable and~~ at least 100% of the injection flow equivalent to a single OPERABLE ECCS train still available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on NRC recommendations (Ref. 3) that are based on a risk evaluation and is a reasonable time for ~~many~~ repairs. 6 edit edit edit edit

~~An ECCS train is inoperable if it is not capable of delivering the design flow to the RCS.~~ edit

The LCO requires the OPERABILITY of a number of independent subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two diverse ~~different~~ components, each in a different train, necessarily result in a loss of function for the ECCS. ~~The intent of this Condition is to maintain a combination of equipment such that 100% of the safety injection flow equivalent to 100% of a single train remains available.~~ This allows increased flexibility in unit ~~plant~~ operations under circumstances when diverse components in opposite trains are inoperable. edit 25 edit edit edit

i.e., an HPI subsystem in one train and an LPI subsystem in the opposite train.

3.5.2-01

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 3) has shown the risk of having one full ECCS train inoperable to be sufficiently low to justify continued operation for 72 hours. 6 edit edit

3.5.2-01

With one or more components inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the unit ~~facility~~ is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be immediately entered. edit

or one or more components are inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available,

B.1 and B.2

If the Required Action and ~~inoperable components cannot be returned to OPERABLE status within the~~ associated Completion Time of Condition A are not met, the unit ~~plant~~ must be brought to a MODE in which the LCO does not apply. To EDIT. EDIT. 5

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

RCS temperature must be reduced to less than or equal to 350°F

achieve this status, the ~~plant~~ Unit must be brought to at least MODE 3 within 6 hours and ~~at least MODE 4~~ within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required ~~plant~~ Unit conditions from full power conditions in an orderly manner and without challenging ~~plant~~ Unit systems.

edit (5)

edit

edit (25)

3.5.2-01 <INSERT B3.5-16A>

SURVEILLANCE REQUIREMENTS

~~SR 3.5.2.1~~

~~Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in position by removal of power or by key locking the control in the correct position ensures that the valves cannot change position as the result of an active failure. These valves are of the type described in Reference 4, which can disable the function of both ECCS trains and invalidate the accident analyses. The 12 hour Frequency is considered reasonable in view of other administrative controls that will ensure the unlikelihood of a mispositioned valve.~~

(8)

SR 3.5.2.~~2~~ 1

edit

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an inoperable valve position would only affect a single train. This Frequency

(continued)

<INSERT B3.5-16A>

3.5.2-01

C.1

Condition A is applicable with one or more trains inoperable. The allowed Completion Time is based on the assumption that at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available. With less than 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the unit is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.~~2~~ 1 (continued) edit

has been shown to be acceptable through operating experience.

~~SR 3.5.2.3~~

~~With the exception of systems in operation, the ECCS pumps are normally in a standby, nonoperating mode. As such, the flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent water hammer, pump cavitation, and pumping of noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an ESFAS signal or during shutdown cooling. The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the existence of procedural controls governing system operation.~~ 8

SR 3.5.2.~~4~~ 2 edit

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the ASME Code (Ref. 7). ~~This type of testing may be accomplished by measuring the pump's developed head at only one point of the pump's characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant accident analysis.~~ <<INSERT B3.5-17A>> SRs are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code. ~~Section XI of the ASME Code provides the activities and Frequencies necessary to satisfy the requirements.~~ edit 17

SR 3.5.2.~~5~~ 3 ~~and SR 3.5.2.6~~ edit

This ~~These~~ SR~~s~~ demonstrate~~,~~s that each automatic ECCS valve actuates to the required position on an actual or simulated ESFAS signal. ~~and that each ECCS pump starts on receipt of an~~ 6

(continued)

<INSERT B3.5-17A>

This testing confirms component OPERABILITY, trends performance, and detects incipient failures by indicating abnormal performance.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.5 3 ~~and SR 3.5.2.6~~ (continued) edit 6

~~actual or simulated ESFAS signal.~~ This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a ~~plant~~ unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of the ESFAS testing, and equipment performance is monitored as part of the Inservice Testing Program. edit edit

<INSERT B 3.5-18A>> 6

~~SR 3.5.2.7~~

~~This Surveillance ensures that these valves are in the proper position to prevent the HPI pump from exceeding its runout limit. This 18 month Frequency is based on the same reasons as those stated for SR 3.5.2.5 and SR 3.5.2.6.~~

~~SR 3.5.2.8~~

~~This Surveillance ensures that the flow controllers for the LPI throttle valves will automatically control the LPI train flow rate in the desired range and prevent LPI pump runout as RCS pressure decreases after a LOCA. The 18 month Frequency is justified by the same reasons as those stated for SR 3.5.2.5 and SR 3.5.2.6.~~ 8

SR 3.5.2.9 5

Periodic inspections of the ~~containment~~ reactor building sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The 18 month Frequency is based on the need to perform this Surveillance ~~under the conditions that apply~~ during a ~~plant~~ unit outage, ~~on the need to preserve access to the location, and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power.~~ Operating experience has shown ~~This Frequency has been found~~ to be ~~sufficient~~ acceptable to detect abnormal degradation. ~~and has been confirmed by operating experience.~~ edit edit edit edit edit edit

(continued)

<INSERT B3.5-18A>

SR 3.5.2.4

The intent of this SR is to verify that the ECCS pumps are capable of automatically starting on an ESAS signal. Because of the system design configuration and the limitations imposed on pump operation during the unit conditions when this test would be conducted, this verification must be conducted through a series of sequential, overlapping or total steps in order to demonstrate functionality. SR 3.5.2.4 demonstrates that each ECCS pump would be capable of starting by verifying that its breaker closes on receipt of an actual or simulated ESAS signal. SR 3.5.2.4 works in conjunction with the Inservice Testing Program (SR 3.5.2.2) which periodically verifies the ability of the pumps to start and operate within limits, and the ESAS actuation logic testing which periodically verifies the ability of the ESAS to sense, process and generate an actuation signal.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

BASES (continued)

REFERENCES

1. SAR, Section 6. edit

3. ~~2.~~ 10 CFR 50.46. edit

4. ~~3.~~ ~~F~~SAR, Section ~~6.3~~. 14.2.2.5.2 edit

6. ~~3.~~ NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975. edit

~~4. IE Information Notice 87-01, "RHR Valve Misalignment Causes Degradation of ECCS in PWRs," January 6, 1987.~~ 8

7. ~~5.~~ ASME, Boiler and Pressure Vessel Code, Section XI, Inservice Inspection~~, Article IWV-3400~~. edit

5. 10 CFR 50.36. 20

2. Letter from A.C. Thadani (NRC) to P.S. Walsh (BWOG) dated March 9, 1993. 19

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS—Shutdown

BASES

BACKGROUND

<INSERT B3.5-20A>

~~The Background section for Bases B 3.5.2, "ECCS—Operating," is applicable to these Bases, with the following modifications.~~

MODE 3 with RCS temperature ≤ 350°F and in

In MODE 4, the required ~~ECCS~~ trains consists of two separate subsystems: high pressure injection (HPI) and low pressure injection (LPI), ~~each consisting of two~~ redundant, 100% capacity trains. move (12)

LPI

instruments, controls,

The ~~ECCS~~ flow paths consist of piping, valves, heat exchangers, and pumps, such that water from the borated water storage tank (BWST) can be injected into the Reactor Coolant System (RCS), following the accidents described in Bases 3.5.2. capable of taking suction to Reactor vessel. and the capability to manually (locally or remotely) transfer suction to the reactor building sump edit edit edit edit

APPLICABLE SAFETY ANALYSES

~~The Applicable Safety Analyses section of Bases 3.5.2 is applicable to these Bases.~~ edit

MODE 3 with RCS temperature ≤ 350 °F and in

~~Due to~~ the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ~~ECCS~~ operational requirements ~~are~~ allow to be reduced. ~~Included in these reductions is that certain automatic Engineered Safety Feature Actuation System (ESFAS) actuation is not available. In this MODE sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.~~ (24)

~~Only one ECCS train is required for MODE 4. This requirement dictates that single failures are not considered during this MODE.~~ edit (20)

<Insert B3.5-20B>

LCO

MODE 3 with RCS temperature ≤ 350°F and in

In MODE 4, ~~one of the~~ two independent and redundant ~~ECCS~~ LPI trains ~~is~~ are required to ensure sufficient ~~ECCS~~ LPI flow is available to the core following a DBA. ~~In MODE 4, an ECCS train consists of an HPI subsystem and an LPI subsystem. Each~~ LPI train includes the piping, instruments, pump, heat exchanger, valves, (12) edit edit

(continued)

<INSERT B3.5-20A>

The ANO-1 SAR states that the High Pressure Injection (HPI), Low Pressure Injection (LPI) and Core Flooding Systems are collectively designed as an Emergency Core Cooling System (ECCS) (Ref. 1). In this Technical Specification, the term ECCS refers to the components associated with the LPI system. The HPI system, in conjunction with the LPI system, is covered by LCO 3.5.2, "ECCS - Operating." The core flood tanks are addressed by LCO 3.5.1, "Core Flood Tanks (CFTs)."

<INSERT B3.5-20B>

In MODE 3 with RCS temperature ≤ 350°F and in MODE 4, the ECCS-Shutdown LCO satisfies Criterion 4 of 10CFR50.36.

BASES

LCO (continued)

and controls to ensure an OPERABLE flow path capable of taking suction from the BWST and transfer~~red~~ suction to the ~~containment~~ sump.

reactor building

edit

During an event requiring ~~ECCS actuation~~, a flow path is required to provide ~~an abundant supply of~~ water from the BWST ~~to the RCS~~ via the ~~ECCS~~ pumps and their respective supply headers, to ~~each of the four cold leg injection nozzles~~. In the long term, this flow path may be switched to take its supply from the ~~containment~~ sump ~~and to supply its flow to the RCS hot and cold legs~~.

LPI

the capability to manually (locally or remotely)

LPI

the reactor vessel.

reactor building

edit

edit

edit

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edit

~~This LCO is modified by a Note that states that HPI actuation may be deactivated in accordance with LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System." Operator action is then required to initiate HPI. In the event of a loss of coolant accident (LOCA) requiring HPI actuation, the time required for operator action has been shown by analysis to be acceptable.~~

A valve that is locked, sealed, or otherwise secured in its ES position is OPERABLE

<INSERT B3.5-21A>

12

APPLICABILITY

In MODES 1, 2, and ~~3~~, the OPERABILITY requirements for the ECCS are covered by LCO 3.5.2.

and

MODE 3 with RCS temperature > 350 °F

5

In MODE 4 ~~with the RCS temperature below 280°F, one OPERABLE ECCS train is acceptable without single failure consideration~~, on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

MODE 3 with RCS temperature ≤ 350°F and in

two OPERABLE LPI trains are acceptable

12

In MODES 5 and 6, ~~plant~~ conditions are such that the probability of an event requiring ~~ECCS~~ injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "~~DHR~~ and Coolant Circulation—High Water Level," and LCO 3.9.5, "~~DHR~~ and Coolant Circulation—Low Water Level."

unit

LPI

edit

edit

Decay Heat Removal (DHR)

ACTIONS

C ~~A~~.1

12

If no LPI ~~subsystem~~ train is OPERABLE, the unit is not prepared to respond to ~~a LOCA~~ ~~or~~ to continue cooldown using

edit

move

an event requiring low pressure injection

and may not be prepared

(continued)

<INSERT B3.5-21A>

This LCO is modified by a Note that allows a Decay Heat Removal (DHR) train to be considered OPERABLE during alignment, when aligned, or when operating for decay heat removal, if it is capable of being manually (locally or remotely) realigned to the LPI mode of operation and is not otherwise inoperable. This provision is necessary because of the dual requirements of the components that comprise the low pressure injection/decay heat removal system.

BASES

ACTIONS

C ~~A~~.1 (continued)

the LPI pumps and ~~decay~~ DHR heat exchangers. The Completion Time of immediately, which would initiate action to restore at least one ~~ECCS~~ LPI train ~~subsystem~~ to OPERABLE status, ensures that prompt action is taken to restore the required LPI ~~cooling~~ capacity. Normally, in MODE 4, reactor decay heat must be removed by an LPI train operating with suction from the RCS. If no LPI train is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generator(s). The alternate means of heat removal must continue until one of the inoperable ~~ECCS~~ LPI trains ~~subsystem~~ can be restored to operation so that continuation of decay heat removal (DHR) is provided.

With both DHR pumps and heat exchangers inoperable, it would be unwise to require the unit ~~plant~~ to go to MODE 5, where the only available heat removal system is the LPI trains operating in the DHR mode. Therefore, the appropriate action is to initiate measures to restore one ECCS LPI train ~~subsystem~~ and to continue the actions until the ~~subsystem~~ is restored to OPERABLE status.

move

<<INSERT 3.5-22A>>

A ~~B~~.1

If one LPI train ~~no ECCS HPI subsystem~~ is inoperable ~~OPERABLE~~, ~~due to the inoperability of the HPI pump or flow path from the BWST~~, the unit ~~plant~~ is not prepared to provide redundant, single failure proof LPI in ~~high pressure~~ response to ~~Design Basis~~ Events requiring ES~~F~~AS. The 48 ~~1~~ hour Completion Time to restore the LPI ~~at least one ECCS HPI~~ subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the unit ~~plant~~ in MODE 5, where an LPI ~~ECCS~~ train is not required.

B ~~C~~.1

When the Required Action and associated Completion Time of Condition B A are not met ~~cannot be completed within the required Completion Time~~, a controlled cooldown ~~shutdown~~ should be initiated. The allowed Completion Time of 24 hours is reasonable, based on operating experience, to reach MODE 5 from MODE 3 with RCS temperature ≤ 350°F ~~full power conditions~~ in an orderly manner and without challenging unit ~~plant~~ systems.

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(continued)

<INSERT B3.5-22A>

C.2

Required Action C.2 requires that the unit be placed in MODE 5 within 24 hours. This Required Action is modified by a Note that states that this Required Action is only required to be performed if one DHR train is OPERABLE. This Required Action provides for those circumstances where the LPI trains may be inoperable but are otherwise capable of providing the necessary decay heat removal. Under this circumstance, the prudent action is to remove the unit from the Applicability of the LCO and place the unit in a stable condition in MODE 5. The Completion Time of 24 hours is reasonable, based on operating experience, to reach MODE 5 in an orderly manner and without challenging unit systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

B

The applicable Surveillance descriptions from Bases 3.5.2 apply. ~~This SR is modified by a Note that allows an DHR~~ LPI ~~train to be considered OPERABLE during alignment and operation for DHR, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the DHR mode during MODE 4, if necessary.~~ LPI edit

12

REFERENCES

The applicable references from Bases 3.5.2 apply. edit

B

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Borated Water Storage Tank (BWST)

BASES

BACKGROUND

The BWST supports the ECCS and the ~~containment~~ Reactor Building Spray System by providing a source of borated water for ECCS and ~~containment~~ reactor building spray pump operation. In addition, the BWST supplies borated water to the refueling ~~pool~~ canal for refueling operations. edit edit edit

The BWST supplies two ECCS trains, each by a separate, redundant supply header. Each header also supplies one train of the ~~containment~~ Reactor Building Spray System. A ~~normally open,~~ motor operated isolation valve is provided in each header to allow the operator to isolate the BWST from the ECCS after the ECCS pump suction has been transferred to the ~~containment~~ reactor building sump following depletion of the BWST during a loss of coolant accident (LOCA). Use of a single BWST to supply both ECCS trains is acceptable because the BWST is a passive component, and passive failures are not assumed ~~in~~ ~~the analysis of Design Basis Events (DBEs)~~ to occur coincidentally with the Design Basis Accident (DBA). edit edit edit edit

~~The ECCS and containment spray pumps are provided with recirculation lines that ensure each pump can maintain minimum flow requirements when operating at shutoff head conditions.~~ 14

This LCO ensures that:

a. The BWST contains sufficient borated water to support the ECCS during the injection phase;

b. Sufficient water volume exists in the ~~containment~~ reactor building sump to support continued operation of the ECCS and ~~containment~~ reactor building spray pumps at the time of transfer to the recirculation mode of cooling; and edit edit

c. The reactor remains ~~subcritical~~ adequately shutdown following a LOCA. 22

Insufficient water inventory in the BWST could affect NPSH and result in insufficient cooling ~~capacity of~~ Capability by the ECCS when the transfer to the recirculation mode occurs. edit

(continued)

BASES

BACKGROUND (continued)

Improper boron concentrations could result in a reduction of ~~SDM or~~ excessive boric acid ~~precipitation~~ in the core following a LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside ~~containment~~.

adequate an concentration edit the reactor building

APPLICABLE SAFETY ANALYSES

During accident conditions, the BWST provides a source of borated water to the high pressure injection (HPI), low pressure injection (LPI), and ~~containment~~ spray pumps. As such, it provides ~~containment~~ cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown. The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of ~~Specifications~~ B 3.5.2, "ECCS—Operating," and ~~B 3.6.6, "Containment~~ Spray and Cooling Systems." ~~These analyses are used to assess changes to the BWST in order to evaluate their effects in relation to the acceptance limits.~~

reactor building edit edit Bases B 3.6.5, "Reactor Building edit edit

The limits on ~~volume of [≥ 415,200 gallons and ≤ 449,000 gallons]~~ are ~~based on several factors~~. ~~Sufficient deliverable volume must be available to provide at least 20 minutes of full flow of all ECCS pumps prior to the transfer to the containment sump for recirculation. Twenty minutes gives~~ the operator adequate time to prepare for switchover to ~~containment~~ sump recirculation.

level of ≥ 38.4 feet and ≤ 42 feet
<INSERT 3.5-25A>
the accident analysis assumed values. edit edit edit edit edit

A second factor that affects the minimum required BWST volume is the ability to support continued ECCS pump operation after the manual transfer to recirculation occurs. When ECCS pump suction is transferred to the sump, there must be sufficient water in the sump to ensure adequate net positive suction head (NPSH) for the LPI and ~~containment~~ spray pumps. This NPSH calculation is described in the ~~F~~SAR (Ref. 1), and the amount of water that enters the sump from the BWST and other sources is one of the input assumptions. ~~Since the BWST is the main source that contributes to the amount of water in the sump following a LOCA, the~~ The calculation does not take credit for more than the minimum volume ~~of usable water~~ from the BWST.

reactor building edit edit edit assumed edit edit

The third factor is that the volume of water in the BWST must be within a range that will ensure the solution in the

(continued)

<INSERT B3.5-25A>

These levels correspond to a volume of approximately 375,096 gallons and 405,090 gallons, respectively. These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

BASES

APPLICABLE SAFETY ANALYSES (continued)

sump following a LOCA is within a specified pH range that will minimize the evolution of iodine and the effect of chloride and caustic stress corrosion cracking on the mechanical systems and components. edit

<INSERT B 3.5-26A> ~~The volume range ensures that refueling requirements are met and that the capacity of the BWST is not exceeded. Note that the volume limits refer to total, rather than usable, volume required to be in the BWST;~~ Cap level the safety analysis assumed level a certain amount of water is unusable because of tank discharge line location or other physical characteristics, and the time assumed and for the operator to accomplish swapover to the sump. edit edit edit edit edit

The ~~[~~2270~~]~~ ppm limit for ~~minimum boron concentration was~~ established to ensure that, following a LOCA, with a minimum BWST level, the reactor will remain subcritical in the cold condition following mixing of the BWST and Reactor Coolant System (RCS) water volumes. ~~Large break LOCAs assume that all control rods remain withdrawn from the core.~~ 23

The minimum and maximum concentration limits both ensure that the long term solution in the sump following a LOCA is within a specified pH range that will minimize the evolution of iodine and the effect of chloride and caustic stress corrosion cracking on the mechanical systems and components. edit

The 2670 ~~[2450]~~ ppm maximum limit for boron concentration in the BWST is also based on the potential for boron precipitation in the core during the long term cooling period following a LOCA. For a cold leg break, the core dissipates heat by pool nucleate boiling. Because of this boiling phenomenon in the core, the boric acid concentration will increase in this region. If allowed to proceed in this manner, a point may be reached where boron precipitation will occur in the core. <INSERT B3.5-26B> ~~Post~~ LOCA emergency procedures direct the operator to establish dilution flow paths in the LPI System to prevent this condition by establishing a forced flow path through the core regardless of break location. These procedures are based on the minimum time in which precipitation could occur, assuming that maximum boron concentrations exist in the borated water sources used for injection following a LOCA. edit 19

~~Boron concentrations in the BWST in excess of the limit could result in precipitation earlier than assumed in the analysis.~~

(continued)

<INSERT B3.5-26A>

The fourth factor is that the volume of water in the BWST must be limited to ensure that the resulting post-LOCA maximum reactor building water level is less than that used for environmental qualification of safety related components in the reactor building.

<INSERT B3.5-26B>

B&W has evaluated the flowpath inherent in the reactor vessel internals, commonly called the leakage gap flowpath, and demonstrated that the flowpath would be sufficient by itself to preclude boron precipitation (Ref. 2). As a secondary measure,

BASES

APPLICABLE SAFETY ANALYSES (continued)

<Insert B3.5-27A>

The 40°F lower limit on the temperature of the solution in the BWST was established to ensure that the solution will not freeze. ~~This temperature also helps prevent boron precipitation and ensures that water injection in the reactor vessel will not be colder than the lowest temperature assumed in reactor vessel stress analysis.~~ The 110 ~~[200]~~°F upper limit on the temperature of the BWST contents is consistent with the maximum ~~injection~~ water temperature assumed in the ~~LOCA~~ safety analysis. edit 13

~~The numerical values of the parameters stated in the SR are actual values and do not include allowance for instrument errors.~~ edit

In MODES 2, 3 and 4, the BWST satisfies Criterion 4 of 10CFR50.36.

In MODE 1, the BWST satisfies Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref 3). 20

LCO

The BWST exists to ensure that an adequate supply of borated water is available to cool and depressurize the ~~containment~~ reactor building in the event of a DBA; to cool and cover the core in the event of a LOCA, thereby ensuring the reactor remains ~~subcritical~~ adequately shutdown following a DBA; and to ensure an adequate level exists in the ~~containment~~ reactor building sump to support ECCS and ~~containment~~ reactor building spray pump operation in the recirculation ~~MODE~~ mode. To be considered OPERABLE, the BWST must meet the limits for water ~~volume~~ level, boron concentration, and temperature established in the SRs. edit 22 edit edit

APPLICABILITY

In MODES 1, 2, 3, and 4, the BWST OPERABILITY requirements are dictated by the ECCS and ~~Containment~~ Reactor Building Spray System OPERABILITY requirements. Since both the ECCS and ~~Containment~~ Reactor Building Spray System must be OPERABLE in MODES 1, 2, 3, and 4, the BWST must be OPERABLE to support their operation. edit edit

Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled," respectively. MODE 6 core cooling requirements are addressed by LCO 3.9.4, "~~DHR~~ Decay Heat Removal (DHR) and Coolant Circulation—High Water Level," and LCO 3.9.5, "~~DHR~~ DHR and Coolant Circulation—Low Water Level." edit edit

(continued)

<INSERT B3.5-27A>

These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

BASES (continued)

ACTIONS

A.1

With either the BWST boron concentration or borated water temperature not within limits, the condition must be corrected within 8 hours. In this condition, neither the ECCS nor the Reactor Building Spray System ~~can~~ may be able to perform its design functions. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the ~~plant~~ unit in a MODE in which these systems are not required. The 8 hour limit to restore the temperature or boron concentration to within limits was developed considering the time required to change boron concentration or temperature and assuming that the contents of the tank are still available for injection.

edit
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B.1

With the BWST inoperable for reasons other than Condition A (e.g., water volume), ~~levels~~ the BWST must be restored to ~~within required limits~~ OPERABLE status within 1 hour. In this condition, neither the ECCS nor the ~~Containment~~ Reactor Building Spray System can perform its design functions. Therefore, prompt action must be taken to restore the ~~tank~~ BWST to OPERABLE status or to place the ~~plant~~ unit in a MODE in which the BWST is not required. The allowed Completion Time of 1 hour to restore the BWST to OPERABLE status is based on this condition simultaneously affecting multiple redundant trains.

edit
edit

C.1 and C.2

If the Required Actions and ~~BWST cannot be restored to OPERABLE status within the~~ associated Completion Time~~s~~ are not met, the ~~plant~~ unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the ~~plant~~ unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required ~~plant~~ unit conditions from full power conditions in an orderly manner and without challenging ~~plant~~ unit systems.

edit
edit
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(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

Verification every 24 hours that the BWST water temperature is within the specified temperature band ensures that the ~~boron will not precipitate, the~~ fluid will not freeze~~; the fluid temperature entering the reactor vessel will not be colder than assumed in the reactor vessel stress analysis;~~ and that the fluid temperature ~~entering the reactor vessel~~ will not be hotter than assumed in the ~~LOCA~~ safety analysis. The 24 hour Frequency is sufficient to identify a temperature change that would approach either temperature limit~~, and has been shown to be acceptable through operating experience~~.

<INSERT B 3.5-29A>

The SR is modified by a Note that requires the Surveillance to be performed only when ambient air temperatures are outside the operating temperature limits of the BWST. With ambient temperatures within this band, the BWST temperature should not exceed the limits.

edit

(13)

SR 3.5.4.2

Verification every 7 days that the BWST contained ~~volume~~ level is ~~within the required range~~ ≥ 38.4 feet and ≤ 42 feet ensures that a sufficient initial supply is available for injection and to support continued ECCS pump operation on recirculation. <INSERT B3.5-29B> Since the BWST ~~volume~~ level is normally stable~~, and provided with a low level alarm,~~ a 7 day Frequency has been shown to be appropriate through operating experience.

edit

edit

edit

<INSERT B3.5-29C>

SR 3.5.4.3

Verification every 7 days that the boron concentration of the BWST fluid is ~~within the required band~~ ≥ 2270 ppm and ≤ 2670 ppm ensures that the reactor will remain ~~subcritical~~ adequately shutdown following a LOCA. Since the BWST ~~volume~~ level is normally stable, a 7 day sampling Frequency is appropriate and has been shown to be acceptable through operating experience.

edit

REFERENCES

1. ~~F~~SAR, Section ~~6.1~~.

edit

2. Letter from A.C. Thadani (NRC) to P.S. Walsh (BWOG) dated March 9, 1993.

(19)

3. 10 CFR 50.36.

(20)

<INSERT B3.5-29A>

These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

<INSERT B3.5-29B>

These levels correspond to a volume of approximately 375,096 gallons and 405,090 gallons, respectively. These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.

<INSERT B3.5-29C>

These parameter values do not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are included in the implementing procedures.