ITS DISCUSSION OF DIFFERENCES

ITS Section 3.7: Plant Systems

1 NUREG-3.7.1 - The main steam safety valves (MSSV) Specification is reformatted to omit the table of specific lift setpoints and to replace the figure for determining the allowable power level and trip settings with predetermined values. The specific lift setpoints are currently required to be tested by current Technical Specification (CTS) Table 4.1-2, item 4. However, the CTS does not contain the specific setpoints. These setpoints are currently identified in the Inservice Testing (IST) Program and are adequately controlled therein under the design change and procedural control programs which include evaluations of changes in accordance with 10 CFR 50.59. Control of these setpoints is proposed to be retained in these programs. A minor editorial change is proposed to clarify that the 1% tolerance is only applicable to the "as-left" settings. The NUREG figure for determining the allowable nuclear overpower-high trip setpoint is provided for units which have MSSVs with different relief capacities. Since it would not be possible to predetermine which valves would be inoperable for the condition, a figure is not provided to calculate the required trip setpoint. However, all MSSVs at ANO-1 are of the same relieving capacity. Therefore, the allowable setpoint for the trip function can be predetermined based on the minimum number of OPERABLE valves per steam generator. This evaluation has been done and provided in a new Table 3.7.1-1, rather than by a figure, for the operators convenience. Also, the wording of Required Action A.1 is revised since the terminology of "reduced power requirement" from the figure is not used in the new Table. The proposed wording is consistent with the wording of Required Action A.2.

The LCO is revised to require that 14 MSSVs (7 on each main steam line) be OPERABLE regardless of power level. This means that Condition A merely allows continued operation rather than restoring compliance with the LCO. The NUREG-1430 Required Action A.1 restores compliance with the LCO and negates the requirement to change the setpoint in Required Action A.2 and control the setpoint during continued operation. This LCO change ensures that continued unit operation with an inoperable MSSV is in accordance with a Required Action.

3.7-01

A Note is added to the LCO to retain the hydrotesting exception provided by CTS 3.4.1.2 Note *. This provides the capability to perform the hydrotesting using steam in lieu of water which would require additional supports due to the added weight. This exception is discussed ANO-1 license Amendment No. 90 (1CNA128405) and its associated request submittal (1CAN108401).

3.7-22

The Bases associated with NUREG 3.7.1 have been revised to incorporate the changes discussed above. The NUREG 3.7.1 Bases Applicability discussion also states that two MSSVs are required when below 18% RTP and when above 18% RTP, the number of MSSVs required to be Operable must be within the acceptable region of Figure 3.7.1-1. With the proposed changes to NUREG 3.7.1, this statement is no longer required to be included in the Applicability discussion, as the proposed Table, Table 3.7.1-1, provides the number of MSSV required to be Operable at all times in MODE 1. Therefore, this information is deleted.

2 Incorporated TSTF-235, Rev. 1.

3.7-02 3 NUREG 3.7.2 - The Applicability of this LCO is revised to MODES 1, 2, and 3, consistent with CTS 3.4.1. The MSIVs, MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves perform an accident mitigation function when there is significant mass and energy in the secondary system. In MODES 4, 5, and 6, the secondary side energy is low and these valves are not required to provide isolation. This change is consistent with current license basis.

3.7-02 4 NUREG 3.7.2 – CTS 3.4.2 allowed action times for inoperable MSIVs are retained in the proposed ITS ACTIONS. Condition A entry conditions are expanded to include one or more MSIVs in both MODES 1 and 2. The unit design includes only two MSIVs; therefore, closure of an MSIV is not practical in either of these MODES. CTS 3.4.2 allows continued operation for 24 hours with either one or two inoperable MSIVs with no action required. This is retained in proposed Required Action A.1 for inoperable MSIVs. This proposed Completion Time allows time to prepare and implement activities necessary for restoration of OPERABILITY if the cause of the inoperability is restorable without a shutdown. Additionally, for MSIVs inoperable in MODE 3, the proposed Required Action C.1 is consistent with the CTS 3.4.2 Completion Time of 48 hours. Finally, the CTS 3.4.2 Completion Time of 24 hours to exit the MODE of Applicability is retained in Required Action D.1. Although the main steam system is not credited as a closed system, under normal conditions it does not provide a direct path from the reactor building atmosphere to the environment. Therefore, these Completion Times are reasonable, and provide for diagnosis and repair of many MSIV problems, thereby avoiding unnecessary shutdown. This change is consistent with current license basis.

5 Not used.

3.7-26

6 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.

The 10 CFR 50.36 Criterion satisfied by the ITS LCOs was modified to preserve consistency with the ANO-1 license basis. The NUREG Criterion specified were modified to be consistent with the analysis assumptions regarding equipment availability and operating condition (i.e., MODE).

7 NUREG 3.7.2 - Incorporates TSTF-209, Rev 1.

8 NUREG 3.7.2 & 3.7.3 – Incorporated TSTF-289.

The specific required closure (isolation) time for the MSIVs and MFIVs is not incorporated. These values are not included in the CTS, have been adequately controlled in the Inservice Test Program, and are proposed to continue to be administratively controlled. This change is consistent with current license basis.

In accordance with unit design and operation, automatic closure capability is bypassed at ≤ 750 psig in the secondary system to avoid unintentional closure during normal shutdowns. Therefore, a Note is included in the automatic actuation surveillance (SR 3.7.2.2 and SR 3.7.3.2) to indicate that automatic isolation capability is not required when the secondary system pressure is ≤ 750 psig which is consistent with CTS 3.5.1.16. This change preserves current license basis requirements and accommodates unit specific design characteristics.

3.7-02 9 NUREG 3.7.3 and Bases – ITS 3.7.3 and Bases incorporate ANO-1 specific terminology. The ANO-1 main feedwater isolation is accomplished by either an MFIV, or the associated Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve combination. MFIV refers to a specific component, and is not used to refer to the Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve combination. This change is consistent with current license basis, as discussed in SAR Section 14.2.2.1. The NUREG 3.7.3 Bases markup only shows the insertion of the ANO-1 terminology once at the top of each page due to the large number of edits that would be required. This is done to provide a clear markup of the Bases. All occurrences of "MFIVs" and "MFCVs, or associated SFCVs" will be replaced references to the "MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves," as appropriate. In addition, a new Condition C as been added to ensure that all valve combinations are adequately addressed in the ITS. The existing NUREG 3.7.3 Conditions C, D and E have been re-lettered accordingly.

3.7-14 10 NUREG 3.7.14 and Bases - "Fuel Storage Pool" has been revised to "Spent Fuel Pool" at each occurrence for consistency with the ANO-1 license basis. Although several terms are used in the ANO-1 SAR to refer to this component, "Spent Fuel Pool" is the most prevalent. This change is considered to be administrative in nature and is consistent with the current license basis.

11 NUREG 3.7.4 – SAR Section 14 discusses the use of the ADVs in one accident, the
3.7-03 Loss of All Unit AC Power.

Although the station blackout (SBO) event is beyond the ANO-1 design basis, certain aspects of ADV operation were discussed in ANO's resolution of this issue. The air operated atmospheric dump valves limit challenges to the MSSVs during a SBO event.

The emergency operating procedures (EOPs) instruct the operators to establish pressure control using the ADVs from within the control room. If control power or instrument air is not available, the valves can be manually operated locally.

In the Supplemental Safety Evaluation for the Arkansas Nuclear One Units 1 and 2 Station Blackout Rule (0CNA109111), the NRC staff concluded "that following an SBO event and upon the loss of compressed air, the licensee will be able to manually operate the ADVs for decay heat removal. Therefore, the staff considers this issue related to compressed air resolved."

The ADVs and ADV block valves perform no active safety-related function. The ADVs are normally closed and designed to fail closed. Use of the ADVs has never been credited in the ANO-1 accident analyses for performance of any safety-related function. Use of the ADVs for pressure relief has been mentioned in accident analyses (i.e., complete loss of all unit AC power), but has not been required since the MSSVs are safety-related and always available for pressure relief. It should be noted that ANO-1 is in a safe shutdown condition when it is in "hot shutdown." Therefore, it is acceptable to maintain the plant in "hot shutdown" and to rely on the MSSVs for pressure control. Post-accident cool-down to below 525° F (which would require use of an ADV) is not required. The ANO-1 safety analysis does not credit the atmospheric dump valves (ADVs) for events which meet the criteria of 10 CFR 50.36. Also, the CTS does not contain any requirements for the ADVs. Therefore, controls for these valves are proposed to continue to be administrative and not incorporated in the Technical Specifications. This change is consistent with current license basis.

12 NUREG 3.7.5 - The safety related emergency feedwater (EFW) system contains only two pumps and associated flow paths. All NUREG references to a third train or pump have been deleted. This change is consistent with current license basis.

13 NUREG 3.7.5 - Incorporates TSTF-101.

14 NUREG 3.7.5 and Bases – Note 1 is omitted for SR 3.7.5.3 and SR 3.7.5.4. This testing is currently performed at low pressures to avoid either: a) making the system inoperable by tagging out the injection valves which would also open on the actuation signal, or b) injecting cold condensate into the steam generators. Valve and pump actuation can be demonstrated at low pressures, and along with full pressure, manual opening of the steam admission valves and pump flow testing, adequately demonstrates the capability of the system to perform these required safety functions. This change is consistent with current license basis.

ANO-289 Note 2 has been revised to incorporate TSTF-284, Rev 3.

Additionally, the wording of Note 2 for SR 3.7.5.3 and SR 3.7.5.4 revised for clarity and consistency with the Applicability. The "applicable" MODES are addressed only in the portion of the Specification entitled "APPLICABILITY" (with the exception of where applicable SRs of one specification are referenced by another specification, e.g., when a shutdown specification identifies the "applicable" SRs from the operating specification rather than repeat each "required" SR). Thus, Note 2 has been modified to clearly correlate with the Applicability. These changes are consistent with the NUREG Writer's Guide, and current license basis (CTS 3.4.3).

15 NUREG 3.7.5.5 – CTS 4.8.1.c requires this surveillance be performed only on manual valves. This is acceptable because it verifies the position of those valves that would not be easily detected through installed instrumentation and indication available to the operator or through the performance of a pump surveillance. In addition, this SR effectively replicates the requirements of SR 3.7.5.1 which must be performed prior to entry into the MODE of Applicability for this Specification. This change is consistent with current license basis.

In addition, the unit specific designation for the "Q" condensate storage tank (QCST) was provided to clarify which condensate storage tank is the subject of this SR (reference CTS 4.8.1.c). This change is consistent with current license basis.

16 NUREG 3.7.5 - The unit design does not include EFW pump suction pressure interlocks. Therefore, SR 3.7.5.6 and SR 3.7.5.7 are not incorporated. This change is consistent with current license basis.

17 NUREG 3.7.7 - The ANO-1 safety analysis does not credit the intermediate cooling water system for events which meet the criteria of 10 CFR 50.36. The safety related cooling water requirements are met by the service water system (see SAR Section 9.3). Therefore, only the service water system is proposed to be incorporated in the Technical Specifications. This change is consistent with current license basis.

3.7-08

18 NUREG 3.7.8.3 - The service water system is equipped with three pumps, only two of which are required to be in service at any given time (SAR Section 9.3.2.1). The 'A' and 'C' service water pumps normally supply their respective service water loops. The 'B' service water pump is a 'swing' pump that can be aligned to supply either service water loop in the event either the 'A' or 'C' pump is inoperable. The 'A' and 'C' pumps do not receive an engineered safety (ES) actuation signal. Instead, if these pumps are the pumps in service at the time an ES signal is initiated, they will remain in service. If offsite power is lost, these pumps will autostart following restoration of voltage to the ES buses. In the event one of these two pumps fails to start following restoration of ES bus voltage with an ES signal present, the 'B' pump is automatically started approximately 5 seconds later and is realigned to the appropriate service water loop. SR 3.7.8.3 has been modified as SR 3.7.7.3 to require testing of the 'required' service water pumps. This change allows one service water pump to be taken out of service without affecting the OPERABILITY of the service water system since the two remaining service water pumps are single-failure proof. This change is consistent with current license basis as specified in CTS 3.3.1.C.

19 NUREG 3.7.9 - The ANO-1 ultimate heat sink does not utilize cooling towers, nor cooling tower fans. Therefore, the ACTIONS related to fans and SR 3.7.9.3 are not applicable. This change is consistent with current license basis.

ITS DISCUSSION OF DIFFERENCES

20 NUREG 3.7.9 - SR 3.7.9.1, SR 3.7.9.2, and SR 3.7.9.3 and associated Bases are revised to verify the appropriate parameters for an emergency cooling pond consistent with CTS 4.13. ITS SR 3.7.8.3 will verify the pond contains the necessary volume when the water level is ≥ 5 ft, and ITS SR 3.7.8.1 will verify the pond level is ≥ 5 ft on a more frequent basis. The Frequency for ITS SR 3.7.8.3 is every 12 months since the degradation of the pond is gradual. ITS SR 3.7.8.2 is limited to only require the temperature verifications during the summer months when there is sufficient potential to exceed the limits to warrant the surveillance.

3.7-11 NUREG 3.7.9 has also been revised to retain the requirements of CTS 4.13.1.4 by the addition of SR 3.7.8.4 and associated Bases. This SR requires a visual inspection of the ECP and spillway to ensure any physical degradation from wave action, or other changes in appearance is within acceptable limits. These changes are consistent with current license basis.

21 NUREG 3.7.1 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 3.7.1 BACKGROUND - Only 8 MSSVs are provided for each SG header.

B 3.7.1 ASA - Revised discussion of applicable transients and accidents in accordance with the current SAR.

B 3.7.1 LCO – Only 7 of the 8 MSSVs on each header are required for mitigation from full power.

B 3.7.1 LCO - The discussions of OPERABILITY are revised to prevent misinterpretation. These paragraphs incorrectly imply that they contain all requirements for OPERABILITY. They are revised to indicate that these are required to attain OPERABILITY but that compliance with these does not necessarily assure OPERABILITY. This is only determined by compliance with the definition of OPERABLE/OPERABILITY.

B 3.7.1 RA B.1 & B2 - The entry condition description is revised to match the Specification requirements.

B 3.7.1 References - A reference to Framatome Document 86-1266156-00, "ANO-1 Overpressure Protection," dated October 31, 1997, has been added to provide a reference for the MSSV relief capacity.

22 NUREG 3.7.2 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 3.7.2 BACKGROUND - Revised discussion of isolation signal to refer to more detailed description of initiating signals.

B 3.7.2 ASA - Revised discussion of Applicable Safety Analyses to be consistent with the unit specific analyses and license basis.

B 3.7.2 LCO - The discussions of OPERABILITY are revised to prevent misinterpretation and to be consistent with the unit specific analyses and license basis. These paragraphs incorrectly imply that they contain all requirements for OPERABILITY. They are revised to indicate that these are required to attain OPERABILITY but that compliance with these does not necessarily assure OPERABILITY. This is only determined by compliance with the definition of OPERABLE/OPERABILITY.

3.7-25

B 3.7.2 RA A.1- The Completion Time discussion has been revised for consistency with the ANO-1 SAR which states that the MSIVs are not considered as reactor building isolation valves (SAR Table 5-1).

B 3.7.2 RA B.1 - The condition description is corrected for consistency with similar statements throughout the ITS Bases and with the wording of the Condition.

B 3.7.2 RA D.1 and D.2 - The condition description is corrected for accuracy.

23 NUREG 3.7.3 Bases - This change provides unit specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures. References to Feedwater Line Break (FWLB) analyses and the excess feedwater event have been deleted from the ITS 3.7.3 Bases as these accidents/events are not part of the ANO-1 safety analyses as provided in SAR Section 14. These changes are consistent with current license basis.

3.7-27

B 3.7.3 BACKGROUND - Revised discussion of Emergency Feedwater Initiation and Control (EFIC) System to refer to more detailed description of initiating signals, and omit non-applicable discussions. Revised discussions of the purpose of MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves in the ITS. to be consistent with the unit specific analyses and license basis.

3.7-02

B 3.7.3 ASA - Revised discussion of Applicable Safety Analyses to be consistent with the unit specific analyses and license basis.

B 3.7.3 LCO - The discussions of OPERABILITY are revised to prevent misinterpretation. These paragraphs incorrectly imply that they contain all requirements for OPERABILITY. They are revised to indicate that these are required to attain OPERABILITY but that compliance with these does not necessarily assure OPERABILITY. This is only determined by compliance with the definition of OPERABLE/OPERABILITY.

B 3.7.3 LCO - Revised discussion to omit non-applicable discussions based on the unit specific analyses and license basis.

B 3.7.3 RA E.1 and E.2 - The condition description is corrected for accuracy.

24 NUREG 3.7.17 Bases - This change incorporates a thyroid dose conversion factor reference to the defined term of DOSE EQUIVALENT I-131 in Section 1.1, Definitions.

25 NUREG 3.7.5 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 3.7.5 General - The EFW system description is revised to reflect unit design and nomenclature.

B 3.7.5 General - Revised discussion of Emergency Feedwater Initiation and Control (EFIC) System to refer to more detailed description of initiating signals, and omit non-applicable discussions.
B 3.7.5 RA C.1 and C.2 - The condition description is corrected for accuracy.
B SR 3.7.5.1 - Clarification is provided for the "correct" position for automatic valves which may reposition upon an actuation signal.

26 NUREG 3.7.6 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 3.7.6 BACKGROUND - The CST description is revised to reflect unit design.
B 3.7.6 ASA & LCO - The CST discussion of the applicable safety analysis is revised to be consistent with the unit specific analyses and license basis.
B 3.7.6 LCO - The discussion is clarified to identify the necessary volume if both units are relying on the "Q" CST, T-41B, and to revise the associated levels based on the latest calculations.
B 3.7.6 APPLICABILITY - The discussion is revised to address all conditions; "MODE with steam generators not being relied upon for heat removal" was missing.
B 3.7.6 RA B.1 & B.2 - The Required Actions do not provide a time for entry into MODE 4. However, the discussion of "an additional 6 hours" implies that MODE 4 must be entered within 12 hours. Since there is no such requirement, this misleading statement is omitted.

27 NUREG 3.7.8 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 3.7.8 BACKGROUND, ASA & LCO - The service water system description is revised to reflect unit design and nomenclature.

28 NUREG 3.7.8 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 3.7.8 BACKGROUND & ASA - The UHS description is revised to reflect unit design.

29 NUREG 3.7.10 - Required Actions C.2.1 and D.2 are omitted since they are not consistent with the Applicability of the Specification. Further, retention would be of no consequence since as soon as the concurrent action of "immediately suspend movement of irradiated fuel assemblies" is complete, the Specification will no longer be applicable and the CORE ALTERATIONS would no longer be controlled by this Specification. Finally, omission of these Required Actions is consistent with the "bracketed" identification of similar Required Actions in NUREG-1430 Specification 3.3.16.

30 NUREG 3.7.15 - Incorporates TSTF-070, Rev. 1.

The word "spent" was added to the revised Required Action A.2.2 to clearly establish that this applies to the spent fuel pool storage area consistent with the wording of

Required Actions A.1 and A.2.1. This editorial change is consistent with the terminology used in the current license basis.

3.7-12 31 NUREG 3.7.10 - NUREG SR 3.7.10.4 is not adopted. Per Standard Review Plan Section 6.4, only control room emergency ventilation system designs with < 0.25 volume changes per hour are required to provide periodic verification of the pressurization capability for the control room. SAR 9.7.2.1 indicates that the ANO-1 CREVS is based on ≥ 3 volume changes per hour. However, this has determined to be the recirculation flow rate, not the pressurization flow rate. The ANO-1 control room emergency ventilation system was designed for a pressurization flow rate of ~ 0.5 volume changes per hour. Therefore, this Surveillance is not adopted. This change is consistent with current license basis.

3.7-12 32 NUREG 3.7.10 - NUREG SR 3.7.10.5 is retained in the ITS as SR 3.7.9.4. See DOD-31 for a discussion of the ANO-1 control room emergency ventilation system pressurization flow rate. Per Standard Review Plan Section 6.4, those control room emergency ventilation system designs with greater than or equal to 0.5 volume changes per hour should be subject to periodic verification (every 18 months) that the makeup is ± 10% of design value. ITS SR 3.7.9.4 specifies an acceptance criteria of ≥ 300 cfm and ≤ 366 cfm (±10% of the makeup air design flow rate of 333 cfm (SAR Section 9.7)). NUREG-1430 does not specify any Bases for NUREG SR 3.7.10.5. Therefore, appropriate Bases have been proposed. In addition, SRP 6.4 was added to the reference section following the SR Bases. The incorporation of this SR is a more restrictive requirement than currently contained in the license basis.

33 NUREG 3.7.12 & 3.7.13 - NUREG SR 3.7.12.5 is not incorporated for the penetration room ventilation system since no such action (opening) of the damper is provided in the system. NUREG SR 3.7.13.5 is not incorporated for the fuel handling area ventilation system since no such dampers are provided in the system. These changes are consistent with current license basis.

34 NUREG 3.7.13 - The Applicability and Required Actions of the requirements for the Fuel Handling Area Ventilation System are revised to include only those requirements associated with the handling of irradiated fuel assemblies in the fuel handling area. This is consistent with CTS 3.15.1 and with the safety analysis assumptions for operation of the filtration system. This change is consistent with current license basis.

Included with this change is an ACTIONS Note to indicate that LCO 3.0.3 is not applicable (consistent with CTS 3.15.2). Since the movement of irradiated fuel could occur in the fuel handling area during operation in MODES 1, 2, 3, or 4, if the applicable Required Actions could not be met, LCO 3.0.3 would require shutdown. However, this is inappropriate since operation of the unit is unrelated to fuel movement in the fuel handling area. This change is consistent with current license basis.

35 NUREG 3.7.13 - The LCO and Actions are revised to require the fuel handling area ventilation system to be in operation when moving irradiated fuel in the fuel handling area consistent with CTS 3.15.1 requirements. ITS SR 3.7.12.1 is also included to

periodically verify the system to be in operation during fuel handling in the area. NUREG SR 3.7.13.1 and SR 3.7.13.3 are not incorporated for the fuel handling area ventilation system since the system is placed in service prior to irradiated fuel movement in the fuel handling area and is not started on an actuation signal. These changes are consistent with current license basis.

36 NUREG 3.7.12 & 3.7.13 - NUREG SR 3.7.12.4 and SR 3.7.13.4 are not adopted. Per Standard Review Plan Section 6.4, only control room emergency ventilation system designs with < 0.25 volume changes per hour are required to provide periodic verification of the pressurization capability for the control room. The penetration room ventilation system (PRVS) also provides ≥ 0.25 volume changes per hour. Therefore, this Surveillance is also not adopted for the PRVS. The fuel handling area ventilation system (FHAVS) is not designed to pressurize the fuel handling area. Rather it provides a suction from the area immediately above the fuel pool. Therefore, the pressurization test is also not adopted for the FHAVS. These changes are consistent with current license basis.

37 NUREG 3.7.10 - The Note in Required Action C.1 is not required for this unit since the toxic gas mode of operation is the same as the radiation protection (emergency) mode, i.e., isolation, filtration, and pressurization with makeup air. The wording of Required Action C.1 is also revised to reflect that the CREVS must be placed in operation since there is only the emergency mode of operation, i.e., CREVS does not operate in a "normal" operation mode. This change is consistent with current license basis.

38 NUREG 3.7.18 - The unit safety analysis does consider a steam generator inventory; however, the inventory assumed in the analysis for a main steam line break is conservatively considered to be well above the level at which the steam generator aspirator ports would be flooded. Administrative controls have been sufficient to assure compliance with the safety analysis assumption, and an upper steam generator level is not included in the CTS. Therefore, the controls for these valves are proposed to continue to be administrative and not incorporated in the ITS. This change is consistent with current license basis.

39 NUREG 3.7.10 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 3.7.10 Background - The CREVS description is revised to reflect unit design.
B 3.7.10 ASA - The CREVS discussion of the applicable safety analysis is revised to be consistent with the unit specific analyses and license basis.
B 3.7.10 LCO - The discussion is revised to identify the correct components, i.e., no heater, demister or valves, and to use unit specific terminology.
B 3.7.10 Condition C - The discussion is revised to omit a misleading statement. Placing the system in operation does not ensure that "the remaining train is OPERABLE."
B 3.7.10 SR 3.7.10.1 - The discussion is revised to reflect unit design, i.e., without heaters.

B 3.7.10 SR 3.7.10.2 - The statement regarding compliance with Regulatory Guide 1.52 is redundant to the requirements of the VFTP and is unnecessary.

40 NUREG 3.7.11 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 3.7.11 Background - The CREACS description is revised to reflect unit design.
B 3.7.11 ASA, LCO and Applicability - The CREACS discussion of the applicable safety analysis is revised to also address the habitability requirements portion of the license basis.
B 3.7.11 Condition B and Condition C - The condition description is corrected for accuracy.
B 3.7.11 Condition C - The discussion is revised to omit a misleading statement. Placing the system in operation does not ensure that "the remaining train is OPERABLE," and the system does not automatically actuate.

41 NUREG 3.7.12 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 3.7.12 Background, LCO, and Applicability - The PRVS description is revised to reflect unit design.
3.7-34 B 3.7.12 ASA – The PRVS description is revised to reflect unit design. In addition, information concerning types of system failures that are analyzed is deleted. The ANO-1 LOCA analysis assumes 50% of all leakage is processed by the PRVS with a 90% efficiency. The analysis does not consider failures of the type described in the NUREG Bases. This maintains the current license basis.
B 3.7.12 Required Action A.1 - The condition description is corrected to identify that the PRVS supports mitigation of reactor building leakage, not support the ECCS.
B 3.7.12 Required Actions B.1 and B.2 - The condition description is corrected for accuracy.
B 3.7.12 SR 3.7.12.1 - The discussion is revised to reflect unit design, i.e., without heaters.
B 3.7.12 SR 3.7.12.2 - The statement regarding compliance with Regulatory Guide 1.52 is redundant to the requirements of the VFTP and is unnecessary.

42 NUREG 3.7.13 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 3.7.13 BACKGROUND, ASA, and LCO - The FHAVS description is revised to reflect unit design.

B 3.7.13 SR 3.7.13.2 - The statement regarding compliance with Regulatory Guide 1.52 is redundant to the requirements of the VFTP and is unnecessary.

43 NUREG 3.7.6 and Bases - Incorporates TSTF-140, except for the incorporation of the criterion in the Applicable Safety Analyses, as was described in DOD-6.

44 NUREG 3.7.15 Bases - This change provides unit specific revisions to discussions of design, analysis, reference documents, or operational parameters or procedures.

B 3.7.15 All - The Spent Fuel Pool Boron Concentration Bases discussions are revised to reflect unit specific design and analysis.

3.7-35 B 3.7.17 Background, ASA, and LCO - The secondary specific activity Bases discussions are revised to reflect unit specific design and analysis. The ANO-1 secondary activity limit is based on a consideration of the activity in the mass released following a rupture of a steam generator tube, a steam line break outside the reactor building, and a loss of load incident. The Safety Evaluation for Amendment 2, dated May 9, 1975, states that the controlling accident to determine the allowable secondary coolant activity would be the rupture of a steam generator tube. The NUREG is based on the assumption that the secondary activity limit is based on the steam line break. The ITS 3.7.4 Bases has been revised to delete the NUREG characterization of the basis for the secondary activity limit and information from the CTS Bases has been incorporated in the ASA discussion. This change retains the current license basis. This change also incorporates TSTF-173.

45 NUREG 3.7.16 Bases - Incorporates TSTF-210.

46 ITS SR 3.7.5.6 – This change incorporates CTS 4.8.1.e.4 requirements to verify that feedwater is delivered to each steam generator using the electric motor-driven EFW pump. This SR is required to be performed on an 18 month Frequency as established in CTS 4.8.1.e. The addition of this SR complements NUREG SR 3.7.5.5 in verifying that feedwater can actually be delivered to the steam generators. This change is consistent with current license basis.

47 NUREG SR 3.7.5.5 and Bases – Incorporates TSTF-268.

48 NUREG 3.7.6 and Bases - Incorporates TSTF-174.

3.7-14 49 NUREG-3.7.16 and Bases (ITS 3.7.15 and Bases) - Incorporates TSTF-255, Rev 1.

50 NUREG 3.7.12 and Bases (ITS 3.7.11 and Bases) Condition B has been revised to also apply when both PRVS trains are inoperable. CTS 3.13.1 requires two independent circuits of the PRVS to be operable. If one circuit of PRVS is made or found to be inoperable for any reason, 3.13.2 allows operation during the succeeding seven days provided the other circuit is operable. Failure to meet the requirements of 3.13.1 or 3.13.2 results in performing the actions of 3.13.3, which requires placing the reactor in the cold shutdown condition within 36 hours. NUREG 3.7.12 does not contain a Condition for both trains inoperable. Therefore, LCO 3.0.3 would be invoked. The CTS for PRVS does not require entry into LCO 3.0.3 since actions are provided in CTS 3.13.3, which would result in placing the reactor in cold shutdown in 36 hours, similar to the shutdown requirements of LCO 3.0.3. This change is consistent with the current license basis.

51 NUREG 3.7.1 Bases and NUREG 3.7.17 Bases - The term "AOO" is used in the GDCs, but the ANO-1 license basis is contingent upon discussion of "abnormalities" as defined and listed in SAR, Section 14.1. The ANO-1 SAR was written partially based on the guidance given in a "Guide to the Organization and Contents of Safety Analysis Reports" issued by the Atomic Energy Commission on June 30, 1966. This document discusses what transients or "abnormalities" should be considered for Core and Coolant Boundary Protection Analysis. Statements concerning the GDC criteria are modified in the ITS to reference the current license basis description in the Unit 1 SAR.

52 NUREG SR 3.7.11.1 and Bases - NUREG SR 3.7.11.1 has been deleted. The ITS will retain the current testing requirements specified in CTS 4.10.1.a and CTS 4.10.1.b. These surveillances were approved by the NRC for ANO-1 in a Safety Evaluation associated with Amendment 196 dated May 19, 1999. The ANO CREACS trains are not instrumented to an extent that would allow the specific requirement of NUREG SR 3.7.11.1 to be adequately performed. Generic Letter 89-13, Enclosure 2, describes a program acceptable to the NRC for heat exchanger testing. Frequent regular maintenance of a heat exchanger in lieu of testing for degraded performance is provided as an acceptable alternative action acceptable to the NRC. Periodic maintenance was credited for the CREACS in the ANO response to GL 89-13. The current combination of monthly functional testing and 18 month flow verification, when combined with preventative maintenance activities is sufficient to ensure the availability of the CREACS. This change retains the current license basis.

53 NUREG 3.7.16 Bases - This change provides unit specific revisions to discussions of design, analysis, or operational parameters or procedures.

ANO-290

54. Incorporated TSTF-287, Rev. 5, with the following exceptions.

TSTF-287 has been revised in ITS 3.7.11 to refer to plant specific terminology of the PRVS negative pressure boundary. In addition, according to SAR Section 6.5, the purpose of the PRVS is to collect and process potential reactor building penetration leakage to minimize environmental activity levels resulting from post-accident reactor building leaks. The system is not used to protect plant personnel from potential hazards such as radioactive contamination, toxic chemical, smoke, temperature and relative humidity, and physical security. The applicable GDCs have been revised to refer only to GDC 64, as GDCs 19 and 63 do not apply to the design of the PRVS. This change is consistent with the current license basis.

TSTF-287 has not been incorporated in ITS 3.7.12 due to the plant specific design of the FHAVS, and the analytical basis for the FHAVS that recognizes that in the event the supply fan is inoperable, air will be made up through the non-insulated metal siding and through the door at the railroad track and up through the equipment hatch (SAR Table 9-24).

ITS DISCUSSION OF DIFFERENCES

ANO-292 55. Incorporated TSTF-340, Rev. 3. TSTF-340 has been revised to reflect the ANO-1 specific design that consists of one turbine driven EFW pump and one motor driven EFW pump. These changes are editorial in nature.

ANO-293 56. Incorporated TSTF-352, Rev 1.

3.7-31 57. NUREG 3.7.8 Bases (ITS 3.7.7 Bases) – LCO discussion has been revised to incorporate a statement that for both SWS loops to be considered OPERABLE, the required SW pumps must be powered from independent essential buses, to provide redundant and independent flow paths. This change is consistent with the current license basis.

3.7-14 58. NUREG 3.7.14 Bases (ITS 3.7.13) – Applicable Safety Analyses discussion has been revised to be consistent with the Applicable Safety Analyses discussion in the Bases for ITS 3.9.6, "Refueling Canal Water Level." Since the water level limits in both ITS 3.9.6 and ITS 3.7.13 are based on the fuel handling accident, the same discussion should be incorporated in both Bases. This provides more consistency within the ITS Bases, and reduces the effort required to make changes to these Bases sections. This change is considered to be administrative in that it affects Bases information only, and provides greater consistency.

59. NUREG 3.7.8 and 3.7.9 Bases (ITS 3.7.7 and 3.7.8 Bases) – The CTS do not provide guidance on the Operability of the Service Water System (SWS) or Emergency Cooling Pond (ECP) during operation in MODE 5 and MODE 6. ANO has historically interpreted these systems to be support systems during operations in those conditions in which the CTS does not specifically require their Operability. The ITS incorporates a statement from the NUREG that states that in MODES 5 and 6, the OPERABILITY requirements of the Ultimate Heat Sink (UHS) are determined by the systems it supports. This would appear to state that in MODES 5 and 6, that the UHS is considered to be support system. Incorporating this statement in the ITS for the ECP is not intended to result in the addition of any requirements above those specified in the CTS. In order to clarify this for the operator, the Applicability discussion has been revised by the addition of information concerning Operability in MODE 5 and MODE 6. This additional information is consistent with guidance contained in Generic Letter 91-18, "Information to Licensees Regarding Two New Inspection Manual Sections On Resolutions Of Degraded and Nonconforming Conditions And On Operability and with ANO interpretations of the CTS."

ANO-239 60. NUREG 3.7.10 and Bases (ITS 3.7.9 and Bases) – Revised to provide an LCO Note that one CREVS train shall be capable of automatic actuation.

NUREG SR 3.7.10.3 and Bases (ITS SR 3.7.9.3 and Bases) – Revised to retain the CTS 4.10.2.d.2 wording the requires a verification that the Control Room automatically isolates and switches into a recirculation mode of operation..

This is consistent with the ANO-1 current license basis, as discussed in DOC-A16.

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 ~~The~~ Seven MSSVs shall be OPERABLE ~~as specified in Table 3.7.1-1 and Figure 3.7.1-1~~ on each main steam line. 3.4.1.2 1 Note *

<INSERT 3.7-1A>

APPLICABILITY: MODES 1, 2, and 3. 3.4.1

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each MSSV. NA

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One or more required MSSVs inoperable.	A.1 Reduce power ~~to less than the reduced power requirement of~~ in accordance with ~~Figure~~ Table 3.7.1-1.	4 hours	3.4.2 1
	AND		
	A.2 Reduce the nuclear overpower trip setpoint in accordance with ~~Figure~~ Table 3.7.1-1.	~~12~~ 36 hours	2 NA 1
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours	3.4.2
OR	AND		
One or more steam generators with less than ~~3~~ two MSSVs OPERABLE.	B.2 Be in MODE 4.	12 hours	3.4.2

<INSERT 3.7-1A>

-----------------------NOTE---
During main steam system hydrotesting in MODE 3, one MSSV is required to be OPERABLE on each main steam line with lift setpoints adjusted to allow testing.

3.7-01

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	-------------------NOTE------------------- Only required to be performed in MODES 1 and 2. --	
	Verify each required MSSV lift setpoint ~~per Table 3.7.1-1~~ in accordance with the Inservice Testing Program. Following testing, as-left lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program

NA

1

T4.1-2 #4

edit

Combine into SR 3.7.1.1

Table 3.7.1-1 (page 1 of 1)
Main Steam Safety Valve Lift Settings

VALVE NUMBER	LIFT SETTING (psig ± [3]%)
[2] MSSVs/steam generator	[1050]
[7] MSSVs/steam generator	[≤ 1100]

1

NA

Allowable Power Level and RPS Nuclear Overpower Trip Allowable Value Versus OPERABLE Main Steam Safety Valves

MINIMUM NUMBER OF MSSVs OPERABLE (PER SG)	MAXIMUM ALLOWABLE POWER LEVEL (% RTP)	RPS NUCLEAR OVERPOWER TRIP ALLOWABLE VALUE (% RTP)
6	85.7	89.9
5	71.4	74.9
4	57.1	59.9
3	42.8	44.9
2	28.5	29.9

1

NA

3.705-01

Move to Bases (RA A.1 & A.2)

1

NA

$$\frac{WY}{Z} = SP; \quad RP = \frac{Y}{Z} \times 100\%$$

W = Nuclear overpower trip setpoint for four pump operation as specified in LCO 3.3.1.

Y = Total OPERABLE MSSV relieving capacity per steam generator based on summation of individual OPERABLE MSSV relief capacities per steam generator [lb/hour].

Z = Required relieving capacity per steam generator of [6,585,600] lb/hour.

SP = Nuclear overpower trip setpoint (not to exceed W).

RP = Reduced power requirement (not to exceed RTP).

These equations are graphically represented below.
Operation is restricted to the area below and to the right of line BCDE.

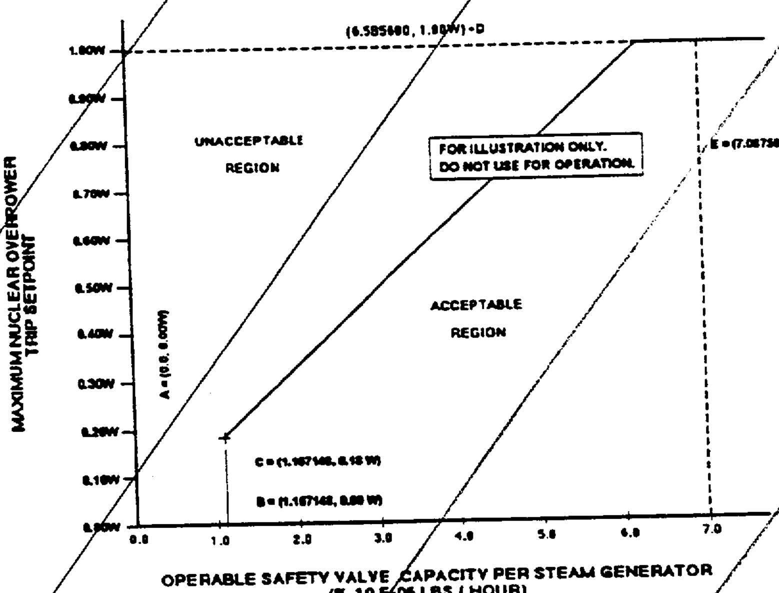


Figure 3.7.1-1 (page 1 of 1)
Reduced Power and Nuclear Overpower Trip Setpoint versus OPERABLE Main Steam Safety Valves

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

LCO 3.7.2 Two MSIVs shall be OPERABLE. 3.4.1.5

APPLICABILITY: MODE 1, 2, and 3. ~~MODES 2 and 3 except when all MSIVs are closed and [deactivated].~~ 3 3.4.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One ~~MSIV~~ or more MSIV(s) inoperable in MODE 1 or 2.	A.1 Restore MSIV(s) to OPERABLE status.	~~[8]~~ 24 hours	4 edit 3.4.2
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE ~~2~~ 3.	~~8~~ 12 hours	4 3.4.2
C. ---------NOTE--------- Separate Condition entry is allowed for each MSIV. ---------------------- One or more MSIV(s) inoperable in MODE ~~2 or~~ 3.	C.1 Close MSIV. AND C.2 Verify MSIV is closed.	~~[8]~~ 48 hours Once per 7 days	NA 3.4.2 4 NA edit 4
D. Required Action and associated Completion Time of Condition ~~B or~~ C not met.	D.1 Be in MODE ~~3~~ 4. ~~AND~~ ~~D.2 Be in MODE 4.~~	~~8~~ 24 hours ~~12 hours~~	3.4.2 4 7

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
SR 3.7.2.1	---NOTE--- Only required to be performed in MODES 1 and 2. --- Verify isolation ~~closure~~ time of each MSIV is ~~≤ [6] seconds on an actual or simulated actuation signal.~~ within the limits specified in the Inservice Testing Program.	In accordance with the Inservice Testing Program ~~or [18] months~~	NA T4.1-2 # 13 8
SR 3.7.2.2	---NOTES--- 1. Only required to be performed in MODES 1 and 2. 2. Not required to be met when SG pressure is <750 psig. --- Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.	18 months	NA 3.5.1.16 NA

3.7-02

~~[MFSVs, MFCVs, and Associated SFCVs]~~ I
3.7.3

Main Feedwater Block Valves, Low Load Feedwater Control Valve and Startup Feedwater Control Valves

9

3.7 PLANT SYSTEMS

3.7.3 [Main Feedwater ~~Stop~~ Isolation Valves (MFIVs), ~~Main Feedwater Control Valves (MFCVs), and Associated Startup Feedwater Control Valves (SFCVs)~~]

9

LCO 3.7.3 ~~[Two] [MFSVs], [MFCVs], [or associated SFCVs]~~ All MFIVs shall be OPERABLE. 3.4.1.5

APPLICABILITY: MODES 1, 2, and 3 ~~except when all [MFSVs], [MFCVs], [or associated SFCVs] are closed and [deactivated] [or isolated by a closed manual valve]~~. 3.4.1 3

ACTIONS

---NOTE---

Separate Condition entry is allowed for each valve. NA

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One ~~[MFSV]~~ MFIV in one or more flow paths inoperable.	A.1 Close or isolate ~~[MFSV]~~ MFIV. AND A.2 Verify ~~[MFSV]~~ MFIV is closed or isolated.	~~[8 or~~ 72~~]~~ hours Once per 7 days	NA 3.4.2 9 NA
B. One ~~[MFCV]~~ Main Feedwater Block Valve in one or more flow paths inoperable.	B.1 Close or isolate ~~[MFCV]~~. AND B.2 Verify ~~[MFCV]~~ is closed or isolated.	~~[8 or~~ 72~~]~~ hours Once per 7 days	NA 3.4.2 9 NA

(continued) 9

<INSERT 3.7-7A>

3.7-02

<INSERT 3.7-7A>

			CTS
C. One Low Load Feedwater Control Valve in one or more flow paths inoperable.	C.1 Close or isolate Low Load Feedwater Control Valve. AND C.2 Verify Low Load Feedwater Control Valve is closed or isolated.	72 hours Once per 7 days	N/A

3.7-02

9

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
D ~~C~~. One ~~[SFCV]~~ Startup Feedwater Control Valve in one or more flow paths inoperable.	D ~~C~~.1 ~~Close or isolate [SFCV].~~ AND D ~~C~~.2 Verify ~~[SFCV]~~ is closed ~~or isolated~~.	~~[8 or~~ 72 ~~]~~ hours Once per 7 days	NA 9
E ~~D~~. Two valves in the same flow path inoperable for one or more flow paths.	E ~~D~~.1 Isolate affected flow path.	8 hours	NA
F ~~E~~. Required Action and associated Completion Time not met.	F ~~E~~.1 Be in MODE 3. AND F ~~E~~.2 Be in MODE 4.	6 hours 12 hours	3.4.2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.7.3.1	-------------------NOTE------------------- Only required to be performed in MODES 1 and 2. --- Verify the isolation ~~closure~~ time of each MFIV ~~[MFIV],~~ ~~[MFCV], and [SFCV]~~ Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves is ~~≤ [7] seconds on an actual or simulated actuation signal~~ within the limits provided in the Inservice Testing Program.	In accordance with the of Inservice Testing Program ~~or [18] months]~~	NA 9 T4.1-2 #14 8

<INSERT 3.7-8A>

<INSERT 3.7-8A>

SR 3.7.3.2	---------------------NOTE--------------------- 1. Only required to be performed in MODES 1 and 2. 2. Not required to be met when SG pressure is < 750 psig. ---		NA 3.5.1.16
3.7-02	Verify that each MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve, and Startup Feedwater Control Valve actuates to the isolation position on an actual or simulated actuation signal.	18 months	NA

{ Reviewers Note: RSTS 3.7.17 has been renumbered and moved to fill ITS LCO 3.7.4 }

11

NA

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Vent Valves (AVVs)

LCO 3.7.4 [Two] AVVs [lines per steam generator] shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required AVV [line] inoperable.	A.1 ---------NOTE--------- LCO 3.0.4 is not applicable. ---------------------- Restore required AVV [line] to OPERABLE status.	[7 days]
[B. Two or more required AVV [lines] inoperable.	B.1 Restore one AVV [line] to OPERABLE status.	24 hours]
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 Be in MODE 4 without reliance upon steam generator for heat removal.	18 hours

(11)

NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify one complete cycle of each AVV.	[18] months
[SR 3.7.4.2	Verify one complete cycle of each AVV block valve.	[18] months]

3.7-04

3.7 PLANT SYSTEMS

3.7.5 Emergency Feedwater (EFW) System

12

LCO 3.7.5 Two ~~[Three]~~ EFW trains shall be OPERABLE. 3.4.3.1 3.4.3.2

----------NOTE----------
Only one EFW train, which includes a motor driven pump, is required to be OPERABLE in MODE 4. 3.4.3.1

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal. 3.4.3.1 3.4.3.2

AND-292 3.7-05

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
~~[A. One steam supply to turbine driven EFW pump inoperable.]~~	A.1	Restore ~~steam supply~~ affected equipment to OPERABLE status.	7 days ~~AND~~ ~~10 days from discovery of failure to meet the LCO~~	3.4.4.2 NA 55
B. One EFW train inoperable ~~[~~for reasons other than Condition A~~]~~ in MODE 1, 2, or 3. OR ---- NOTE ---- Only applicable if MODE 2 has not been entered following refueling ------ Turbine driven EFW pump inoperable in MODE 3 following refueling.	B.1	Restore EFW train to OPERABLE status.	72 hours ~~AND~~ ~~10 days from discovery of failure to meet the LCO~~	3.4.4.3 NA

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME	
C. Required Action and associated Completion Time of Condition A ~~or~~ (or B) not met. ~~OR~~ ~~Two EFW trains inoperable in MODE 1, 2, or 3.~~	C.1	Be in MODE 3.	6 hours	3.4.4.2 3.4.4.3
	AND			
	C.2	Be in MODE 4.	6 (18) 6 hours	3.4.4.2 3.4.4.3 12
D. Two ~~Three~~ EFW trains inoperable in MODE 1, 2, or 3.	D.1	---------NOTE--------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one EFW train is restored to OPERABLE status. --------------------- Initiate action to restore one EFW train to OPERABLE status.	Immediately	12 3.4.4.5 3.4.4.5
E. Required EFW train inoperable in MODE 4.	E.1	Initiate action to restore EFW train to OPERABLE status.	Immediately	3.4.4.1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
SR 3.7.5.1	Verify each EFW manual, power operated, and automatic valve in each water flow path and in both steam supply flow paths to the steam turbine driven pump~~s~~, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days	4.8.1.b
SR 3.7.5.2	-------------------NOTE------------------- Not required to be performed for the turbine driven ~~EFW pumps~~, until 9 ~~[~~24~~]~~ hours after reaching ~~[800]~~ ≥ 750 psig in the steam generators. -- Verify the developed head of each EFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program ~~[31] days on a STAGGERED TEST BASIS~~	3.4.3.2 Note ** edit 4.8.1.a.1 4.8.1.a.5 13 4.8.1.a.2
SR 3.7.5.3	-------------------NOTE~~S~~------------------- ~~1. Not required to be performed until [24] hours after reaching [800] psig in the steam generators.~~ ~~2.~~ Not ~~applicable~~ required to be met in MODE 4~~.~~ when steam generator is relied upon for heat removal -- Verify each EFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	9 9 ~~[18]~~ months	NA 4.8.1.e.2 NA edit 14 4.8.1.e.1 4.8.1.e.2 4.8.1.e.5

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY	
SR 3.7.5.4	------NOTE~~S~~------ ~~1. Not required to be performed until [24] hours after reaching [800] psig in the steam generators.~~ ~~2.~~ Not ~~applicable~~ required to be met in MODE 4 when steam generator is relied upon for heat removal --------------- Verify each EFW pump starts automatically on an actual or simulated actuation signal.	~~[18]~~ 18 months	NA & 4.8.1.e.2 (14) edit NA 4.8.1.e.2 4.8.1.e.3
SR 3.7.5.5	Verify proper alignment ~~of the required~~ EFW flow paths by verifying ~~f~~valve manual alignment ~~[flow]~~ from the condensate storage tank to each steam generator. "Q"	Prior to entering MODE 2 whenever ~~plant~~ unit has been in MODE 5 ~~or 6 for~~ , MODE 6, or defueled for a cumulative period of > 30 days	4.8.1.c edit (15) (47)
~~[SR 3.7.5.6~~	~~Perform a CHANNEL FUNCTIONAL TEST for the EFW pump suction pressure interlocks.~~	~~31 days]~~	(16)
~~[SR 3.7.5.7~~	~~Perform a CHANNEL CALIBRATION for the EFW pump suction pressure interlocks.~~	~~[18] months]~~	(16)
SR 3.7.5.6	Verify that feedwater is delivered to each steam generator using the motor-driven EFW pump.	18 months	(46)

Generic term change
CST → QCST

edit

3.7 PLANT SYSTEMS

3.7.6 Q Condensate Storage Tank (CST) 43

LCO 3.7.6 The ~~[two]~~ CST ~~level(s)~~ shall be OPERABLE ~~≥ [250,000] gal~~. 3.4.1.3

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal. 3.4.1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
A. The ~~[two]~~ CST ~~level(s) not within limits~~ inoperable.	A.1	Verify by administrative means OPERABILITY of backup water supply.	4 hours AND Once per 12 hours thereafter	43
	AND			
	A.2	Restore CST ~~level(s)~~ to ~~within limit~~ OPERABLE status.	7 days	43 3.4.2
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours	3.4.2
	AND			
	B.2	Be in MODE 4 without reliance on steam generator for heat removal.	~~[18]~~ 24 hours	3.4.2 56

ANO-293

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.6.1	Verify CST ~~[level]~~ volume is ≥ ~~[250,000]~~ 32,300 gal.lons	12 hours

edit NA

17

NA

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Two CCW trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CCW train inoperable.	A.1	--------NOTES-------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources—Operating," for emergency diesel generator made inoperable by CCW. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops—MODE 4," for decay heat removal made inoperable by CCW. -------------------- Restore CCW train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	36 hours

(17)

NA

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.7.1	-------------------NOTE------------------- Isolation of CCW flow to individual components does not render CCW System inoperable. -- Verify each CCW manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.7.2	Verify each CCW 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 3.7.7.3	Verify each CCW pump starts automatically on an actual or simulated actuation signal.	[18] months

3.7 PLANT SYSTEMS

3.7.~~8~~ 7 Service Water System (SWS) 3.3.1 (C) edit 3.3.1 (E) 3.3.1 (I) 3.3.4(D)

LCO 3.7.~~8~~ 7 Two SWS ~~trains~~ loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4. 3.3.1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
A. One SWS ~~train~~ loop inoperable.	A.1	--------NOTES-------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources—Operating," for emergency diesel generator made inoperable by SWS. 2. Enter Applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops—MODE 4," for decay heat removal made inoperable by SWS. --------------------- Restore SWS ~~train~~ loop to OPERABLE status.	72 hours	edit NA edit NA edit 3.3.6

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME	
B. Required Action and associated Completion Time ~~of Condition A~~ not met.	B.1	Be in MODE 3.	6 hours	3.3.6 edit
	AND			
	B.2	Be in MODE 5.	36 hours	3.3.6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.7.~~8~~ 7.1	-------------------NOTE------------------- Isolation of SWS flow to individual components does not render the SWS inoperable. ---		NA
	Verify each SWS manual, power operated, and automatic valve in the flow path servicing safety related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days	NA
SR 3.7.~~8~~ 7.2	Verify each SWS 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.	9 ~~[18]~~ months	T4.1-2 #9 & 4.5.1.1.2(a)(2) 4.5.2.1.2.C.3
SR 3.7.~~8~~ 7.3	Verify each required SWS pump starts automatically on an actual or simulated actuation signal.	[18] months	18 T4.1-2 #9

3.7-08

ECP ~~UHS~~
3.7.~~9~~ 8

3.7 PLANT SYSTEMS

3.7.~~9~~ 8 ~~Ultimate Heat Sink (UHS)~~ Emergency Cooling Pond (ECP)

LCO 3.7.~~9~~ 8 The ~~UHS~~ ECP shall be OPERABLE. 3.11.1

APPLICABILITY: MODES 1, 2, 3, and 4. 3.11.1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
~~A. One or more cooling towers with one cooling tower fan inoperable.~~	~~A.1~~	~~Restore cooling tower fan(s) to OPERABLE status.~~	~~7 days~~	19
A. ECP inoperable. ~~B. Required Action and associated Completion Time of Condition A not met~~ ~~OR~~ ~~UHS inoperable [for reasons other than Condition A]~~.	A ~~B~~.1 AND A ~~B~~.2	Be in MODE 3. Be in MODE 5.	6 hours 36 hours	3.11.2 3.11.2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.7.~~9~~ 8.1	Verify water level of ~~UHS~~ ECP is ≥ ~~[562]~~ 5 ft ~~[mean sea level]~~.	24 hours	20 4.13.1.1 3.11.1

(continued)

--- NOTE ---
Only required to be performed from June 1 through September 30.

ECP
~~UHS~~
3.7.~~9~~ 8

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
[SR 3.7.~~9~~ 8.2	Verify average water temperature ~~of UHS~~ is ≤ ~~[90]~~ 100 °F.	24 hours]	20 4.13.1.2 3.11.1
~~[SR 3.7.9.3~~	~~Operate each cooling tower fan for ≥ [15] minutes.~~	~~31 days]~~	19
SR 3.7.8.3	Verify contained water volume of ECP ≥ 70 acre-ft at a water level of 5 ft.	12 months	20 4.13.1.3 3.11.1
SR 3.7.8.4	Verify earth portions of stone covered embankments and spillway of ECP: a. Have not been eroded or undercut by wave action, and b. Do not show apparent changes in visual appearance or other abnormal degradation from as-built condition.	12 months	4.13.1.4

3.7-11

3.7 PLANT SYSTEMS

3.7.~~10~~ 9 Control Room Emergency Ventilation System (CREVS)

AND-290
ANO-239

LCO 3.7.~~10~~ 9 Two CREVS trains shall be OPERABLE.

60
3.9.2.1
3.8.18
N/A
54
3.9.2.1
3.8.18

<INSERT 3.7-23A>

APPLICABILITY: MODES 1, 2, 3, and 4, ~~[5, and 6,]~~
[During movement of irradiated fuel assemblies ~~]~~.
~~[During CORE ALTERATIONS]~~.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREVS train inoperable.	A.1 Restore CREVS train to OPERABLE status.	7 days
~~B~~ C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	~~B~~ C.1 Be in MODE 3. AND ~~B~~ C.2 Be in MODE 5.	6 hours 36 hours
~~C~~ D. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies ~~[, or during CORE ALTERATIONS]~~.	~~C~~ D.1 ~~NOTE: Place in emergency mode if automatic transfer to emergency mode inoperable.~~ Place OPERABLE CREVS train in emergency recirculation mode. OR	Immediately
B. Two CREVS trains inoperable due to inoperable control room boundary in MODES 1, 2, 3 and 4.	B.1 Restore control room boundary to OPERABLE status.	24 hours

3.9.2.2
3.9.2.2
3.9.2.2
54
37
NA
NA
37

AND-290

(continued)

ANO-239
ANO-290

<INSERT 3.7-23A>

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

1. The control room boundary may be opened intermittently under administrative controls.

2. One CREVS train shall be capable of automatic actuation.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
D ~~C~~. (continued)	~~C.2.1~~	~~Suspend Core ALTERATIONS.~~	~~Immediately~~	29 NA
	~~AND~~			
	D ~~C~~.2.~~2~~	Suspend movement of irradiated fuel assemblies.	Immediately	54 NA
E ~~D~~. Two CREVS trains inoperable during movement of irradiated fuel assemblies ~~[, or during CORE ALTERATIONS]~~.	E D.1	Suspend movement of irradiated fuel assemblies.	Immediately	NA
	~~AND~~			
	~~D.2~~	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~	29
F ~~E~~. Two CREVS trains inoperable during MODE 1, 2, 3, or 4 for reasons other than Condition B.	F ~~E~~.1	Enter LCO 3.0.3.	Immediately	54 NA

AND-290

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.7.~~10~~ 9.1	Operate each CREVS train for ~~[≥ 10 continuous hours with the heaters operating or (for system without heaters)~~ ≥ 15 minutes~~]~~.	31 days	4.10.2.a

(continued)

the CREVS automatically isolates the Control Room and switches into a recirculation mode of operation

AN0-239

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.7.~~10~~ 9.2	Perform required CREVS filter testing in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~.	In accordance with the ~~[~~VFTP~~]~~	4.10.2
SR 3.7.~~10~~ 9.3	Verify ~~[each CREVS train actuates] [or the control room isolates]~~ on an actual or simulated actuation signal.	~~[~~18~~]~~ months	4.10.2.d.2 60
~~SR 3.7.10.4~~	~~Verify one CREVS train can maintain a positive pressure of ≥ [0.125] inches water gauge relative to the adjacent [area] during the [pressurization] mode of operation at a flow rate of ≤ [3300] cfm.~~	~~[18] months on a STAGGERED TEST BASIS~~	31
SR 3.7.~~10.5~~ 9.4	Verify the system makeup flow rate is ≥ ~~[270]~~ 300 and ≤ ~~[330]~~ 366 cfm when supplying the the control room with outside air.	[18] months	32

3.7-12

3.7 PLANT SYSTEMS

3.7.~~11~~ 10 Control Room Emergency Air ~~Temperature Control~~ Conditioning System (CREA~~T~~CS)

LCO 3.7.~~11~~ 10 Two CREA~~T~~CS trains shall be OPERABLE. 3.9.1.1

APPLICABILITY: MODES 1, 2, 3, and 4, ~~[5, and 6,]~~. 3.9.1.1
[During movement of irradiated fuel assemblies~~,~~].
~~[During CORE ALTERATIONS]~~.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
A. One CREA~~T~~CS train inoperable.	A.1	Restore CREA~~T~~CS train to OPERABLE status.	30 days	3.9.1.2
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1	Be in MODE 3.	6 hours	3.9.1.2
	AND			
	B.2	Be in MODE 5.	36 hours	3.9.1.2
[C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies ~~[, or during CORE ALTERATIONS]~~.	C.1	Place OPERABLE CREA~~T~~CS train in operation.	Immediately	NA
	OR			
	C.2	Suspend movement of irradiated fuel assemblies.	Immediately]	NA

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME	
D. Two CREAFCS trains inoperable during movement of irradiated fuel assemblies [, or during CORE ALTERATIONS].	D.1	Suspend movement of irradiated fuel assemblies.	Immediately	NA
E. Two CREAFCS trains inoperable during MODE 1, 2, 3, or 4.	E.1	Enter LCO 3.0.3.	Immediately	NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
~~SR 3.7.11.1~~	~~Verify each CREATCS train has the capability to remove the assumed heat load.~~	~~[18] months~~
SR 3.7.10.1	Verify each CREACS train starts, operates for at least 1 hour, and maintains control room air temperature ≤ 84 °F D.B.	31 days
SR 3.7.10.2	Verify system flow rate of 9900 cfm ±10%	18 months

52

3.7-13

ANO-290

3.7 PLANT SYSTEMS

3.7.~~12~~ 11 ~~Emergency~~ Penetration Room Ventilation System ~~(EVS)~~ (PRVS)

LCO 3.7.~~12~~ 11 Two PR~~E~~VS trains shall be OPERABLE. 3.13.1

Note

The penetration room negative pressure boundary may be opened intermittently under administrative control. 54 N/A

APPLICABILITY: MODES 1, 2, 3, and 4. 3.13.1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
A. One PR~~E~~VS train inoperable.	A.1	Restore PR~~E~~VS train to OPERABLE status.	7 days	3.13.2
B. Two PRVS trains inoperable due to inoperable penetration room negative pressure boundary	B.1	Restore penetration room negative pressure boundary to OPERABLE status	24 hours	54 NA
C ~~B~~. Required Action and associated Completion Time not met. OR Both PRVS trains inoperable	C ~~B~~.1	Be in MODE 3.	6 hours	
	AND C ~~B~~.2	Be in MODE 5.	36 hours	3.13.3 50

ANO-290

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.7.~~12~~ 11.1	Operate each PR~~E~~VS train for ~~[≥ 10 continuous hours with the heaters operating or (for systems without heaters)~~ ≥ 15 minutes].	31 days	4.11.5
SR 3.7.~~12~~ 11.2	Perform required PR~~E~~VS filter testing in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~. 6	In accordance with the ~~[~~VFTP~~]~~ 6 6	3.13.1 4.11.1 4.11.2 4.11.4

(continued)

PR EVS
3.7.11

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.7.11.3	Verify each PR EVS train actuates on an actual or simulated actuation signal.	~~[18]~~ 6 months	4.11.3 3.13.1.f
~~SR 3.7.12.4~~	~~Verify one EVS train can maintain a pressure ≤ [ ] inches water gauge relative to atmospheric pressure during the [post accident] mode of operation at a flow rate of ≤ [3000] cfm.~~	~~[18] months on a STAGGERED TEST BASIS~~	36
~~SR 3.7.12.5~~	~~Verify each EVS filter cooling bypass damper can be opened.~~	~~[18] months~~	NA 33

FHAVS
~~FSPVS~~
3.7.~~13~~ 12

3.7 PLANT SYSTEMS

3.7.~~13~~ 12 Fuel ~~Storage Pool~~ Handling Area Ventilation System (F~~SP~~ HA VS) 35 3.15.1

LCO 3.7.~~13~~ 12 ~~[Two] FSPVS trains~~ The FHAVS shall be OPERABLE ~~,~~ and in operation.

APPLICABILITY: ~~[MODES 1, 2, 3, and 4,]~~ During movement of irradiated fuel assemblies in the fuel ~~building~~ handling area. 34 3.15.1

--- NOTE ---
LCO 3.0.3 is not applicable.
34 3.15.2

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
~~A. One FSPVS train inoperable.~~	~~A.1~~	~~Restore FSPVS train to OPERABLE status.~~	~~7 days~~	NA
~~[B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.~~ ~~OR~~ ~~Two FSPVs trains inoperable in MODE 1, 2, 3, or 4.~~	~~B.1~~ ~~AND~~ ~~B.2~~	~~Be in MODE 3.~~ ~~Be in MODE 5.~~	~~6 hours~~ ~~36 hours]~~	NA 34
~~C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the fuel building.~~	~~C.1~~ ~~OR~~ ~~C.2~~	~~Place OPERABLE FSPVS train in operation.~~ ~~Suspend movement of irradiated fuel assemblies in the fuel building.~~	~~Immediately~~ ~~Immediately~~	NA

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A ~~C~~. FHAVS ~~Two FSPVS trains~~ inoperable ~~during movement of irradiated fuel assemblies in the fuel building.~~ or not in operation	A ~~C~~.1 Suspend movement of irradiated fuel assemblies in the fuel ~~building.~~ handing area.	Immediately	3.15.2 35

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
~~SR 3.7.13.1~~	~~Operate each FSPVS train for [≥ 10 continuous hours with the heaters operating or (for systems without heaters) ≥ 15 minutes].~~	~~31 days~~	35
SR 3.7.~~13~~ 12.2	Perform required FHAVS ~~FSPVS~~ filter testing in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~.	In accordance with the ~~[~~VFTP~~]~~	3.15.1 4.17.1 4.17.2 4.17.3
~~SR 3.7.13.3~~	~~Verify each FSPVS train actuates on an actual or simulated actuation signal.~~	~~[18] months~~	35
~~SR 3.7.13.4~~	~~Verify one FSPVS train can maintain a pressure ≤ [ ] inches water gauge with respect to atmospheric pressure during the [post accident] mode of operation at a flow rate ≤ [3000] cfm.~~	~~[18] months on a STAGGERED TEST BASIS~~	36
SR 3.7.12.1	Verify FHAVS in operation.	12 hours	35 NA

(continued)

FHAVS
~~FSPVS~~
3.7.~~13~~ 12

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.7.13.5~~	~~Verify each FSPVS filter bypass damper can be opened.~~	~~[18] months~~

33

3.7-14

CTS

3.7 PLANT SYSTEMS Spent

3.7.~~14~~ 13 Fuel ~~Storage~~ Pool Water Level

LCO 3.7.~~14~~ 13 The fuel storage pool water level shall be ≥ 23 ft over the top of irradiated fuel assemblies seated in the storage racks. N/A

APPLICABILITY: During movement of irradiated fuel assemblies in Spent fuel ~~storage~~ pool. 10 N/A

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent ~~fuel~~ fuel ~~storage~~ pool water level not within limit.	A.1 --------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Suspend movement of irradiated fuel assemblies in Spent fuel ~~storage~~ pool.	Immediately

N/A 10

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.14.1 Verify the Spent fuel ~~storage~~ pool water level is ≥ 23 ft above the top of irradiated fuel assemblies seated in the storage racks.	7 days

10 N/A

3.7 PLANT SYSTEMS

3.7.~~15~~ 14 Spent Fuel Pool Boron Concentration

LCO 3.7.~~15~~ 14 The spent fuel pool boron concentration shall be ≥ ~~[500]~~ 1600 ppm. 3.8.17

APPLICABILITY: When fuel assemblies are stored in the spent fuel pool and a spent fuel pool verification has not been performed since the last movement of fuel assemblies in the spent fuel pool. 3.8.17

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. Spent fuel pool boron concentration not within limit.	-------------NOTE------------- LCO 3.0.3 is not applicable. -----------------------------		NA
	A.1 Suspend movement of fuel assemblies in the spent fuel pool.	Immediately	NA
	AND		
	A.2.1 Initiate action to restore spent fuel pool boron concentration to within limit.	Immediately	NA
	OR		
Initiate action to perform	A.2.2 ~~Verify by administrative means~~ a ~~[Region 2]~~ spent fuel pool verification ~~has been performed since the last movement of fuel assemblies in the spent fuel pool.~~	Immediately	30

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.~~13~~ 14.1	Verify the spent fuel pool boron concentration is ~~within limit.~~ ≥ 1600 ppm.	7 days

T4.1-3
#4
edit

49

3.7 PLANT SYSTEMS

3.7.~~16~~ 15 Spent Fuel ~~Assembly~~ Pool Storage

49

LCO 3.7.~~16~~ 15 The combination of initial enrichment and burnup of each ~~spent~~ fuel assembly stored in [Region 2] shall be within the acceptable ~~[burnup domain]~~ range of Figure 3.7.~~16~~ 15-1 or in accordance with Specification 4.3.1.1.

3.8.16 49
edit

APPLICABILITY: Whenever any fuel assembly is stored in [Region 2] of the spent fuel pool.

3.8.16

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 --------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Initiate action to move the noncomplying fuel assembly from [Region 2].	Immediately

3.8.16

NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.~~16~~ 15.1	Verify by administrative means the initial enrichment and burnup of the fuel assembly is in accordance with Figure 3.7.~~16~~ 15-1 or Specification 4.3.1.1.	Prior to storing the fuel assembly in [Region 2]

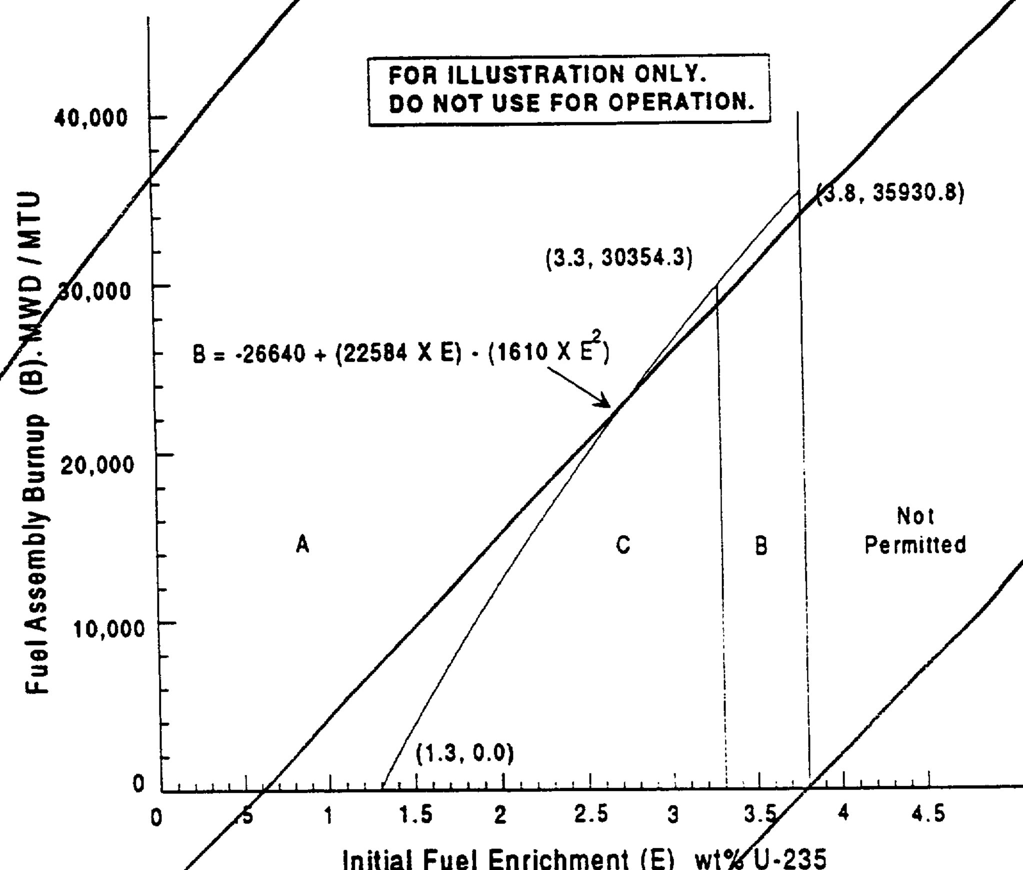
3.8.16
edit

3.7-16

⟨INSERT 3.7-37A⟩

Replace with ANO-1 CTS Figure 3.8.2

49



Category "A" Fuel - May be located anywhere within the storage racks.
Category "B" Fuel - Shall only be located adjacent to Category "A" Fuel or water holes within the storage racks.
Category "C" Fuel - Shall not be located adjacent to Category "B" Fuel.

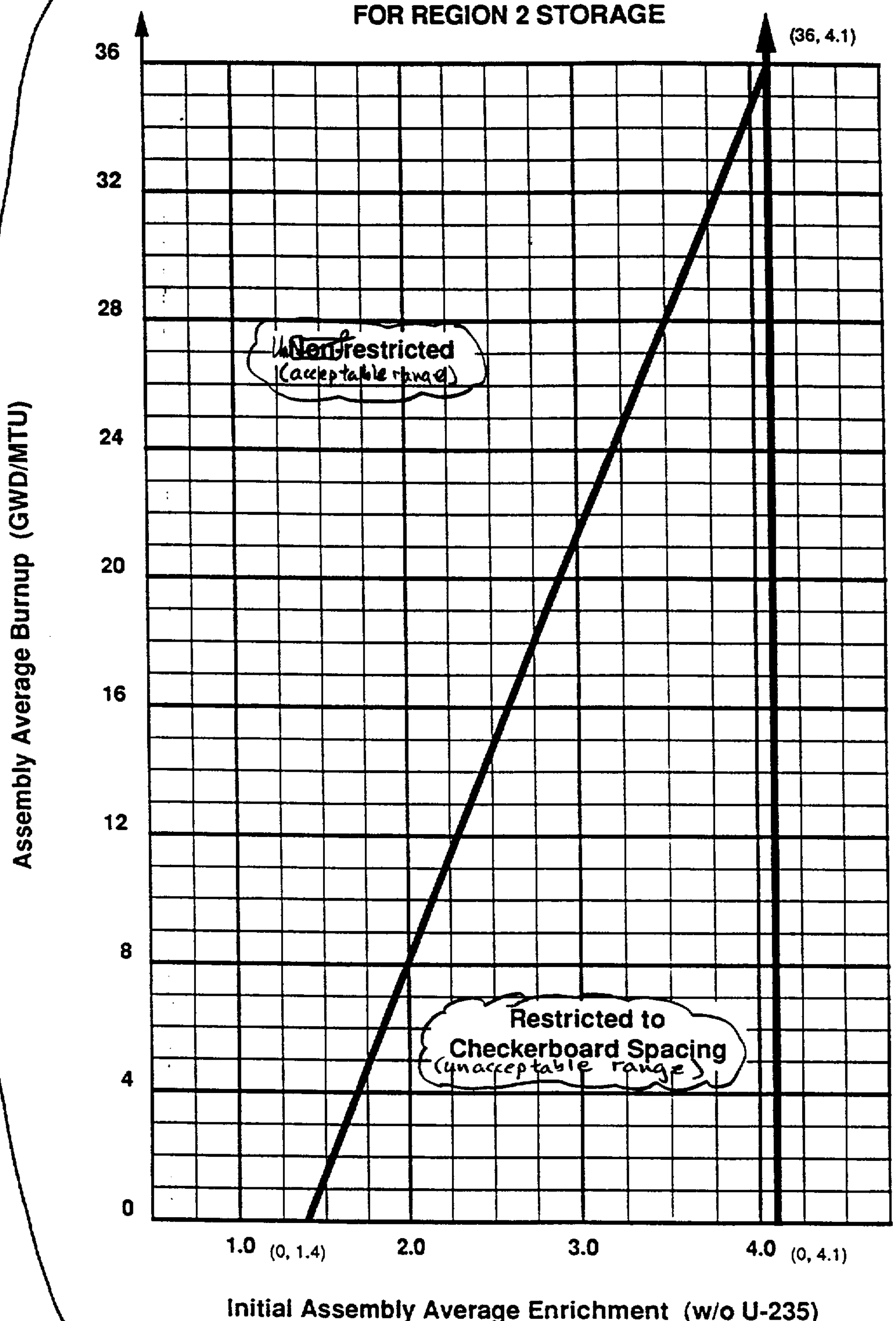
Figure 3.7.~~18~~ 15-1 (page 1 of 1)
Burnup versus Enrichment Curve for
Spent Fuel Storage Racks

F3.8.2

INSERT 3.7-37A

~~FIGURE 3.8.2~~

MINIMUM BURNUP VS. INITIAL ENRICHMENT FOR REGION 2 STORAGE



3.7 PLANT SYSTEMS

3.7.~~17~~ 4 Secondary Specific Activity

LCO 3.7.~~17~~ 4 The specific activity of the secondary coolant shall be ≤ ~~[0.10]~~ 0.17 μCi/gm DOSE EQUIVALENT I-131. 3.10

T4.1-3,#7
T4.1-3,#10

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. Specific activity not within limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours	3.10 3.10

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY	
SR 3.7.~~17~~ 4.1 Verify the specific activity of the secondary coolant is ≤ ~~[0.10]~~ 0.17 μCi/gm DOSE EQUIVALENT I-131.	~~[31]~~ 31 days	T4.1-3 #5.b & Note 4

38

3.7 PLANT SYSTEMS

3.7.18 Steam Generator Level

LCO 3.7.18 Water level of each steam generator shall be less than or equal to the maximum water level shown in Figure 3.7.18-1.

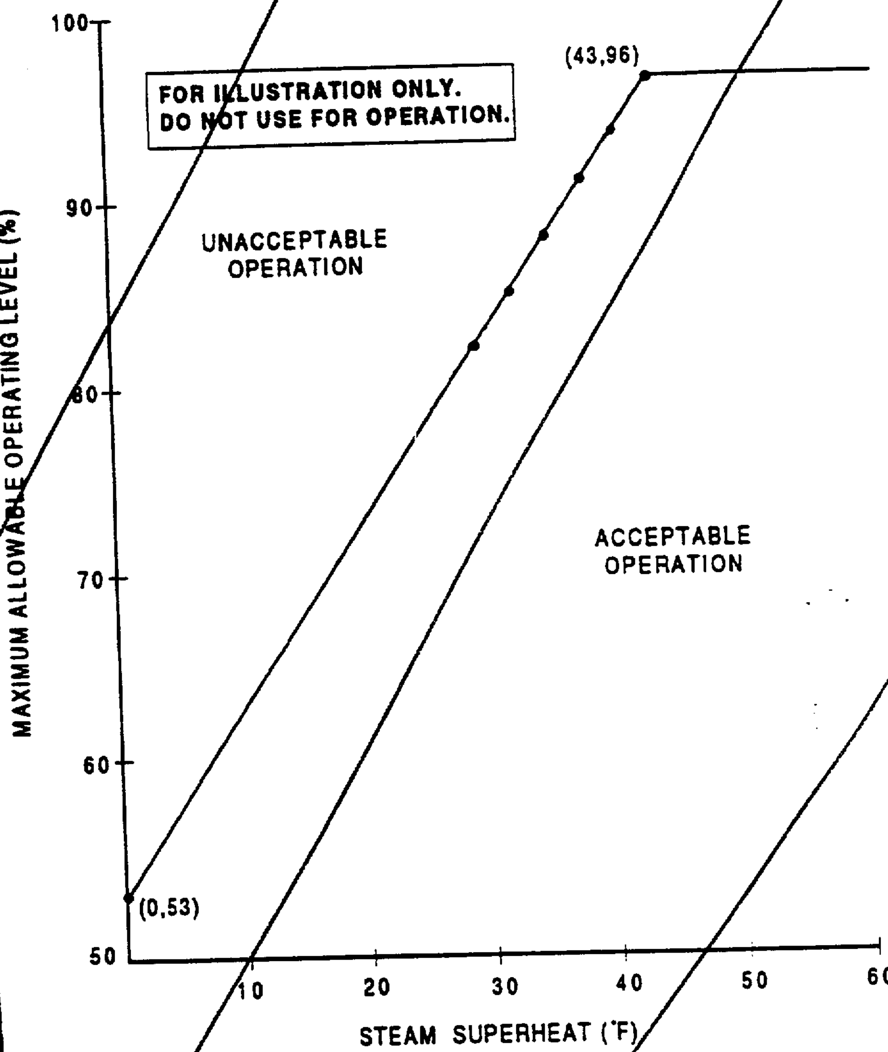
APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Water level in one or more steam generators greater than maximum water level in Figure 3.7.18-1.	A.1 Restore steam generator level to within limit.	15 minutes
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.18.1 Verify steam generator water level to be within limits.	12 hours



38

Figure 3.7.18-1 (page 1 of 1)
Maximum Allowable Steam Generator Level

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

~~Nine~~ Eight MSSVs are located on each main 10.3 steam header, outside ~~containment~~ the reactor building, upstream of the main steam isolation valves, as described in the ~~F~~SAR, Section ~~[5.2]~~ (Ref. 1). The MSSV ~~rated~~ capacity ~~passes the full steam flow at 112% RTP with the valves full open. This meets~~ is adequate to meet the requirements of the ASME Code, Section III ~~(Ref. 2)~~ (Ref. 1). The total capacity of 14 MSSVs is greater than the total steam flow at 102% RTP. The MSSV design includes staggered setpoints~~, according to Table 3.7.1-1 in the accompanying LCO,~~ so that only the needed number of valves will actuate. Staggered setpoints reduce the potential for valve chattering because of insufficient steam pressure to fully open ~~all~~ the valves ~~following a turbine reactor trip~~.

21 — 1 — edit.

APPLICABLE SAFETY ANALYSES

The design basis of the MSSVs ~~comes from~~ Ref~~erence~~ 2 ~~and its purpose~~ is to limit secondary system pressure to ≤ 110% of design pressure when passing ~~100%~~ 102% of design steam flow (100% plus 2% heat balance error). ~~This design basis is sufficient to cope with any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.~~ The MSSVs ensure that the design basis requirement is met for any abnormality or accident considered in the SAR.

edit — edit — 51

The events that ~~challenge the relieving capacity~~ may assume use of the MSSVs~~, and thus RCS pressure,~~ are those characterized as decreased heat removal events~~, and are presented in the FSAR, Section [15.2] (Ref. 3). Of these, the full power turbine trip coincident with a loss of condenser heat sink is the limiting AOO. For this event, the Condenser Circulating Water System is lost and, therefore, the Turbine Bypass Valves are not available to relieve Main Steam System pressure. Similarly,~~ MSSV ~~relief capacity is utilized in the FSAR for~~ use may be assumed during mitigation of the following events:

edit — 21 — edit

a. Loss of ~~main feedwater~~ load (SAR, Chapter 14 (Ref. 3));

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~b. Steam line break;~~

b ~~c~~. Steam generator tube rupture; and

C ~~d~~. Small break loss of coolant (Ref. 3). ~~Excessive heat removal due to feedwater system malfunction.~~

<INSERT B3.7-2A>

IN MODES 1 and 2, The MSSVs satisfy Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 5). 21 6

<INSERT B3.7-2E>

LCO

The MSSVs setpoints are ~~established~~ provided to prevent overpressurization as discussed in the Applicable Safety Analysis section of these Bases. The LCO requires all fourteen MSSVs (seven on each main steam line) to be OPERABLE to ensure compliance with the ASME Code following DBAs initiated at full power. Operation with less than a full the required complement of MSSVs requires a limitations on unit THERMAL POWER and adjustment of the Reactor Protection System (RPS) nuclear overpower trip setpoints. This effectively limits the Main Steam System steam flow while the MSSV relieving capacity is reduced due to valve inoperability. To be OPERABLE, lift setpoints must remain within limits, according to ~~Table 3.7.1-1 in the accompanying LCO~~ SR 3.7.1.1. 21 edit 1

<INSERT B3.7-2B>

The ~~OPERABILITY~~ safety function of the MSSVs is ~~defined as the ability~~ to open, ~~within the setpoint tolerances,~~ relieve steam generator overpressure, and reseat when pressure has been reduced. 21

~~The~~ OPERABILITY of the MSSVs ~~is determined by~~ requires periodic surveillance testing in accordance with the Inservice Testing Program.

<INSERT B3.7-2C>

The lift settings, ~~according to Table 3.7.1-1 in the accompanying LCO,~~ correspond to ambient conditions of the valve at nominal operating temperature and pressure. 21

This LCO provides assurance that the MSSVs will perform the design safety function. ~~to mitigate the consequences of accidents that could result in a challenge to the RCPB.~~ edit 1

<INSERT B3.7-2D>

APPLICABILITY

~~In MODE 1 above [18]% RTP, the number of MSSVs per steam generator required to be OPERABLE must be within the acceptable region, according to Figure 3.7.1-1 in the~~ 1

(continued)

<INSERT B3.7-2A>

The full power turbine trip coincident with a loss of condensate heat sink establishes the required MSSV relief capacity (Ref. 4).

<INSERT B3.7-2B>

The minimum number of OPERABLE MSSVs per steam generator for various power levels and the associated maximum allowable nuclear overpower trip setpoint are identified in Table 3.7.1-1.

<INSERT B3.7-2C>

With all MSSVs OPERABLE, at least one MSSV per steam generator is set at 1050 psig nominal, while the remaining MSSVs per steam generator are set at varied pressures up to and including 1100 psig nominal.

<INSERT B3.7-2D>

The LCO is modified by a Note that allows all but one MSSV on each main steam header to be gagged and the setpoints for the two (one on each header) OPERABLE MSSVs to be reset for the duration of hydrotesting in MODE 3. This is necessary to allow the hydrotest pressure to be attained.

<INSERT B3.7-2E>

In MODE 3, the MSSVs satisfy Criterion 4 of 10 CFR 50.36.

BASES

APPLICABILITY (continued)

~~accompanying LCO. Below [18% RTP]~~ In MODES 1, 2, and 3, the ~~only two~~ MSSVs are required to be OPERABLE ~~per steam generator~~ to prevent overpressurization of the main steam system. 1

In MODES 4 and 5, there is no credible transient requiring the MSSVs.

The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized. There is no requirement for the MSSVs to be OPERABLE in these MODES. edit

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1 and A.2

An alternative to restoring the inoperable MSSV(s) to OPERABLE status is to reduce power so that the available MSSV relieving capacity meets ASME Code requirements for the power level. Operation may continue, provided the ALLOWABLE THERMAL POWER and RPS nuclear overpower trip setpoint are reduced as required by Table 3.7.1-1. These values are based on ~~by the application of~~ the following formulas: 1

$$RP = \frac{Y}{Z} \times 100\%$$

and

$$SP = \frac{Y}{Z} \times W$$

where:

W = Nuclear overpower trip setpoint for four pump operation as specified in LCO 3.3.1, "Reactor Protection System (RPS)";

Y = Total OPERABLE MSSV relieving capacity per steam generator based on a summation of individual OPERABLE MSSV relief capacities per steam generator ~~[lb/hour]~~

the available capacity of each MSSV is 801,428 lbm/hour

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Z = Required relieving capacity per steam generator of ~~6,585,600 lb/hour~~; 5,610,000 lbm/hr

RP = Reduced power requirement (not to exceed RTP); and

SP = Nuclear overpower trip setpoint (not to exceed W).

~~These equations are graphically represented in Figure 3.7.1-1, in the accompanying LCO. Operation is restricted to the area below and to the right of line BCDE.~~ 1

~~The operator should limit the maximum steady state power level to some value slightly below this setpoint to avoid an inadvertent overpower trip.~~

The 4 hour Completion Time for Required Action A.1 is a reasonable time period to reduce power level and is based on the low probability of an event occurring during this period that would require activation of the MSSVs. An additional 32 ~~8~~ hours is allowed in Required Action A.2 to reduce the setpoints ~~in recognition of the difficulty of resetting of all channels of this trip function within a period of 4 hours~~. The Completion Time of 36 ~~12~~ hours for Required Action A.2 is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, on operating experience in resetting all channels of a protective function~~s~~ and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period. 2

B.1 and B.2

With one or more steam generators with less than two MSSVs OPERABLE, or if the Required Actions and ~~MSSVs inoperable, a verification by administrative means that at least two required MSSVs per steam generator are OPERABLE, with each valve from a different lift setting range, is performed.~~ 21

~~If the MSSVs cannot be restored to OPERABLE status in the~~ associated Completion Time are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of ~~each~~ MSSV lift setpoints in accordance with the Inservice Testing Program. The ~~ASME Code, Section XI (Ref. 4) requires that~~ safety and relief valve tests ~~be~~ are performed in accordance with ANSI/ASME OM-1-1987 (Ref. ~~5~~ 6) and include ~~According to Reference 5,~~ the following ~~tests are required~~ for MSSVs: edit

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity ~~device~~ on balanced valves. edit

The ANSI/ASME Standard requires the testing of all valves every 5 years, with a minimum of 20% of the valves tested every 24 months. Reference ~~7~~ 6 provides the activities and frequencies necessary to satisfy the requirements and ~~Table 3.7.1-1~~ allows ~~a~~ an as-found ± ~~[3]~~ 3% setpoint tolerance. ~~for OPERABILITY; however~~ Although not required by the IST Program, the valves are reset to ± 1% during the Surveillance to allow for drift. edit

1 0 (1)

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

3.7-01

(continued)

BASES (continued)

REFERENCES

1. ØSAR, Section ~~[5.2]~~. 10.3.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. ØSAR, ~~Section [15.2]~~. Chapter 14.
4. ~~4. ASME, Boiler and Pressure Vessel Code, Section XI.~~
6. ~~5.~~ ANSI/ASME OM-1-1987.

5. 10 CFR 50.36. (6)

4. Framatome Document 86-1266156-00, "ANO-1 Overpressure Protection," dated October 31, 1997. (21)

edit

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the main steam secondary side of the steam generators following a high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generator. edit

One MSIV is located in each main steam line outside of, but close to, ~~containment~~ the reactor building. The MSIVs are downstream from the main steam safety valves (MSSVs) and emergency feedwater pump turbine's steam supply to prevent their being isolated from the steam generators by MSIV closure. Closing the MSIVs isolates each steam generator from the other, and isolates the turbine, Turbine Bypass System, and other auxiliary steam supplies from the steam generators.

The MSIVs close on a main steam line isolation (MSLI) ~~Steam and Feedwater Rupture Control System~~ signal ~~generated by either low steam generator pressure or steam generator to feedwater differential pressure. The MSIVs fail closed on loss of control or actuation power.~~ ⟨INSERT B3.7-7A⟩ The MSIVs may also be actuated manually. 22

~~A description of the MSIVs is found in the FSAR, Section [10.3] (Ref. 1).~~ edit

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the ~~containment~~ analysis for the ~~large~~ steam line break (SLB) ~~inside containment~~, as discussed in the ~~F~~SAR, Section ~~[6.2]~~ 14.2 (Ref. ~~2~~ 1). ~~It is also influenced by the accident analysis of the SLB events presented in the FSAR, Section [15.4] (Ref. 3).~~ The EFIC System design precludes the blowdown of more than one steam generator, assuming a single active component failure ~~(i.e., the failure of one MSIV to close on demand)~~ as discussed in the SAR, Section 7.1.4 (Ref. 2). edit 22

~~The limiting case for the containment analysis is the SLB inside containment with a loss of offsite power following turbine trip and failure of the MSIV on the affected steam generator to close. At 100% RTP, the steam generator inventory and temperature are at their maximum, maximizing the mass and energy release to the containment.~~

(continued)

<INSERT B3.7-7A>

as described in LCO 3.3.11, "Emergency Feedwater Initiation and Control (EFIC) System Instrumentation." The EFIC System is designed to prevent the simultaneous blowdown of both steam generators.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~Due to reverse flow, failure of the MSIV to close contributes to the total release of the additional mass and energy in the steam headers downstream of the other MSIV. Other failures considered are the failure of a main feedwater isolation valve to close, and failure of an emergency diesel generator (EDG) to start.~~

~~The accident analysis compares several different SLB events against different acceptance criteria.~~ The Large SLB outside containment the reactor building upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The large SLB inside containment at full power is the limiting case for a post trip return to power. ~~The analysis includes scenarios with offsite power available and with a loss of offsite power following turbine trip.~~ With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown. ~~With a loss of offsite power, the response of mitigating systems, such as the High Pressure Injection (HPI) System pumps, is delayed. Significant single failures considered include failure of an MSIV to close, failure of an EDG, and failure of an HPI pump.~~ 22

The MSIVs serve only a Closing safety function in the event of an SLB and remain open during power operation. ~~These valves operate under the following situations:~~

~~a. An HELB, an SLB, or main feedwater line breaks (FWLBs), inside containment. In order to maximize the mass and energy release into the containment, the analysis assumes the MSIV in the affected steam generator remains open. For this scenario, steam is discharged into containment from both steam generators until closure of the MSIV in the intact steam generator occurs. After MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIV in the intact loop.~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. ~~An SLB outside of containment and upstream from the MSIVs is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break and limits the blowdown to a single steam generator.~~

c. ~~A break downstream of the MSIVs will be isolated by the closure of the MSIVs. Events such as increased steam flow through the turbine or the steam bypass valves will also terminate on closing the MSIVs.~~

d. ~~Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generator. In addition to minimizing radiological releases, this enables the operator to maintain the pressure of the steam generator with the ruptured tube below the MSIVs' setpoints, a necessary step toward isolating flow through the rupture.~~

e. ~~The MSIVs are also utilized during other events such as an FWLB.~~ (22)

In MODES 1 and 2, the MSIVs satisfy Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 3). (6)

<INSERT B 3.7-9A>

LCO

This LCO requires that ~~the MSIV in both steam lines be OPERABLE.~~ For an MSIV in each steam line to be OPERABLE. ~~The~~ MSIV~~s~~ ~~are~~ considered OPERABLE~~, when~~ the isolation time~~s~~ ~~are~~ must be within limits and the~~y~~ MSIV must close on an isolation actuation signal when required. (22) (8)

This LCO provides assurance that the MSIVs will perform their design safety function to ~~mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 limits (Ref. 4).~~ isolate an SLB (22)

APPLICABILITY

The MSIVs must be OPERABLE in ~~MODE 1 and in~~ MODES 1, 2, and 3 ~~with any MSIVs open,~~ when there is significant mass and energy in the RCS and steam generator~~s; therefore, the MSIVs must be OPERABLE or closed. When the MSIVs are closed, they are already performing the safety function.~~ to provide isolation of potential main steam line breaks (3)

(continued)

<**INSERT B3.7-9A**>

In MODE 3, the MSIVs satisfy Criterion 4 of 10 CFR 50.36.

BASES

APPLICABILITY (continued)

In MODE 4, the steam generator energy is low. Therefore, the MSIVs are not required to be OPERABLE.

In MODES 5 and 6, the steam generators are depressurized and ~~do not contain much energy because their temperature is below the boiling point of water; therefore,~~ the MSIVs are not required for isolation of potential main steam line ~~high energy secondary system pipe~~ breaks ~~in these MODES~~. edit

Although not credited, the 22

ACTIONS

A.1

With one or more MSIVs inoperable in MODE 1 or 2, action must be taken to restore the component to OPERABLE status within 24 ~~[8]~~ hours. Some repairs can be made to the MSIV with the unit hot. The 24 ~~[8]~~ hour Completion Time is reasonable, considering the probability of an accident that would require actuation of the MSIVs occurring during this time interval. ~~The~~ turbine throttle ~~stop~~ valves ~~are~~ may be available to provide ~~the required~~ isolation for some ~~the~~ postulated accidents. 4 edit

~~The [8] hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides an additional means for containment isolation.~~ 22

INSERT B3.7-10A edit

3.7-26

B.1

If the Required Action and associated Completion Time of Condition A are not met, ~~MSIV cannot be restored to OPERABLE status within [8] hours,~~ the unit must be placed in MODE ~~2~~ 3 ~~and the inoperable MSIV closed~~ within the next ~~6~~ 12 hours. The Completion Time~~s~~ ~~are~~ is reasonable, based on operating experience, to reach MODE ~~2~~ 3. 22 4

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV.

Since the MSIVs are required to be OPERABLE in MODE~~S 2 and~~ 3, the inoperable MSIV~~(s)~~ may either be restored to 4

(continued)

<**INSERT B3.7-10A**>

The main steam and feedwater systems do not provide a direct path from the reactor building atmosphere to the environment. Therefore, the Completion Time is reasonable, and provides for diagnosis and repair of many MSIV problems, thereby avoiding unnecessary shutdown.

BASES

ACTIONS

C.1 and C.2 (continued)

OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

INSERT B3.7-11A

~~The [8] hour Completion Time is consistent with that allowed in Condition A.~~ 4

Inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are closed, must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable~~, based on engineering judgment,~~ in view of MSIV status indications available in the control room, and other administrative controls, to ensure these valves are in the closed position. edit

D.1 and D.2

If the ~~MSIV cannot be restored to OPERABLE status or closed in the associated Completion Time,~~ Required Actions and associated Completion Times of Condition C are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed ~~in at least MODE 3 within 6 hours and~~ in MODE 4 within ~~12~~ 24 hours. The allowed Completion Time~~s~~ ~~are~~ is reasonable, based on operating experience, to reach the required unit conditions from MODE ~~2~~ 3 conditions in an orderly manner and without challenging unit systems. 22 4

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

This SR verifies that the ~~MSIV~~ isolation closure time of each MSIV is ~~≤ [6] seconds on an actual or simulated actuation signal~~ as specified in the Inservice Testing Program. ~~The MSIV closure~~ time is assumed in the accident and ~~containment~~ reactor building analyses. This Surveillance is normally performed ~~upon~~ prior to returning the unit to ~~operation~~ power operation, e.g., during MODE 3, following a refueling outage, because the MSIVs should not be tested at power since even a part stroke exercise increases the risk of a valve closure with the unit generating power. ~~As the MSIVs are not to be tested at power, they are exempt from~~ 8 edit edit edit

(continued)

<INSERT B3.7-11A>

The main steam and feedwater systems do not provide a direct path from the reactor building atmosphere to the environment. Therefore, the Completion Time is reasonable, and provides for diagnosis and repair of many MSIV problems, thereby avoiding unnecessary shutdown.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1 (continued)

the ASME Code, Section XI (Ref. 5) requirements during operation in MODES 1 and 2.

edit

The Frequency for this SR is in accordance with the Inservice Testing Program or [18] months]. The [18] month Frequency to demonstrate the valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

8

edit

normally

This test is conducted in MODE 3, with the unit at operating temperature and pressure, as discussed in the Reference 5 exercising requirements. This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was generated.

INSERT B3.7-12A

REFERENCES

edit

1. FSAR, Section [10.3].

1 2. FSAR, Section [6.2] 14.2.

2 3. FSAR, Section [15.4] 7.1.4.

3. 10 CFR 50.36.

6

4. 10 CFR 100.11.

4 5. ASME, Boiler and Pressure Vessel Code, Section XI.

<INSERT B3.7-12A>

SR 3.7.2.2

This SR verifies that each MSIV can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The Frequency of MSIV testing is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

This SR is modified by two Notes. The first Note allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was established.

SR 3.7.2.2 is also modified by a second Note which indicates that the automatic closure capability is not required to be met when SG pressure is < 750 psig. At < 750 psig, the main steam line isolation Function of EFIC may be disabled to prevent automatic actuation on low steam generator pressure during a unit shutdown.

3.7-02

MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves

B 3.7 PLANT SYSTEMS

B 3.7.3 ~~[~~Main Feedwater ~~Stop~~ Isolation Valves (MF~~S~~IVs) ~~, Main Feedwater Control Valves (MFCVs), and Associated Startup Feedwater Control Valves (SFCVs)]~~ 9

BASES

BACKGROUND

The main feedwater isolation valves (MFIVs) ~~for each steam generator consist of the MFSVs, MFCVs, and the SFCVs. The MFIVs~~ isolate main feedwater (MFW) flow to the secondary side of the steam generators ~~following a high energy line break (HELB)~~. ~~Closure of the MFIVs terminates flow to both steam generators, terminating the event for feedwater line breaks (FWLBs) occurring upstream of the MFIVs. The consequences of events occurring in the main steam lines or in the feedwater lines downstream of the MFIVs will be mitigated by their closure.~~ Closing the MFIVs ~~and associated bypass valves~~ effectively terminates the addition of feedwater to an affected steam generator, limiting the mass and energy release for steam line breaks (SLBs) or FWLBs inside containment and reducing the cooldown effects for SLBs. 9 23 edit

The MFIVs close on receipt of a main steam line isolation (MSLI) ~~Steam and Feedwater Rupture Control System (SFRCS)~~ signal ~~generated by either low steam generator pressure or steam generator/feedwater differential pressure~~ <INSERT B3.7-13A>. The MFIVs can also be closed manually. 23

~~The MFIVs and associated bypass valves close on receipt of a safety injection, low T_{avg} coincident with reactor trip or steam generator water level high high signal. They may also be actuated manually. In addition to the MFIVs and associated bypass valves, a check valve inside containment is available to isolate the feedwater line penetrating containment and to ensure that the consequences of events do not exceed the capacity of the containment heat removal systems.~~ 9

~~A description of the MFIVs is found in the FSAR, Section [10.4.7] (Ref. 1).~~

APPLICABLE SAFETY ANALYSES

The design basis of the MFIVs is established by the analysis for the ~~large~~ SLB as discussed in SAR Section 14.2.2.1 (Ref. 1). ~~It is also influenced by the accident analysis for the large FWLB. Closure of the MFIVs may also be relied on to terminate a steam break for core response~~ edit 23

(continued)

<INSERT B3.7-13A>

3.7-02 as described in LCO 3.3.11, "Emergency Feedwater Initiation and Control (EFIC) System Instrumentation." EFIC maintains the Low Load Feedwater Control Valves and Startup Feedwater Control Valves closed by sending a signal to the Rapid Feedwater Reduction (RFR) circuit of the Integrated Control System (ICS). The Main Feedwater Block Valves are independently closed by a signal from the Reactor Protection System (RPS) upon a reactor trip.

3.7-02

MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves

9

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~analysis and excess feedwater event upon the receipt of a steam generator water level" high signal.~~ 23

Failure of an MFIV to close following an SLB~~, FWLB, or excess feedwater event,~~ can result in additional mass and ~~energy~~ being delivered to the steam generators, contributing to cooldown. ~~This failure also results in additional mass and energy releases following an SLB or FWLB event.~~

The MFIVs satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 2). 6

LCO

<INSERT B3.7-14A>

In MODES 1 and 2, This LCO ensures that the MFIVs will isolate MFW flow to the steam generators following ~~a FWLB or~~ a main steam line break. ~~These valves will also isolate the nonsafety related portions from the safety related portions of the system.~~ edit 9

Two MFIVs ~~[Two] [MFSVs], [MFCVs], [or associated SFCVs]~~ are required to be OPERABLE. For an ~~The~~ MFIV ~~s~~ ~~are~~ considered OPERABLE ~~when~~ the isolation times must be ~~are~~ within limits and the valve must ~~y~~ close on an isolation actuation signal to be. 23

Failure to meet the LCO requirements can result in a more severe cooldown transient and in additional mass and energy being released to reactor building ~~containment~~ following an SLB ~~or FWLB~~ inside ~~containment~~ when required. ~~If the SFRCS on high steam generator level is relied on to terminate an excess feedwater flow event, failure to meet the LCO may result in the introduction of water into the main steam lines.~~ 23

APPLICABILITY

The MFIVs ~~[MFSVs], [MFCVs], [or associated SFCVs]~~ must be OPERABLE in MODES 1, 2, and 3 to ~~whenever there is significant mass and energy in the RCS and steam generators. This~~ ensure~~s~~ that, in the event of an SLB ~~or HELB, a single failure cannot result in the blowdown of more than one steam generator.~~ 9

~~In MODES 1, 2, and 3, the [MFSVs], [MFCVs], [or associated SFCVs] are required to be OPERABLE in order~~ to limit the amount of ~~available fluid~~ energy that could be added to reactor building ~~containment~~ in the case of a secondary system pipe break inside ~~containment. When the valves are closed, they are already performing their safety function.~~ 9 3

the amount of feedwater provided to the affected steam generator is limited. Their closure terminates normal feedwater flow to limit the overcooling transient and

(continued)

<INSERT B3.7-14A>

3.7-02 In MODE 3, the MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves satisfy Criterion 4 of 10 CFR 50.36.

With the exception of the MFIVs, the valves are non-Q and powered from non-vital sources. This is acceptable when crediting feedwater isolation during a SLB since off-site power is assumed to remain available during this event.

3.7-02

MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves

9

BASES

APPLICABILITY (continued)

In MODES 4, 5, and 6, steam generator energy is low. Therefore, the ~~[MFSVs], [MFCVs], [or associated SFCVs]~~ are not required for isolation of potential high energy secondary system pipe breaks in these MODES. 9

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each valve.

A.1 and A.2

With one MFIV ~~[MFSV]~~ in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within ~~[8 or~~ 72~~]~~ hours. When these valves are closed or isolated, they are performing their required safety function.

~~[For units with only one MFIV per feedwater line: The [8] hour Completion Time is reasonable to close the MFIV or its associated bypass valve which includes performing a controlled unit shutdown to MODE 2. The Completion Time is reasonable, based on operating experience, to reach MODE 2 from full power conditions with the MFIVs closed, in an orderly manner and without challenging unit systems.]~~ 9

The ~~[~~72~~]~~ hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths. The ~~[~~72~~]~~ hour Completion Time is reasonable ~~based on operating experience~~ to allow repairs and, if unsuccessful, to isolate the flowpath. edit

Inoperable MFIVs ~~[MFSVs]~~ that are closed or isolated, must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable ~~based on engineering judgment,~~ in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated. edit

(continued)

3.7-02

MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves

~~[MFSVs, MFCVs, and Associated SFCVs]~~ 9

BASES

ACTIONS (continued)

B.1 and B.2 Main Feedwater Block Valve

With one ~~[MFCV]~~ in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, ~~or to~~ close or isolate inoperable affected valves within ~~[8 or]~~ 72 hours. When these valves are closed or isolated, they are performing their required safety function.

~~[For units with only one MFIV per feedwater line: The [8] hour Completion Time is reasonable, based on operating experience, to close the MFIV or its associated bypass valve.]~~ 9

The ~~[~~72~~]~~ hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.

Main Feedwater Block valves

Inoperable ~~[MFCVs]~~ that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable~~, based on engineering judgment,~~ in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated. 9 edit

C.1 and C.2 Startup Feedwater Control Valve

<INSERT B3.7-16A>

With one ~~[SFCV]~~ in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, ~~or to~~ close or isolate inoperable affected valves within ~~[8 or]~~ 72 hours. When these valves are closed or isolated, they are performing their required safety function. 9

~~[For units with only one MFIV per feedwater line: The [8] hour Completion Time is reasonable, based on operating experience, to close the MFIV or its associated bypass valve.]~~ 9

The ~~[~~72~~]~~ hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE valves and the

(continued)

3.7-02 **<INSERT B3.7-16A>**

With one Low Load Feedwater Control Valve in one or more flow paths inoperable, action must be taken to restore the affected valves to OPERABLE status, or to close or isolate inoperable affected valves within 72 hours. When these valves are closed or isolated, they are performing their required safety function.

The 72 hour Completion Time takes into account the redundancy afforded by the remaining OPERABLE associated MFIV and the low probability of an event occurring during this time period that would require isolation of the MFW flow paths.

Inoperable Low Load Feedwater Control Valves that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated.

D.1 and D.2

3.7-02

MFIVs, Main Feedwater Block Valves Low Load Feedwater Control Valves and Startup Feedwater Control Valves

~~[MFSVs, MFCVs, and Associated SFCVs]~~ B 3.7.3 9

BASES

ACTIONS

D.1 and D.2 (continued)

low probability of an event occurring during this time period that would require isolation of the MFW flow paths.

Inoperable SFCVs Startup Feedwater Control Valves that are closed or isolated must be verified on a periodic basis that they are closed or isolated. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, ~~based on engineering judgment,~~ in view of valve status indications available in the control room, and other administrative controls, to ensure that these valves are closed or isolated. 9 edit

E.1

With two inoperable valves in the same flow path there may be no redundant system to operate automatically and perform the required safety function. Although the containment Main Feedwater flow can be isolated with the failure to two valves in parallel in the same flow path, the double failure can be an indication of a common mode failure in the valves of this flow path and as such is treated the same as a loss of the isolation capability of this flow path. Under these conditions, affected valves in each flow path must be restored to OPERABLE status, or the affected flow path isolated within 8 hours. The 8 hour Completion Time is reasonable, based on operating experience, to close the MFIV or otherwise isolate the affected flow path.

F.1 ~~and E.2~~

If the Required Actions and associated Completion Times are not met, ~~[MFSVs], [MFCVs], and [associated SFCVs] cannot be restored to OPERABLE status, or closed, or isolated within the associated Completion Time,~~ the unit must be in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. 23

(continued)

3.7-02

MFIVs, Main Feedwater Block Valves, Low Load Feedwater Control Valves and Startup Feedwater Control Valves

9

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.3.1

This SR verifies that the closure time of each ~~[MFSV], [MFCV], and [associated SFCV]~~ is ~~≤ 7 seconds on an actual or simulated actuation signal.~~ as specified in the Inservice Testing Program. 9

The MFIV ~~[MFSV], [MFCV], and [associated SFCV]~~ ~~closure~~ isolation time is assumed in the accident and ~~containment~~ reactor building analyses. This Surveillance is normally performed ~~upon~~ prior to returning the unit to power operation following a refueling outage. The ~~[MFSV],~~ MFIVs ~~[MFCV], and [associated SFCV] should~~ are not ~~be~~ tested at power e.g. during MODE 3, since even a part stroke exercise increases the risk of a valve closure with the unit generating power. ~~This is consistent with the ASME Code, Section XI (Ref. 2) requirements during operation in MODES 1 and 2.~~ 9 9 8

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR.

The Frequency for this SR is in accordance with the ~~[~~Inservice Testing Program~~, or [18] months]. The Frequency of [18] months for valve closure time is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency.~~

<INSERT B3.7-18A>

REFERENCES

1. ~~F~~SAR, Section ~~[10.4.7]~~. 14.2.2.1 edit
2. ~~ASME, Boiler and Pressure Vessel Code, Section XI.~~ 10 CFR 50.36. edit 6

<INSERT B3.7-18A>

3.7-02 This SR verifies that each MFIV, Main Feedwater Block Valve, Low Load Feedwater Control Valve and Startup Feedwater Control Valve can close on an actual or simulated actuation signal. This Surveillance is normally performed upon returning the unit to operation following a refueling outage.

The Frequency for this SR is every 18 months. The 18 month Frequency for testing is based on the refueling cycle. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, this Frequency is acceptable from a reliability standpoint.

This SR is modified by two Notes. The first Note allows entry into and operation in MODE 3 prior to performing the SR. This allows delaying testing until MODE 3 in order to establish conditions consistent with those under which the acceptance criterion was established.

SR 3.7.3.2 is also modified by a second Note which indicates that the automatic closure capability is not required to be met when the steam generator pressure is < 750 psig. At < 750 psig, the main steam line isolation Function of EFIC may be disabled to prevent automatic actuation on low steam generator pressure during a unit shutdown.

11

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Vent Valves (AVVs)

BASES

BACKGROUND

The AVVs provide a method for cooling the unit to decay heat removal (DHR) entry conditions, should the preferred heat sink via the Turbine Bypass System to the condenser not be available, as discussed in the FSAR, Section [10.3] (Ref. 1). This is done in conjunction with the Emergency Feedwater System, providing cooling water from the condensate storage tank (CST). The AVVs may also be required to meet the design cooldown rate during a normal cooldown when steam pressure drops too low for maintenance of a vacuum in the condenser to permit use of the Turbine Bypass System.

[The AVVs are provided with upstream block valves to permit their being tested at power, and to provide an alternate means of isolation.]

The AVVs are equipped with pneumatic controllers to permit control of the cooldown rate.

[The AVVs are provided with a pressurized gas supply of bottled nitrogen that, on loss of pressure in the normal instrument air supply, automatically supplies nitrogen to operate the AVVs. The nitrogen supply is sized to provide sufficient pressurized gas to operate the AVVs for the time required for Reactor Coolant System (RCS) cooldown to DHR entry conditions.]

A description of the AVVs is found in Reference 1.

APPLICABLE SAFETY ANALYSES

The design basis of the AVVs is established by the capability to cool the unit to MODE 3. The design rate of [75]°F per hour is applicable for both steam generators, each with one AVV. This rate is adequate to cool the unit to DHR entry conditions with only one AVV and one steam generator utilizing the cooling water supply available in the CST.

In the accident analysis presented in Reference 1, the AVVs are assumed to be used by the operator to cool down the unit

~~(continued)~~

11

BASES

APPLICABLE SAFETY ANALYSES (continued)

to MODE 3 for accidents accompanied by a loss of offsite power. Prior to operator actions to cool down the unit, the AVVs and the main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator's pressure and temperature below the design value. This is about 30 minutes following initiation of an event; however, this may be less for a steam generator tube rupture (SGTR) event. Some initiating events falling into this category are a main steam line break upstream of the main steam isolation valves, a feedwater line break, and an SGTR event (although the AVVs on the affected steam generator may still be available following an SGTR event).

For the recovery from an SGTR event, the operator is also required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured steam generator. The time required to terminate the primary to secondary break flow for an SGTR is more critical than the time required to cool down to DHR conditions for this event, and also for other accidents. Thus, the SGTR is the limiting event for the AVVs. The number of AVVs required to be OPERABLE to satisfy the SGTR accident analysis requirements depends upon the consideration of any single failure assumptions regarding the failure of one AVV to open on demand.

[The design must accommodate the single failure of one AVV to open on demand, thus each steam generator must have at least one AVV. The AVVs are equipped with manual block valves in the event an AVV spuriously fails open or fails to close during use.]

The AVVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

[Two] AVVs [lines per steam generator] are required to be OPERABLE. Failure to meet the LCO can result in the inability to cool the unit to DHR entry conditions following an event in which the condenser is unavailable for use with the Steam Bypass System.

An AVV is considered OPERABLE when it is capable of providing a controlled relief of the main steam flow, and is capable of fully opening and closing on demand.

(continued)

BASES (continued)

11

APPLICABILITY	In MODES 1, 2, and 3, and in MODE 4, when steam generator is being relied upon for heat removal, the AVVs are required to be OPERABLE. In MODES 5 and 6, an SGTR is not a credible event.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.4 does not apply.

With one AVV [line] inoperable, action must be taken to restore the inoperable AVV to OPERABLE status. The 7 day Completion Time allows for redundant capability afforded by the remaining OPERABLE AVV and a nonsafety grade backup in the Steam Bypass System and MSSVs.

[B.1

With more than one AVV [line] inoperable, action must be taken to restore [all but one] AVV [lines] to OPERABLE status. As the block valve can be closed to isolate an AVV, some repairs may be possible with the unit at power. The 24 hour Completion Time is reasonable to repair inoperable AVV [lines], based on the availability of the Steam Bypass System and MSSVs, and the low probability of an event occurring during this period that would require the AVV [lines].]

C.1 and C.2

If the AVV [lines] cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 18 hours, without reliance upon the steam generator for heat removal. 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.

(continued)

11

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a controlled cooldown of the RCS, the AVVs must be able to be opened either remotely or locally and throttled through their full range. This SR ensures that the AVVs are tested through a full control cycle at least once per fuel cycle. Performance of inservice testing or use of an AVV during a unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

[SR 3.7.4.2

The function of the block valve is to isolate a failed open AVV. Cycling the block valve closed and open demonstrates its ability to perform this function. Performance of inservice testing or use of the block valve during unit cooldown may satisfy this requirement. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.]

REFERENCES

1. FSAR, Section [10.3].

Generic term. change.
CST → QCST

edit

B 3.7 PLANT SYSTEMS

B 3.7.5 Emergency Feedwater (EFW) System

BASES

BACKGROUND

The EFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System (RCS) upon the loss of normal feedwater supply. The safety related EFW pumps take suction ~~through separate and independent suction lines~~ from the Q condensate storage tank (CST) (LCO 3.7.6, "Q Condensate Storage Tank (CST)"), and pump to the steam generator secondary side through the EFW nozzles. The ~~steam generators function as a heat sink for~~ core decay heat. ~~The~~ heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)"), or atmospheric ~~vent~~ dump valves (~~AVVs~~) ~~(LCO 3.7.4, "Atmospheric Vent Valves (AVVs)")~~. If ~~the main condenser is available, steam may be released via the Turbine Bypass System and recirculated to the CST~~ ADVs, Turbine Bypass valves.

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edit

~~The following system description is provided as an example. Actual system description should be provided by the specific unit.~~ The EFW System ~~consists of~~ includes ~~two~~ one turbine driven EFW pump~~s~~, ~~each of which provides a nominal 100% capacity,~~ and one ~~non~~safety grade motor driven EFW pump. The ~~steam~~ turbine driven EFW pump~~s~~ receives steam from either of the two main steam headers, upstream of the main steam isolation valves (MSIVs). The EFW System supplies a common header capable of feeding either or both steam generators. ~~The 100% capacity~~ Either pump is sufficient to remove decay heat and cool the unit to decay heat removal (DHR) entry conditions. The EFW System ~~normally~~ initially receives a supply of water from the CST. ~~A~~ The assured safety grade source of water is ~~also~~ supplied by the Service Water System (SWS). ~~Automatic~~ valves on the supply piping are manually opened ~~open on low pressure in the supply piping~~ to transfer the water supply from the CST to the SWS. ~~A third source of~~ water can be supplied from other sources by manually aligning ~~the fire protection header to the EFW pump suction.~~ nonsafety grade condensate storage tanks to the EFW pump suction. Thus, ~~the requirement for~~ diversity in motive power sources is provided for the EFW System. ~~are met.~~

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edit

The EFW System is capable of supplying feedwater to the steam generators, if required, during normal unit startup and shutdown evolutions, and during hot standby conditions. However, EFW does not provide a normal source of feedwater during these conditions. The normal supplement to the main feedwater system under these conditions is provided by the auxiliary feedwater system.

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(continued)

BASES

BACKGROUND (continued)

~~The EFW System is designed to supply sufficient water to cool the unit to DHR entry conditions with steam being released through the ADVs or condenser.~~

The EFW actuates automatically ~~on~~ low steam generator level, low steam generator pressure, or loss of four reactor coolant pumps (e.g., on loss of main feedwater pumps, 8).

<INSERT B3.7-24A>

The EFW System is discussed in the FSAR, Sections ~~[9.2.7]~~ 7.1.4 and ~~[9.2.8]~~ 10.4.8 (Refs. 1 and 2, respectively).

25

APPLICABLE SAFETY ANALYSES

The EFW System ~~mitigates the consequences of any event with a loss of normal feedwater~~ is sized to prevent exceeding 110% RCS design pressure for a specified loss of feedwater scenario (Ref. 3).

25

The design basis of the EFW System is to supply water to the steam generators to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures corresponding to the lowest steam generator safety valve set pressure ~~plus 3%~~.

25

~~In addition, the EFW System must supply enough makeup water to replace steam generator secondary inventory being lost as steam as the unit cools to MODE 4 conditions. Sufficient EFW flow must also be available to account for flow losses such as pump recirculation and line breaks.~~

~~The limiting Design Basis Accidents (DBAs) and transients for the EFW System are as follows:~~

~~a. Feedwater line break (FWLB); and~~

~~b. Loss of main feedwater.~~

~~In addition, the minimum available EFW flow and system characteristics are serious considerations in the analysis of a small break loss of coolant accident.~~

25

The EFW System design is such that it can perform its function with only one EFW train available ~~following a loss of the turbine driven main feedwater pumps or an FWLB, combined with a loss of normal or reserve electric power~~.

IN MODES 1 and 2, the EFW System satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 4). In MODE 3 and MODE 4 when steam generator(s) are relied upon for heat removal, the EFW System satisfies Criterion 4 of 10CFR50.36.

6

(continued)

<INSERT B3.7-24A>

as described in LCO 3.3.11, "Emergency Feedwater Initiation and Control (EFIC) System Instrumentation."

BASES (continued)

LCO

This LCO provides assurance that the EFW System will perform its design ~~safety~~ function to mitigate the consequences of ~~accidents~~ events that could result in overpressurization of the reactor coolant pressure boundary. ~~[Three]~~ Two independent ~~EFW pumps, in two diverse~~ trains are required to be OPERABLE to ensure the availability of residual heat removal capability. ~~for all events accompanied by a loss of offsite power and a single failure.~~ ~~[This is accomplished by powering two pumps by steam driven turbines supplied with steam from a source not isolated by the closure of the MSIVs, and one pump from a power source that, in the event of loss of offsite power is supplied by the emergency diesel generator.]~~

12

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edit

For both EFW trains ~~The EFW System~~ to be ~~is~~ considered one ~~to be~~ OPERABLE ~~when~~ the components and flow paths required to ~~provide~~ be capable of providing EFW flow to both ~~the~~ steam generators ~~are OPERABLE~~. This requires that the ~~[two]~~ one turbine driven EFW pump ~~(s)~~ be OPERABLE with ~~redundant~~ two steam supplies from each of the main steam lines upstream of the MSIVs) and capable of supplying EFW flow to ~~either of~~ the ~~two~~ steam generators. The ~~[non~~safety grade~~]~~ motor driven EFW pump ~~(s)~~ is ~~and their associated flow path(s) to the EFW System [are]~~ also required to be OPERABLE and capable of supplying EFW flow to the steam generators. The piping, valves, instrumentation, and controls in the required flow paths ~~shall~~ must also be OPERABLE. The primary and secondary sources of water to the EFW System are required to be OPERABLE. The associated flow paths from the EFW System primary and secondary sources of water to ~~all~~ both EFW pumps also are required to be OPERABLE.

edit

edit

edit

25

The LCO is modified by a Note indicating that only one EFW train, which includes ~~a~~ the motor driven EFW pump, is required in MODE 4. This is because of reduced heat removal requirement, the short duration of MODE 4 in which feedwater is required, and the insufficient steam supply available in MODE 4 to power the turbine driven EFW pump.

edit

APPLICABILITY

In MODES 1, 2, and 3, the EFW System is required to be OPERABLE ~~and~~ in order to function in the event that the main feedwater is lost. In addition, the EFW System is required to supply enough makeup water to replace the steam generator secondary inventory lost as the unit cools to MODE 4 conditions.

edit

(continued)

are relied upon for decay heat removal since EFW is the safety related source of feedwater to the steam generators.

BASES

must be OPERABLE when

APPLICABILITY (continued)

In MODE 4, ~~with RCS temperature above [212]°F~~, the EFW System ~~may be used for heat removal via~~ the steam generators. In MODE 4, the steam generators are used for heat removal until the DHR System is in operation. normally 25

In MODES 5 and 6, the steam generators are not used for DHR and the EFW System is not required.

ACTIONS

A.1

or if the turbine driven EFW pump is inoperable while in MODE 3 immediately following refueling,

With one of the two steam supplies to the turbine driven EFW pump inoperable, action must be taken to restore the ~~steam supply~~ to OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons: 55

affected equipment

<INSERT B 3.7-26A>

~~a. The redundant OPERABLE steam supply to the turbine driven EFW pump(s);~~

~~b. The availability of the redundant OPERABLE motor driven EFW pump; and~~

~~c. The low probability of an event occurring that would require the inoperable steam supply to the turbine driven EFW pump(s).~~ 55

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of ~~Conditions~~ to be inoperable during any continuous failure to meet this LCO. edit

required EFW components

on the

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 7~~2 hours~~ days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met. edit edit 55

<INSERT B 3.7-26B>

ANO-292

ANO-292

B.1

When one of the required EFW trains (pump or flow path) is inoperable, action must be taken to restore the train to

(continued)

<INSERT B3.7-26A>

ANO-292

a. For the inoperability of a steam supply to the turbine driven EFW pump, the 7 day Completion Time is reasonable since there is a redundant steam line for the turbine driven pump.

b. For the inoperability of the turbine driven EFW pump while in MODE 3 immediately subsequent to a refueling, the 7 day Completion Time is reasonable due to the minimal decay heat levels in this situation.

c. For both the inoperability of a steam supply line to the turbine driven pump and an inoperable turbine driven EFW pump while in MODE 3 immediately following a refueling, the 7 day Completion Time is reasonable due to the availability of the redundant OPERABLE motor driven EFW pump; and due to the low probability of an event requiring the use of the turbine driven EFW pump.

<INSERT B3.7-26B>

ANO-292

Condition A is modified by a Note which limits the applicability of the Condition to when the unit has not entered MODE 2 following a refueling. Condition A allows one EFW train to be inoperable for 7 days vice the 72 hour Completion Time in Condition B. This longer Completion Time is based on the reduced decay heat following refueling and prior to the reactor being critical.

BASES

ACTIONS

B.1 (continued)

OPERABLE status within 72 hours. This Condition includes the loss of two steam supply lines to ~~one of~~ the turbine driven EFW pump~~s~~. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the EFW System, time needed for repairs, and the low probability of ~~a DBA~~ an event requiring EFW occurring during this time period. The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of ~~Conditions~~ required EFW components to be inoperable during any continuous failure to meet this LCO. edit

The 10 day Completion Time provides a limitation on the time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which Conditions A and B are entered concurrently. The AND connector between 72 hours and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met. edit

C.1 and C.2

With the ~~When either~~ Required Action ~~A.1 or Required Action B.1 cannot be completed within the required Completion Time, [or when two EFW trains are inoperable in MODE 1, 2, or 3,]~~ and associated Completion Time of Condition A or B not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 4 within ~~[~~18~~]~~ hours. 25

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.

~~In MODE 4, with two EFW trains inoperable, operation is allowed to continue because only one motor driven EFW train is required in accordance with the Note that modifies the LCO. Although not required, the unit may continue to cool down and initiate DHR.~~ 12

edit

edit

(continued)

BASES

ACTIONS (continued)

D.1

Required Action D.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until at least one EFW train is restored to OPERABLE status.

With ~~all~~ both EFW trains inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with nonsafety grade equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore at least one EFW train to OPERABLE status. LCO 3.0.3 is not applicable, as it could force the unit~~s~~ into a less safe condition. edit

E.1

In MODE 4, either the steam generator the required loops or the DHR loops can be used to provide heat removal, which is addressed in LCO 3.4.6, "RCS Loops" MODE 4." With ~~one~~ EFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status. edit

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the EFW water and steam supply flow paths provides assurance that the proper flow paths exist for EFW operation. Correct alignment for automatic valves may be other than the post-accident position provided the valve is otherwise OPERABLE. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since those valves are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. 25

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1 (continued)

The 31 day Frequency is based on ~~engineering judgment, is consistent with~~ the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

Verifying that each EFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that EFW pump performance has not degraded below the established acceptance criteria during the cycle. Flow and differential head are ~~normal tests~~ indicators of pump performance required by Section XI of the ASME Code (Ref. 3). Because it is undesirable to introduce cold EFW into the steam generators while they are operating, this test ~~is~~ may be performed on ~~recirculation flow.~~ a test flow path. edit edit

This test ~~confirms one point on the pump design curve and~~ is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing in the ASME Code, Section XI (Ref. 3), ~~at 3 month intervals,~~ satisfies this requirement. ~~The [31] day Frequency on a STAGGERED TEST BASIS results in testing each pump once every 3 months, as required by Reference 3.~~ edit edit 13

This SR is modified by a Note indicating that the SR ~~should~~ may be deferred until suitable test conditions are established. This deferral is required because there ~~is~~ may be insufficient steam pressure to perform the test. edit edit

SR 3.7.5.3

This SR verifies that EFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates ~~a Steam and~~ an Emergency Feedwater ~~Rupture~~ Initiation and Control (EFIC) System ~~(SFRCS)~~ signal by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. Each automatic valve is also verified to be capable of manual operation by over-riding the actuation 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 25

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.3 (continued)

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 operating experience and design reliability of the equipment. ~~This SR is modified by a Note that states the SR is not required in MODE 4. In MODE 4, the required AFW train is already aligned and operating.~~ This SR is modified by ~~[a]~~ [two] Notes. ~~[Note 1 indicates that the SR be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.]~~ [The] Note [2] states that the SR is not required to be met in MODE 4 when the steam generator is being relied upon for heat removal. ~~[In MODE 4, the required pump is already operating and the autostart function is not required.]~~ [In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required EFW pump.]

edit

edit

14

AND-289

to be met

which

when the steam generator is being relied upon for heat removal.

SR 3.7.5.4

This SR verifies that ~~the turbine driven~~ each EFW pump starts in the event of any accident or transient that generates an ~~SFRCS~~ EFIC signal ~~by demonstrating that each turbine driven EFW pump starts automatically on an actual or simulated actuation~~ signal. ~~These pumps are not required in MODE 4.~~ 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. This SR is modified by ~~[a]~~ [two] Notes. ~~[Note 1 indicates that the SR be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.]~~ [The] Note [2] states that the SR is not required to be met in MODE 4 which ~~[In MODE 4, the required pump is already operating and the autostart function is not required.]~~ [In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required EFW pump.]

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edit

edit

14

AND-289

~~Reviewer's Note: Some plants may not routinely use the AFW for heat removal in MODE 4. The second justification is provided for plants that use a startup feedwater pump rather than AFW for startup and shutdown.~~

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(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.5.5

any combination of MODE 5 or 6 or defueled. (47)

This SR ensures that the EFW System is properly aligned by verifying the position of manual valves in the flow paths to each steam generator prior to entering MODE 2 after more than 30 days in ~~MODE 5 or 6~~. OPERABILITY of EFW flow paths must be demonstrated before sufficient core heat is generated that would require the operation of the EFW System during a subsequent shutdown. The Frequency is reasonable~~, based on engineering judgement,~~ edit (15) in view of other administrative controls, such as SR 3.7.5.1, to ensure that the flow paths are OPERABLE. To further ensure EFW System alignment, flow path OPERABILITY is verified, following extended outages to determine no misalignment of manual valves has occurred. This SR ensures that the flow path from the Q CST to the steam generator is properly aligned. ~~(This SR is not required by those units that use EFW for normal startup and shutdown.)~~ (46)

<INSERT B 3.7-31A>

~~SR 3.7.5.6 and SR 3.7.5.7~~

~~For this facility, the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION for the EFW pump suction pressure interlocks are as follows:~~ (16)

REFERENCES

1. ~~F~~SAR, Section ~~[9.2.7]~~ 7.1.4. edit
2. ~~F~~SAR, Section ~~[9.2.8]~~ 10.4.8. edit
3. NRC Letter dated January 12, 1981, (1CNA018103). (25)
4. 10 CFR 50.36. (6)
5. ~~3.~~ ASME, Boiler and Pressure Vessel Code, Section XI. edit

<INSERT B3.7-31A>

SR 3.7.5.6

This SR ensures that the EFW flowpath to each steam generator is open and that water reaches the steam generators from the EFW System. This test is performed during shutdown to minimize thermal cycles to the emergency feedwater nozzles on the steam generator due to the lower temperature of the emergency feedwater. The motor-driven EFW pump is specified because of its availability at the low steam generator pressure conditions that exist in the shutdown condition. 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.

Generic term change, CST → QCST. — every use —

edit

B 3.7 PLANT SYSTEMS

B 3.7.6 Q Condensate Storage Tank ~~(CST)~~ (QCST)

BASES

condensate storage tank (QCST)

BACKGROUND

The ~~CST~~ provides a ~~safety grade~~ source of demineralized water to the steam generators for removing decay and sensible heat from the Reactor Coolant System (RCS). The CST provides ~~a passive flow~~ the preferred source of water ~~by gravity~~ to the Emergency Feedwater (EFW) System (LCO 3.7.5, "Emergency Feedwater (EFW) System"). ~~The steam produced is released to the atmosphere by the main steam safety valves (MSSVs) or the atmospheric vent valves.~~ 26

edit

~~When the main steam isolation valves are open, the preferred means of heat removal is to discharge to the condenser by the nonsafety grade path of the turbine bypass valves. The condensed steam is returned to the CST by the condensate pump. This has the advantage of conserving condensate while minimizing releases to the environment.~~ 26

Because the CST is ~~a principal component in removing residual heat from the RCS~~ the normally aligned source to EFW, it is designed to withstand earthquakes and other natural phenomena, ~~as well as~~ and a portion is protected from missiles that might be generated by natural phenomena. The CST is designed ~~to~~ as Seismic Category I to ensure availability of the ~~feedwater~~ initial EFW supply. Feedwater is also available from ~~an~~ alternate source~~(s)~~s. 26

A description of the CST is found in the ~~F~~SAR, Section ~~[9.2.6]~~ 10.4.8 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The CST provides the initial source of cooling water to remove any decay heat and cool down the unit following ~~all~~ events ~~in the accident analysis, as discussed in the FSAR, Chapters [6] and [15] (Refs. 2 and 3, respectively).~~ with a loss of normal feedwater. ~~For anticipated operational occurrences and accidents that do not affect the OPERABILITY of the steam generators, the analysis assumption is generally 30 minutes at MODE 3, steaming through the MSSVs, followed by a cooldown to decay heat removal (DHR) entry conditions at the design cooldown rate.~~

~~The limiting event for the condensate volume is the large feedwater line break coincident with a loss of offsite~~ 26

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~power. Single failures that also affect this event include the following:~~

~~a. Failure of the diesel generator powering the motor driven EFW pump to the unaffected steam generator (requiring additional steam to drive the remaining EFW pump turbine); and~~

~~b. Failure of the steam driven EFW pump (requiring a longer time for cooldown using only one motor driven EFW pump).~~

~~These are not usually the limiting failures in terms of consequences for these events.~~

<INSERT B3.7-33A>

26

The CST satisfies Criterion ~~2~~ 4 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 2).

6

LCO

~~To satisfy accident analysis assumptions, the [two] CSTs must contain sufficient cooling water to remove decay heat for 13 hours following a reactor trip from 102% RTP and then to cool down the RCS to DHR System entry conditions, assuming a coincident loss of offsite power and most adverse single failure. While so doing, the CSTs must retain sufficient water to ensure adequate net positive suction head for the EFW pump(s) during the cooldown, to account for any losses from the steam driven EFW pump turbine, as well as losses incurred before isolating EFW to a broken line.~~

~~The level required is equivalent to a usable volume of [250,000] gallons, which is based on holding the unit in MODE 3 for 13 hours, followed by a cooldown to DHR System entry conditions.~~

~~The OPERABILITY of the CST is determined by maintaining the tank level at or above the minimum required level.~~

<INSERT B3.7-33B>

26

APPLICABILITY

In MODES 1, 2, 3, and in MODE 4, when (a) steam generator is being relied upon for heat removal, the CST is required to be OPERABLE.

edit

In MODE 4 when a steam generator is not being relied upon for heat removal and in MODES 5 and 6, the CST is not required because the EFW System is not required.

26

(continued)

<INSERT B3.7-33A>

A portion of the QCST (T-41B) is protected from tornado generated missiles. The protected volume is sufficient to provide a thirty minute supply of water which is adequate to allow manual operator action, if required, to transfer suction of the EFW pumps to service water.

<INSERT B3.7-33B>

The OPERABILITY of the QCST with the minimum required water volume ensures that sufficient water is available to support EFW operation on both units for at least 30 minutes. This provides adequate time fir the operators to manually switch the EFW suction alignment to the Service Water System (SWS), if required. The SWS provides the assured long-term source of cooling water. The required volume considers that the EFW suctions of both units may be aligned to the QCST simultaneously.

The required minimum usable volume includes an allowance for losses due to Unit 2 recirculation line flow. The required volume of 32,300 gallons is equivalent to a tank level of 3 feet 10 inches. This parameter value does not include allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

The tank has sufficient capacity to support more than four hours of cooling in MODE 3 or MODE 4 conditions for both units. This capability is not considered to be a safety related design function and is not controlled by the Technical Specifications.

BASES (continued)

ACTIONS

A.1 and A.2

As an alternative to unit shutdown, the OPERABILITY of the backup water supply should be verified within 4 hours and once every ~~24~~ 12 hours thereafter. The OPERABILITY of the backup feedwater supply must include verification, by administrative means, of the OPERABILITY of flow paths from the backup supply to the EFW pumps and availability of the required volume of water in the backup supply. The CST must be restored to OPERABLE status within 7 days because the backup supply may be performing this function in addition to its normal functions. The 4 hour Completion Time is reasonable, based on operating experience, to verify the OPERABILITY of the backup water supply. Additionally, verifying the backup water supply every 12 hours is adequate to ensure the backup water supply continues to be available. The 7 day Completion Time is reasonable, based on an OPERABLE backup water supply being available, and the low probability of an event occurring during this time period, requiring the use of the water from the Q CST~~(s)~~.

48

48

edit

B.1 and B.2

If the Required Action and ~~CST cannot be restored to OPERABLE status in the~~ associated Completion Time s are not met, the unit must be placed in a MODE in which the LCO does not apply, with the DHR System in operation. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance on steam generators for heat removal, within ~~[18]~~ 24 hours~~]. This allows an additional 6 hours for the DHR System to be placed in service after entering MODE 4.~~

edit

56

26

ANO-293

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 3.7.6.1

This SR verifies that the CST~~(s)~~ contains the required volume of cooling water. The 12 hour Frequency is based on operating experience and the need for operator awareness of unit evolutions that may affect the CST inventory between checks. The 12 hour Frequency is considered adequate in view of other indications in the control room, including

edit

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1 (continued)

alarms, to alert the operator to abnormal deviations in CST levels.

REFERENCES

1. ~~F~~SAR, Section ~~[9.2.6]~~ 10.4.8. edit
2. ~~FSAR, Chapter [6].~~ 10 CFR 50.36. 6 edit
3. ~~FSAR, Chapter [15].~~

17

B 3.7 PLANT SYSTEMS

B 3.7.7 Component Cooling Water (CCW) System

BASES

BACKGROUND

The CCW System provides a heat sink for the removal of process and operating heat from safety related components during a Design Basis Accident (DBA) or transient. During normal operation, the CCW System also provides this function for various nonessential components, as well as the spent fuel pool. The CCW System serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the Service Water System, and thus to the environment.

A typical CCW System is arranged as two independent full capacity cooling loops, and has isolatable nonsafety related components. Each safety related train includes a full capacity pump, surge tank, heat exchanger, piping, valves, and instrumentation. Each safety related train is powered from a separate bus. A surge tank in the system provides sufficient net positive suction head for each pump and isolation of nonessential components on a low tank level signal. The pump in each train is automatically started on receipt of a safety feature actuation signal, and all nonessential components are isolated.

Additional information on the design and operation of the CCW System, along with a list of the components served, is presented in the FSAR, Section [9.2.2] (Ref. 1). The principal safety related function of the CCW System is the removal of decay heat from the reactor via the [decay heat removal (DHR) heat exchanger]. This may utilize the DHR System during a normal or post accident cooldown and shutdown, or during the recirculation phase following a loss of coolant accident.

APPLICABLE SAFETY ANALYSES

The design basis of the CCW System is to provide cooling water to the Emergency Core Cooling System and emergency diesel generators (EDGs) during DBA conditions. The CCW System also supplies cooling water to EDGs during a loss of offsite power.

(continued)

17

BASES

APPLICABLE SAFETY ANALYSES (continued)	The CCW System is designed to perform its function with a single failure of any active component assuming a loss of offsite power.
	The CCW System also functions to cool the unit from [DHR] entry conditions (T_{cold} < [350]°F) to MODE 5 (T_{cold} < [200]°F) during normal and post accident operations. The time required to cool from [350]°F to [200]°F is a function of the number of CCW and [DHR] trains operating. One CCW train is sufficient to remove decay heat during subsequent operations with T_{cold} < [200]°F.
	The CCW System satisfies Criterion 3 of the NRC Policy Statement.
LCO	The CCW trains are independent of each other to the degree that each has separate controls and power supplies and the operation of one train does not depend on the other. In the event of a DBA, one train of CCW is required to provide the minimum heat removal capability assumed in the safety analysis for systems to which it supplies cooling water. To ensure this is met, two CCW trains must be OPERABLE. At least one CCW train will operate assuming the worst case single active failure occurs coincident with loss of offsite power.
	A CCW train is considered OPERABLE when: a. It has an OPERABLE pump and associated surge tank; and b. The associated piping, valves, heat exchanger, and instrumentation and controls required to perform the safety related function are OPERABLE.
	The isolation of CCW from other components or systems not required for safety may render these components or systems inoperable, but does not affect the OPERABILITY of the CCW System.
APPLICABILITY	In MODES 1, 2, 3, and 4, the CCW System is a normally operating system that must be prepared to perform its post

(continued)

BASES

APPLICABILITY (continued)

accident safety functions, primarily Reactor Coolant System heat removal, by cooling the DHR heat exchanger.

In MODES 5 and 6, the OPERABILITY requirements of the CCW System are determined by the systems it supports.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources" Operating," and LCO 3.4.6, "RCS Loops" MODE 4," should be entered if an inoperable CCW train results in an inoperable EDG or DHR loop. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

If one CCW train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE CCW train is adequate to perform the heat removal function. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this period.

B.1 and B.2

If the CCW train cannot be restored to OPERABLE status in the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours.

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

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

This SR is modified by a Note indicating that the isolation of the CCW flow to individual components may render those

(continued)

17

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1 (continued)

components inoperable, but does not affect the OPERABILITY of the CCW System.

Verifying the correct alignment for manual, power operated, and automatic valves in the CCW flow path provides assurance that the proper flow paths exist for CCW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves which cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in their correct position.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.7.2

This SR verifies proper automatic operation of the CCW valves on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated as part of routine testing during normal operation. 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 outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.7.3

This SR verifies proper automatic operation of the CCW pumps on an actual or simulated actuation signal. The CCW System is a normally operating system that cannot be fully actuated

(continued)

17

17

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.7.3 (continued)

as part of routine testing during normal operation. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section [9.2.2].

37-30

B 3.7 PLANT SYSTEMS

B 3.7.~~8~~ 7 Service Water System (SWS)

BASES

BACKGROUND

The SWS provides a heat sink for the removal of process and operating heat from safety related components during a transient or Design Basis Accident (DBA) ~~or transient~~. During normal operation and normal shutdown, the SWS also provides this function for various safety related and nonsafety related components. The safety related ~~position~~ portion is covered by this LCO. edit

The ~~An~~ SWS consists of two ~~separate~~ independent but interconnected, 100% capacity safety related cooling water ~~trains~~ loops. Each ~~train~~ loop consists of a ~~100% capacity~~ pump, ~~one component cooling water (CCW) heat exchanger,~~ piping, valving, and instrumentation. The pumps, sluice gates, and valves are remote manually aligned, ~~except in the unlikely event of a loss of coolant accident (LOCA) as The pumps are automatically started upon receipt of a safety feature actuation signal, and all~~ essential valves are aligned to their post accident positions upon receipt of an engineered safeguards actuation signal. The SWS ~~also~~ provides cooling directly to ~~the Control Room Emergency Ventilation System water cooled condensing unit, the Emergency Core Cooling System (ECCS) pump room coolers, containment air cooler, and turbine driven cooling water systems~~ the assured safety related required plant equipment. The system ~~provides cooling and~~ is also ~~a~~ source of water to the ~~ECCS pump and the~~ emergency feedwater pumps, and can also provide a source of makeup water to the emergency cooling ~~tower~~ pond, and to the spent fuel pool.

Three 100% capacity pumps are provided to supply the two trains.

27

<INSERT B3.7-41A>

Additional information about the design and operation of the SWS, along with a list of the components served, is presented in the ~~F~~SAR, Section ~~9.2.1~~ 9.3 (Ref. 1). The principal safety related function of the SWS is the ~~removal~~ transfer of ~~decay~~ heat from the reactor ~~via the CCW System~~ and safety related components to the heat sink.

APPLICABLE SAFETY ANALYSES

The ~~design basis~~ primary safety function of the SWS is for one SWS train, in conjunction with the ~~CCW System and a 100% capacity containment~~ low Pressure Injection System and the Reactor Building cooling system, (~~containment~~ reactor building spray, ~~containment~~ air coolers, or a combination) to remove core decay heat following a design basis LOCA, as discussed in the ~~F~~SAR, Sections ~~6.2.1 (Ref. 2).~~ 6.2 and 6.3 (Refs. 2 and 3, respectively). ~~This provides for a gradual reduction in the temperature of this fluid, as it is~~

(continued)

<INSERT B3.7-41A>

The requirements of the service water system for cooling water are more severe during normal operation (at full power) than under accident conditions. Normal operation requires at least two of the three service water pumps, and the pumps in operation are periodically rotated. Normal operation also includes the addition of a biocide during the reactor building emergency cooler surveillance, when the water temperature is between 60°F and 80°F, to prevent biological fouling of the coolers. This water temperature range provides conditions under which Asian clams can spawn and produce larvae which could pass through service water system strainers.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~supplied to the Reactor Coolant System (RCS) by the safety injection pumps.~~ edit

The SWS is designed to perform its function with a single failure of any active component, assuming loss of offsite power.

The SWS~~, in conjunction with the CCW System,~~ also cools the unit from Decay Heat Removal (DHR) System~~, as discussed in the FSAR, Section [6.3], (Ref. 4),~~ entry conditions to MODE 5 during normal and post accident operation. 9.5 The time required for this evolution is a function of the number of ~~CCW and~~ DHR System ~~trains~~ loops that are operating. ~~One SWS train is sufficient to remove decay heat during subsequent operations in MODES 5 and 6. This assumes a maximum SWS temperature of [85]°F occurring simultaneously with maximum heat loads on the system.~~ 27

The SWS is also required ~~when needed~~ to ~~support CCW in the removal of~~ transfer heat from the ~~emergency~~ diesel generators (EDGs)~~, or reactor auxiliaries~~. edit

In MODES 1 and 2 ~~The SWS satisfies~~ Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 5). In MODES 3 and 4, the SWS satisfies Criterion 4 of 10 CFR 50.36. 6

LCO

Two SWS ~~trains~~ loops are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst case single active failure occurs coincident with the loss of offsite power. edit

For ~~An~~ an SWS ~~train~~ loop ~~is~~ to be considered OPERABLE, ~~when:~~ it must have: edit

a. ~~It has an~~ One OPERABLE pump; and

b. The associated piping, valves, ~~heat exchanger,~~ sluice gates, and instrumentation and controls required to perform the safety related function ~~are~~ OPERABLE. 27 57

In addition to the requirements above, for both SWS loops to be considered OPERABLE the required SW pumps must be powered from independent essential buses, to provide redundant and independent flow paths.

APPLICABILITY

In MODES 1, 2, 3, and 4, the SWS is a normally operating system that is required to support the OPERABILITY of the equipment serviced by the SWS. ~~and~~ Therefore, the SWS is required to be OPERABLE in these MODES. edit

3.7-31

(continued)

BASES

3.7-32

APPLICABILITY (continued) — In MODES 5 and 6, the OPERABILITY requirements of the SWS are determined by the systems it supports. 59

<INSERT B3.7-43A>

ACTIONS

A.1

loop

If one SWS train is inoperable, action must be taken to restore OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE SWS train is adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the OPERABLE SWS train loop could result in loss of SWS function. Required Action A.1 is modified by two Notes. The first Note indicates that the applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources—Operating," should be entered if an inoperable SWS loop train results in an inoperable EDG. The second Note indicates that the applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops—MODE 4," should be entered if an inoperable SWS loop train results in an inoperable DHR train. The 72 hour Completion Time is based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a DBA occurring during this period.

edit
edit
edit

B.1 and B.2

If the Required Action and ~~SWS train cannot be restored to OPERABLE status within the~~ associated Completion Time are not met, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours.

edit

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 3.7.8.1 7

Verifying the correct alignment for manual, power operated, and automatic valves in the SWS flow path provides assurance that the proper flow paths exist for SWS operation. This SR does not apply to valves that are locked, sealed, or

(continued)

<INSERT B3.7-43A>

Although the systems it supports may be required to be OPERABLE, the SWS is not required to meet the same OPERABILITY requirements in MODES 5 and 6 as it must in MODES 1, 2, 3, and 4. The definition of OPERABILITY embodies the principle that a system can perform its function(s) only if all necessary support systems are capable of performing their related support functions. If the supported system is capable of performing its safety function without reliance on the SWS, then the SWS is not required to be OPERABLE. Similarly, operation with the SWS in a less than fully qualified state is acceptable provided an assessment has been performed to determine that the supported system remains capable of performing its safety function.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.8.1 7 (continued)

otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is ~~based on engineering judgment, is~~ consistent with the existence of procedural controls governing valve operation, and ensures correct valve positions. edit

This SR is modified by a Note indicating that the isolation of ~~the SWS~~ components or systems may render those components inoperable, ~~but~~ does not affect the OPERABILITY of the SWS. supported by the SWS However, such isolation edit

SR 3.7.8.2 7

The SR verifies proper automatic operation of the SWS valves. The SWS is a normally operating system that cannot be fully actuated as part of the normal testing. 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. Operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint. edit

SR 3.7.8.3 7

<INSERT B 3.7-44A> ~~The SR verifies proper automatic operation of the SWS pumps on an actual or simulated actuation signal. The SWS is a normally operating system that cannot be fully actuated as part of normal testing during normal operation.~~ 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

3.7-08

(continued)

3.7-08 **<INSERT B3.7-44A>**

This SR requires verification that the normally operating SWS pumps (A and C) automatically restart following restoration of power to the respective bus. In addition, the B SWS pump, normally in the standby condition, must be verified to start to support each SWS train for which it is expected to be aligned upon associated ES actuation (with time delay) with simulated failure of the normally operating pump for that train.

3.7-08

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.7.3 (continued)

Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at an [18] month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. ~~F~~SAR, Section ~~[9.2.1]~~ 9.3.
2. ~~F~~SAR, Section ~~[~~6.2~~]~~.
3. ~~F~~SAR, Section ~~[~~6.3~~]~~.
4. SAR, Section 9.5.
5. 10 CFR 50.36.

edit

6

B 3.7 PLANT SYSTEMS

B 3.7.~~9~~ 8 ~~Ultimate Heat Sink (UHS)~~ Emergency Cooling Pond (ECP)

BASES

BACKGROUND

The ~~UHS~~ ECP provides a shared heat sink for ~~process and~~ removing operating heat from safety related components ~~during a transient or accident as well as during normal operation.~~ if the heat sink provided by the Dardanelle Reservoir is unavailable. This is done utilizing the Service Water System (SWS).

The ~~UHS has been defined as that~~ ECP is a portion of the complex of water sources~~,~~ which fulfill the ultimate heat sink requirements for ANO. This complex includes the ~~including~~ necessary retaining structures ~~(e.g., a pond with its dam, or a river with its dam),~~ and the ~~canals or~~ piping ~~conduits~~ connecting the sources with, but not including, the ~~cooling water system~~ SWS intake structure~~s~~, as discussed in the ~~F~~SAR, Section ~~9.2.5~~ 9.3 (Ref. 1). ~~If cooling towers or portions thereof are required to accomplish the UHS safety functions, they should meet the same requirements as the sink.~~ The ~~two~~ principal function~~s~~ of the ~~UHS are~~ ECP is the dissipation of residual heat after a reactor shutdown. ~~and dissipation of residual heat after an accident.~~

~~A variety of complexes is used to meet the requirements for a UHS. A lake or an ocean may qualify as a single source. If the complex includes a water source contained by a structure, it is likely that a second source will be required.~~

The basic performance requirements for both units are that a 30 day supply of water be available~~,~~ and that the design basis temperatures of safety related equipment not be exceeded. ~~Basins of cooling towers generally include less than a 30 day supply of water, typically 7 days or less. A 30 day supply would be dependent on another source(s) and a makeup system(s) for replenishing the source in the cooling tower basin. For smaller basin sources, which may be as small as a 1 day supply, the systems for replenishing the basin and the backup source(s) become of sufficient importance that the makeup system itself may be required to meet the same design criteria as an Engineered Safety Feature (e.g., single failure considerations and multiple makeup water sources may be required).~~

Additional information on the design and operation of the system~~, along with a list of components served,~~ can be found in Reference 1.

28

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The ~~UHS~~ ECP is the sink for heat removal from the reactor core following ~~all accidents~~ and~~/or~~ anticipated operational ~~occurrences~~ abnormality in which the unit is cooled down and placed on ~~a~~ decay heat removal following a loss of the Dardanelle Reservoir inventory which would be considered a single failure. ~~Its maximum post accident heat load occurs approximately 20 minutes after a design basis loss of coolant accident (LOCA). Near this time, the unit switches from injection to recirculation and the containment cooling systems are required to remove the core decay heat.~~ 57 28

The operating limits are based on conservative heat transfer analyses for the worst case ~~LOCA~~ <INSERT B3.7-47A>. Reference 1 provides the details of the assumptions used in the analysis. ~~These assumptions include: worst expected meteorological conditions, conservative uncertainties when calculating decay heat, and the worst case failure (e.g., single failure of a manmade structure).~~ <INSERT B3.7-47B> The ~~UHS~~ ECP is designed in accordance with Regulatory Guide 1.27 (Ref. 2), which requires a 30 day supply of cooling water ~~in the UHS~~. 28

The ~~UHS~~ ECP satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 3). 6

LCO

The ~~UHS~~ ECP is a backup system that is required to be OPERABLE to support the SWS. To be ~~and is~~ considered OPERABLE, ~~[1]~~ the ECP must contain~~s~~ a sufficient volume of water at or below the maximum temperature~~s~~ that would allow the SWS to operate for at least 30 days following the design basis ~~LOCA~~ event without ~~the loss of net positive suction head (NPSH), and without~~ exceeding the maximum design temperature of the equipment served by the SWS. To meet this condition, the ECP initial ~~UHS~~ temperature should not exceed ~~[90]~~ 100 °F, and the ~~level~~ volume of water should not fall below ~~[562 ft mean sea level]~~ 70 acre-feet during normal unit operation. edit 20

APPLICABILITY

In MODES 1, 2, 3, and 4, the ~~UHS~~ ECP is a ~~normally operating~~ backup system that is required to support the OPERABILITY of the equipment serviced by the ~~UHS~~ SWS and is required to be OPERABLE in these MODES.

In MODES 5 and 6, the OPERABILITY requirements of the ~~UHS~~ ECP are determined by the systems it supports. edit

<INSERT B3.7-47C> 59

3.7-32

(continued)

<INSERT B3.7-47A>

initial conditions that could be present considering a Unit 2 Design Basis Accident concurrent with a normal shutdown of Unit 1 and a loss of the Dardanelle Reservoir water inventory.

<INSERT B3.7-47B>

The minimum ECP requirements take into account: water loss from evaporation due to heat load and climatological conditions, fire pump usage, ECP bottom irregularities, suction pipe level at the ECP, and operator action in transferring the service water system from the Dardanelle Reservoir. Operator action is credited in the inventory analysis during the transfer of the service water system to the ECP. Specifically, pump returns are transferred to the ECP shortly after the Dardanelle Reservoir loss of inventory event begins and pump suctions are transferred later in the event depending on pump bay level. In the time frame between the transfer of the returns and suctions to the ECP, lake water is pumped into the ECP, increasing level. This additional water is required, along with that maintained in the ECP, to ensure a 64.5 inch depth, which corresponds to a 30 day supply of cooling water.

3.7-32

<INSERT B3.7-47C>

Although the systems it supports may be required to be OPERABLE, the ECP is not required to meet the same OPERABILITY requirements in MODES 5 and 6 as it must in MODES 1, 2, 3, and 4. The definition of OPERABILITY embodies the principle that a system can perform its function(s) only if all necessary support systems are capable of performing their related support functions. If the supported system is capable of performing its safety function without reliance on the ECP, then the ECP is not required to be OPERABLE. Similarly, operation with the ECP in a less than fully qualified state is acceptable provided an assessment has been performed to determine that the supported system remains capable of performing its safety function. It is important to recognize that single failure criteria is not applicable in MODES 5 and 6. Therefore, the availability of Lake Dardanelle as a heat sink during periods of ECP unavailability may be acceptable provided the probability of a loss of lake and the time to respond to a loss of lake event are considered when planning ECP unavailability periods.

BASES (continued)

ACTIONS

~~A.1~~

~~If one or more cooling towers have one fan inoperable (i.e., up to one fan per cooling tower inoperable), action must be taken to restore the inoperable cooling tower fan(s) to OPERABLE status within 7 days.~~

~~The 7 day Completion Time is reasonable, based on the low probability of an accident occurring during the 7 days that one cooling tower fan is inoperable in one or more cooling towers, the number of available systems, and the time required to complete the Required Action.~~ 19

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

ECP

If ~~the cooling tower fan cannot be restored to OPERABLE status within the associated Completion Time, or if~~ the ~~UHS~~ ECP is inoperable ~~[for reasons other than Condition A]~~, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. 19

SURVEILLANCE REQUIREMENTS

3.7-11

SR 3.7.~~9~~8.1 (together with SR 3.7.8.3 and SR 3.7.8.4)

This SR verifies that adequate long term (30 days) cooling ~~can be maintained~~ inventory is available. The level specified also ensures NPSH is available for operating the SWS pumps. The 24 hour Frequency is based on operating experience related to the trending of the ~~parameter variations~~ ECP level during the applicable MODES. This SR verifies that the ~~UHS~~ ECP indicated water level is ≥ ~~[ ]~~ 5 ft ~~[mean sea level]~~. 20

SR 3.7.~~9~~8.2

This SR ~~verifies~~ provides assurance that the heat sink for the SWS can ~~cool the CCW System to at least its maximum design temperature within the maximum~~ dissipate 20

(continued)

.SES

JRVEILLANCE
EQUIREMENTS

SR 3.7.~~9~~ 8.2 (continued) <INSERT B3.7-49A> 20

accident or normal heat loads for 30 days following ~~a~~ the ~~Design Basis~~ ~~Accident~~ event. The 24 hour Frequency is based on operating experience related to the trending of the ~~parameter~~ ECP temperature ~~variations~~ during the applicable MODES. This SR verifies that the ~~UHS~~ ECP average water temperature at the point of discharge from the emergency cooling pond (i.e., SWS suction) is ≤ ~~[90]~~ 100 °F. 20

<INSERT B3.7-49B>

SR 3.7.~~9~~ 8.3

<INSERT B3.7-49C>

~~Operating each cooling tower fan for ≥ [15] minutes ensures that all fans are OPERABLE and that all associated controls are functioning properly. It also ensures that fan or motor failure, or excessive vibration, can be detected for corrective action. The 31 day Frequency is based on operating experience, known reliability of the fan units, the redundancy available, and the low probability of significant degradation of the UHS cooling tower fans occurring between surveillances.~~ 19 20

REFERENCES

1. ~~F~~SAR, Section ~~[9.2.5]~~ 9.3. edit

2. Regulatory Guide 1.27, Rev. 1, "Ultimate Heat Sink for Nuclear Power Plants," March 1974. edit

3. 10 CFR 50.36. 6

<INSERT B3.7-49A>

3.7-10 The temperature, measured at the point of discharge from the ECP is considered a conservative average of total ECP conditions since solar gain, wind speed, and thermal current effects throughout the ECP will essentially be at equilibrium conditions under initial stagnant conditions.

<INSERT B3.7-49B>

This SR is modified by a Note indicating that the temperature monitoring is required to be performed only during the summer months (i.e., June 1 to September 30). During other periods of the year, the ECP temperature will not have the potential to reach the temperature limit.

<INSERT B3.7-49C>

3.7-11 This SR (together with SR 3.7.8.1 and 3.7.8.4) verifies that adequate inventory exists to support long term (30 days) cooling. Soundings are performed to ensure the water volume is within limits and that the indicated water level is indicative of an equivalent water volume for accident mitigation. The 12 month Frequency reflects the gradual pace of degradation of the physical properties of the ECP.

SR 3.7.8.4

This SR (together with SR 3.7.8.1 and 3.7.8.3) verifies that adequate inventory exists to support long term (30 days) cooling. Visual inspections of the loose stone (riprap) placed on the banks of the ECP and of the concrete slab spillway are performed to ensure any physical degradation is within acceptable limits to enable the ECP to fulfill its safety function. An engineering evaluation is performed of any apparent changes in visual appearance or other abnormal degradation to determine OPERABILITY. The 12 month Frequency reflects the gradual pace of degradation of the physical properties of the ECP.

B 3.7 PLANT SYSTEMS

B 3.7.~~10~~ 9 Control Room Emergency Ventilation System (CREVS)

BASES

BACKGROUND

The CREVS is a shared system which provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity~~, chemicals, or toxic gas~~.

39

The CREVS consists of two independent, redundant, fan and filter assemblies. Each fan circulates control room air through a filter train consist~~s~~ of a roughing filter, a high efficiency particulate air (HEPA) filter, and a charcoal ~~filter~~ adsorber. For control room pressurization, each train provides additional outside air filtered through a four inch bed of charcoal adsorber.

3.7-33

The CREVS is an emergency system. Upon receipt of ~~the~~ a unit specific high radiation ~~activating~~ signal~~(s)~~, the associated unit's normal control room ventilation system is ~~automatically~~ shut down, the control room envelope is isolated, and the CREVS ~~can be~~ is ~~manually~~ started. ~~The roughing filters and water condensing units remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA and charcoal filters.~~ The control room envelope is maintained sufficiently leak tight to minimize unfiltered air inleakage. ~~A single train will pressurize the control room with a 1.5 ft² LEAKAGE area to about 1/8 inch water gauge.~~ The CREVS operation is discussed in the U~~F~~SAR, Section ~~9.4~~ 9.7 (Ref. 1).

The CREVS is designed to maintain the control room for 30 days of continuous occupancy after a Design Basis Accident (DBA), without exceeding a 5 rem whole body dose or its equivalent to any part of the body.

APPLICABLE SAFETY ANALYSES

The shared CREVS components are arranged in ~~redundant~~ two safety related ventilation trains~~. The location of components and ducting within the control room envelope~~ which ensure~~s~~ an adequate supply of filtered air to all areas requiring access. The CREVS provides airborne radiological protection for the control room operators ~~as demonstrated by the control room accident dose analyses~~ for the ~~most limiting~~ design basis loss of coolant accident fission product release ~~presented in the FSAR, Chapter [15], (Ref. 2)~~ and for a fuel handling accident.

39

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The worst case single active failure of a CREVS component, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

~~For this unit, there are no sources of toxic gases or chemicals that could be released to affect control room habitability.~~ 39

In MODES 1 and 2, and during movement of irradiated fuel assemblies, ~~The~~ the CREVS satisfies Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 2). 6

<INSERT B3.7-51B>

edit

LCO

Two ~~independent and redundant~~ CREVS trains are required to be OPERABLE to ensure that at least one is available if a single failure disables the other train. Total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a large radioactive release.

edit

For a ~~The~~ CREVS train to be ~~is~~ considered OPERABLE~~, when~~ the ~~individual components necessary to control operator exposure are OPERABLE in both trains. A CREVS train~~ ~~is considered OPERABLE when~~ must include the associated:

<INSERT B3.7-51A>

a. ~~Fan is OPERABLE;~~ 39

b. OPERABLE HEPA filter and charcoal adsorber ~~are not excessively restricting flow, and are capable of performing their filtration functions~~; and

c. ~~Heater, demister,~~ ductwork~~, valves,~~ and dampers ~~are OPERABLE, and~~ sufficient to maintain air circulation can be ~~maintained~~. 39

In addition, the control room envelope ~~boundary~~, including the integrity of the walls, floors, ceilings, ductwork, and access doors, must be maintained within the assumptions of the design analysis and provide adequate makeup air flow. 32

<INSERT B3.7-51C> 54

ANO-239
ANO-290

APPLICABILITY

In MODES 1, 2, 3, and 4, the CREVS must be OPERABLE to ensure that the control room will remain habitable during and following a DBA. 60

(continued)

<INSERT B3.7-51A>

OPERABLE fan capable of being powered from both a normal and an OPERABLE emergency power source (Note: Because this is a shared system and may be powered from a Unit 2 source and distribution system for which there are no specific ANO-1 requirements, OPERABILITY includes requirements for both normal and emergency power sources and the associated distribution systems. If the CREVS train power sources or distribution system become inoperable, LCO 3.8.1, "AC Sources-Operating," is applicable for ANO-1 power sources, LCO 3.8.6, "Distribution Systems-Operating," is applicable for ANO-1 distribution systems, and LCO 3.0.6 allows the appropriate ACTIONS for these Specifications to be applied. However, if a required Unit 2 power source or distribution system becomes inoperable, the ACTIONS of ANO-1 LCO 3.7.9 must be applied for inoperable CREVS train(s).);

<INSERT B3.7-51B>

In MODES 3 and 4, the CREVS satisfies Criterion 4 of 10 CFR 50.36.

ANO-239
ANO-290

<INSERT B3.7-51C>

The LCO is modified by two Notes. Note 1 allows the control room boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for control room isolation is indicated. Note two requires that one CREVS train be capable of automatic actuation. The other train may be started manually, on failure of the first train.

AND-290

BASES

APPLICABILITY (continued)

During movement of irradiated fuel assemblies ~~[and during CORE ALTERATIONS]~~, the CREVS must be OPERABLE to cope with a release due to a fuel handling accident.

ACTIONS

A.1

With one CREVS train inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE CREVS train is adequate to perform the control room radiation protection function. However, the overall reliability is reduced because a failure in the OPERABLE CREVS train could result in loss of CREVS function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

<INSERT B3.7-52A>

B.1 ~~and B.2~~

or control room boundary — 54

In MODE 1, 2, 3, or 4, if the inoperable CREVS train cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

~~C~~ D.1, ~~C~~ D.2~~.1~~, ~~and C.2.2~~

Required Action and associated Completion Time of Condition A are not met

9

~~[In MODE 5 or 6, or]~~ during movement of irradiated fuel assemblies ~~[, or during CORE ALTERATIONS]~~, if the ~~inoperable CREVS train cannot be restored to OPERABLE status within the required Completion Time,~~ the OPERABLE CREVS train must immediately be placed in the emergency recirculation mode. This action ensures that ~~the remaining train is OPERABLE, that~~ no failures preventing automatic actuation will occur, and that any active failure will be readily detected. ~~Required Action C.1 is modified by a Note indicating to place the system in the emergency mode if automatic transfer to emergency mode is inoperable.~~ — 39 — 37

edit

(continued)

ANO-290

<INSERT B3.7-52A>

If the control room boundary is inoperable in MODES 1, 2, 3, and 4, the CREVS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE control room boundary within 24 hours. During the period that the control room boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactivity, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the Condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possible repair, and test most problems with the control room boundary.

C.1 and C.2

BASES

ACTIONS

~~D~~ ~~C~~.1, ~~C~~.2, and ~~C.2.2~~ (continued)

An alternative to Required Action ~~C~~ D.1 is to immediately suspend ~~activities~~ movement of irradiated fuel assemblies since this is an activity that could release radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

edit

~~D~~ E.1

~~[In MODE 5 or 6, or]~~ during movement of irradiated fuel assemblies ~~[, or during CORE ALTERATIONS]~~, when two CREVS trains are inoperable, action must be taken immediately to suspend ~~activities~~ that could release radioactivity that could enter the control room. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

edit

~~E~~ F.1

If both CREVS trains are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable control room boundary (i.e., Condition B), the CREVS may not be capable of performing the intended function and ~~the unit is in a condition outside the accident analysis~~ a loss of safety function has occurred. Therefore, LCO 3.0.3 must be entered immediately.

54

edit

SURVEILLANCE REQUIREMENTS

SR 3.7.~~10~~ 9.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every month adequately checks this system. ~~Monthly heater operations dry out any moisture that has accumulated in the charcoal because of humidity in the ambient air. [Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Systems without heaters]~~ need only be operated for ≥ 15 minutes to demonstrate the function of the system. ~~]~~ The 31 day Frequency is based on the known reliability of the equipment and the two train redundancy available.

39

and

This test is conducted on alternating trains semimonthly by initiating flow through the roughing filters, HEPA filters and charcoal adsorbers. The CREVS is designed

(continued)

ANO-290

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.~~10~~ 9.2

This SR verifies that the required CREVS testing is performed in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~. ~~The CREVS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 3).~~ The ~~[~~VFTP~~]~~ includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal. Specific test frequencies and additional information are discussed in detail in the ~~[~~VFTP~~]~~.

39 edit

SR 3.7.~~10~~ 9.3

This SR verifies that ~~[each CREVS train starts] [or the control room isolates] and operates~~ the CREVS automatically isolates the Control Room within 10 seconds and switches into a recirculation mode of operation with flow through the HEPA filters and Charcoal adsorber banks on an actual or simulated actuation signal. The Frequency of ~~[~~18~~]~~ months is consistent with ~~that specified~~ the guidance provided in ~~Reference 3~~ Regulatory Guide 1.52 (Ref. 3).

60 edit

ANO-239

~~SR 3.7.10.4~~

~~This SR verifies the integrity of the control room enclosure, and the assumed inleakage rates of the potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify that the CREVS is functioning properly. During the emergency mode of operation, the CREVS is designed to pressurize the control room ≥ [0.125] inches water gauge positive pressure with respect to adjacent areas, to prevent unfiltered inleakage. The CREVS is designed to maintain this positive pressure with one train at a flow rate of ≤ [3300] cfm. This value includes [300] cfm of outside air. The Frequency of [18] months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration SRs.~~

31

3.7-12

<INSERT B 3.7-54A>

32

REFERENCES

1. ~~F~~SAR, Section ~~[9.4.1]~~ 9.7 edit

2. ~~FSAR, Chapter [15]~~ 10 CFR 50.36. 6 edit

3. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants," Rev. 2, March 1978.

3.7.12

4. Standard Review Plan, Rev. 2, July 1981, Section 6.4, "Control Room Habitability Systems." 32

3.7-12

<INSERT B3.7-54A>

SR 3.7.9.4

This SR verifies the ability of the CREVS to provide outside air at a flow rate of approximately 333 cfm ±10%. Many factors must be taken into account to determine the overall expected dose consequences for control room personnel during various off-normal events. The CREVS makeup airflow is one of these factors that must be considered. Excessive makeup air or the inability of the CREVS units to supply design flow rates could result in an increase in the overall dose consequence to control room personnel. The flow verification ensures that an assumed amount of makeup air is available to account for boundary leak paths. If control room boundary leakage to adjacent areas is minimal, the makeup airflow rate will decrease accordingly as the differential pressure between the control room and adjacent areas increases. Therefore, the verification of makeup airflow capability may require creating leak paths (opening a door) when the control room envelope leak paths are minimal. The flowrate verification is consistent with SRP Section 6.4 (Reference 4) for those control rooms having a design makeup rate of ≥ 0.5 volume changes per hour. The Frequency of 18 months is considered adequate to detect any degradation of the outside air flow rate before it is reduced to a point at which sufficient pressurization will not occur.

B 3.7 PLANT SYSTEMS

B 3.7.~~14~~ 10 Control Room Emergency Air Conditioning ~~Temperature Control~~ System (CREA~~T~~CS)

BASES

BACKGROUND

The CREA~~T~~CS provides temperature control for the control room following isolation of the control room.

The CREA~~T~~CS consists of two independent and redundant trains that provide cooling of recirculated control room air. A cooling coil and a water cooled condensing unit are provided for each system to provide suitable temperature conditions in the control room for operating personnel and safety related control equipment. Ductwork, ~~valves or~~ dampers, and instrumentation also form part of the system. ~~Two redundant air cooled condensing units are provided as a backup to the water cooled condensing unit. Both the water cooled and air cooled condensing units must be OPERABLE for the CREATCS to be OPERABLE.~~ During ~~emergency~~ operation, the CREA~~T~~CS maintains the temperature ~~between 70°F and 85°F~~ in a range consistent with personnel comfort and long term equipment operation. The CREA~~T~~CS is a subsystem ~~providing air temperature control for the control room.~~ 40

3.7-33

The CREA~~T~~CS is an emergency system. On detection of high ~~containment building pressure or~~ radiation, ~~low Reactor Coolant System pressure, or high noble gas radioactivity in the station vent~~ the control room envelope is isolated, the normal control room ventilation system is ~~automatically~~ shut down, and the ~~Control Room Emergency Ventilation System can be manually~~ CREACS is started. A single train will provide the required temperature control. The CREA~~T~~CS operation to maintain control room temperature is discussed in the ~~F~~SAR, Section ~~[9.4]~~ 9.7 (Ref. 1). edit

APPLICABLE SAFETY ANALYSES

The design basis of the CREA~~T~~CS is to maintain control room temperature for 30 days of continuous occupancy.

The CREA~~T~~CS components are arranged in redundant, safety related trains. ~~During emergency operation, the CREATCS maintains the temperature between [70]°F and [95]°F.~~ A single active failure of a CREA~~T~~CS component does not impair the ability of the system to perform as designed. The CREA~~T~~CS is designed in accordance with Seismic Category I requirements. The CREA~~T~~CS is capable of removing sensible and latent heat loads from the control room, including

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

consideration of equipment heat loads and personnel occupancy requirements, to ensure a habitable environment and ~~equipment OPERABILITY~~. 40

In MODES 1 and 2, and during movement of irradiated fuel assemblies, The CREAOCS satisfies Criterion 3 of ~~the NRC Policy Statement~~ <INSERT B3.3-56B> 10 CFR 50.36 (Ref. 2). 6

LCO

Two independent and redundant trains of the CREAOCS are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other train. Total system failure could result in the ~~equipment operating~~ control room temperature exceeding limits in the event of an accident. <INSERT B3.7-56A> 40

~~The~~ For a CREAOCS ~~is~~ train to be considered OPERABLE ~~when~~ the individual components that are necessary to maintain control room temperature ~~are~~ must be OPERABLE ~~in both trains~~. These components include the cooling coils, ~~water cooled~~ condensing units, and associated temperature control instrumentation. In addition, the CREAOCS must be ~~OPERABLE to the extent that~~ capable of maintaining air circulation ~~can be maintained~~. edit 40

APPLICABILITY

In MODES 1, 2, 3, and 4, ~~[5, and 6,]~~ and during movement of irradiated fuel assemblies ~~and during CORE ALTERATIONS~~, the CREAOCS must be OPERABLE to ensure that the control room temperature will not exceed habitability and equipment OPERABILITY requirements following isolation of the control room. 40

ACTIONS

A.1

With one CREAOCS train inoperable, action must be taken to restore OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CREAOCS train is adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a failure in the OPERABLE CREAOCS train could result in a loss of CREAOCS function. The 30 day Completion Time is based on the low probability of an event occurring requiring control room isolation, the consideration that the remaining train ~~can~~ provide the required capabilities, and ~~the~~ alternate ~~safety or~~ nonsafety related cooling means that are available.

(continued)

<INSERT B3.7-56A>

(Note: Because this is a shared system and is normally powered from a Unit 2 source and distribution system for which there are no specific ANO-1 requirements, OPERABILITY includes requirements for both normal and emergency power sources and the associated distribution systems. If the CREVS train power sources or distribution system become inoperable, LCO 3.8.1, "AC Sources-Operating," is applicable for ANO-1 power sources, LCO 3.8.6, "Distribution Systems-Operating," is applicable for ANO-1 distribution systems, and LCO 3.0.6 allows the appropriate ACTIONS for these Specifications to be applied. However, if a required Unit 2 power source or distribution system becomes inoperable, the ACTIONS of ANO-1 LCO 3.7.10 must be applied for inoperable CREACS train(s).)

<INSERT B3.7-56B>

In MODES 3 and 4, the CREACS satisfies Criterion 4 of 10 CFR 50.36.

BASES

ACTIONS

A.1 (continued)

~~Concurrent failure of two CREATCS trains would result in the loss of function capability; therefore, LCO 3.0.3 must be entered immediately.~~

edit

B.1 and B.2

In MODE 1, 2, 3, or 4, if the ~~inoperable CREATCS train cannot be restored to OPERABLE status within the required Completion Time,~~ the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner without challenging unit systems.

Required Action and associated Completion Time of Condition A are not met,

40

C.1 and C.2

~~[In MODE 5 or 6, or]~~ ~~d~~uring movement of irradiated fuel ~~[, or during CORE ALTERATIONS]~~, if the ~~inoperable CREATCS train cannot be restored to OPERABLE status within the required Completion Time~~, the OPERABLE CREA~~T~~CS train must be placed in operation immediately. This action ensures that ~~the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that~~ any active failure will be readily detected.

40

An alternative to Required Action C.1 is to immediately suspend activities that could release radioactivity that might require the isolation of the control room. This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1

~~[In MODE 5 or 6, or]~~ ~~d~~uring movement of irradiated fuel assemblies ~~[, or during CORE ALTERATIONS]~~, with two CREA~~T~~CS trains inoperable, action must be taken to immediately suspend activities that could release radioactivity that might require isolation of the control room. This places

(continued)

BASES

ACTIONS

D.1 (continued)

the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

E.1

If both CREA~~T~~CS 9 trains are inoperable in MODE 1, 2, 3, or 4, ~~the CREATCS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore,~~ a loss of safety function has occurred, and LCO 3.0.3 must be entered immediately. edit

SURVEILLANCE REQUIREMENTS

edit

SR 3.7.11.1

<INSERT B 3.7-58A> ~~This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the [safety analyses]. This SR consists of a combination of testing and calculations. An [18] month Frequency is appropriate, as significant degradation of the CREATCS is slow and is not expected over this time period.~~ 52

REFERENCES

1. ~~F~~SAR 9, Section ~~[9.4]~~ 9.7. edit
2. 10 CFR 50.36. 6

<INSERT B3.7-58A>

SR 3.7.10.1 and SR 3.7.10.2

These SRs, in conjunction with periodic preventative maintenance activities, provide verification that the CREACS will maintain the control room temperature within acceptable bounds. SR 3.7.10.1 is performed on a staggered basis with one train being tested every two weeks. The Frequencies (31 days and 18 months) are appropriate as periodic preventative maintenance activities are routinely performed and significant degradation of the CREACS is not expected over these time periods.

B 3.7 PLANT SYSTEMS Penetration Room

B 3.7.~~12~~ 11 ~~Emergency~~ Ventilation System (EVS) PR

BASES

penetration areas in the event of penetration leakage from the reactor building

BACKGROUND

The PR EVS filters air from the ~~area of the active Emergency Core Cooling System (ECCS) components~~ during ~~the recirculation phase of~~ a loss of coolant accident (LOCA).

The PR EVS consists of two independent, redundant trains. Each train consists of ~~a~~ prefilter, a high efficiency particulate air (HEPA) filter, and ~~an~~ activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, ~~valves~~ or dampers, and instrumentation also form part of the system. The system initiates filtered ventilation of the ~~Auxiliary Building negative pressure area~~ penetration rooms following receipt of ~~a safety features~~ an engineered safeguards actuation system (ESAS) signal. ~~(SFAS)~~

<INSERT from ASA discussion B3.7-60>

~~The EVS is a standby system. During emergency operations, the EVS dampers are realigned, and fans are started to begin filtration.~~ Upon receipt of the ~~SFAS~~ ESAS signal~~(s)~~, normal air discharges from the ~~negative pressure~~ area are isolated, and ~~the stream of ventilation~~ air discharges through the system filter 15 ~~trains~~ s. The prefilters remove any large particles in the air, and any entrained water droplets present, to prevent excessive loading of the HEPA filters and charcoal adsorbers. 41

The PR EVS is discussed in the ~~F~~SAR, Sections ~~[6.2.3], [9.4.2],~~ 6.5 and ~~[15.4.5]~~ 14.2.2.5 (Refs. 1, ~~2~~, and ~~3~~ 2, respectively).

APPLICABLE SAFETY ANALYSES

The design basis of the PR EVS is established by the ~~large break~~ LOCA. The system ~~evaluation assumes a passive failure of the ECCS outside containment, such as an ECCS pump seal failure during the recirculation mode. In such a case, the system limits radioactive release to within 10 CFR 100 (Ref. 4) requirements.~~ provides filtration for the most likely location of reactor building leakage, i.e., at the penetrations. The analysis of the effects and consequences of a ~~large break~~ LOCA is presented in Reference ~~3~~ 2. ~~The EVS also actuates following a small break LOCA, in those cases where the unit goes into the recirculation mode of long term cooling, and to cleanup releases of smaller leaks, such as from valve stem packing.~~ 41

(continued)

3.7-34

BASES

APPLICABLE SAFETY ANALYSES (continued)

Insert in Background

~~Two types of system failures are considered in the accident analysis: complete loss of function, and excessive LEAKAGE. Either type of failure may result in a lower efficiency of removal of any gaseous and particulate activity released to the ECCS pump rooms following a LOCA.~~ 41

lead PRVS

~~Following a LOCA, an ESFAS signal starts the EVS fans and opens the dampers located in the penetration room outlet ductwork. The ESFAS signal closes all containment isolation valves and purge system valves. The purge system fans, if running, are shut down automatically.~~ 41

If proper flow is not achieved within 20 seconds, the lead system is automatically stopped and 5 seconds later the standby system starts.

PR

The ~~E~~VS satisfies Criterion 3 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 3). 6

In MODES 1 and 2,

LCO

<INSERT B3.7-60A>

Two ~~independent and~~ redundant trains of the ~~E~~VS are required to be OPERABLE to ensure that at least one is available, assuming that a single failure disables the other train coincident with loss of offsite power. ~~Total system failure could result in atmospheric release from the negative pressure area boundary exceeding Reference 4 limits in the event of a Design Basis Accident (DBA).~~ 41

~~The EVS is considered OPERABLE when the individual components necessary to maintain the negative pressure area boundary filtration are OPERABLE in both trains.~~ edit

For a ~~An~~ ~~E~~VS train ~~is~~ to be considered OPERABLE, ~~when~~ its associated:

a. Fan ~~is~~ must be OPERABLE;

b. HEPA filter and charcoal adsorber ~~are~~ must not be excessively restricting flow, and ~~are~~ must be capable of performing their filtration functions; and

c. ~~Heater, demister,~~ Required ductwork, ~~valves~~ and dampers ~~are~~ must be OPERABLE. ~~and air circulation can be maintained.~~ 41 54

AN0-290

<INSERT B3.7-60B>

APPLICABILITY

In MODES 1, 2, 3, and 4, the ~~E~~VS is required to be OPERABLE consistent with the OPERABILITY requirements of the ~~ECCS~~ reactor building.

In MODES 5 and 6, the ~~E~~VS is not required to be OPERABLE since the ~~ECCS~~ reactor building is not required to be OPERABLE. 41

(continued)

<INSERT B3.7-60A>

In MODES 3 and 4, the PRVS satisfies Criterion 4 of 10 CFR 50.36.

ANO-290

<INSERT B3.7-60B>

The LCO is modified by a Note allowing the PRVS negative pressure boundary to be opened intermittently under administrative controls. For entry and exit through doors the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for PRVS negative pressure boundary isolation is indicated.

BASES (continued)

ACTIONS

A.1

PR

With one ~~E~~VS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time, the remaining OPERABLE train is adequate to perform the ~~E~~VS PR safety function. However, the overall reliability is reduced because a single failure in the OPERABLE ~~E~~VS train could result in loss of ~~E~~VS PR function.

reactor building

The 7 day Completion Time is appropriate because the risk contribution is less than that of the ~~ECCS~~ (~~72~~ 1 hour Completion Time), and this system is not a direct support system for the ~~ECCS~~. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability. 41

reactor building

AND-290

<INSERT B3.7-61A>

~~B~~ C.1 and ~~B~~ C.2 54

Required Action and 47

are not met, or with both PRVS trains inoperable, 50

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

SURVEILLANCE REQUIREMENTS

SR 3.7.~~12~~ 11.1

Standby systems should be checked periodically to ensure that they function properly. Since the environment and normal operating conditions on this system are not severe, testing each train once a month provides an adequate check on this system. ~~Monthly heater operations dry out any moisture that may have accumulated in the charcoal from humidity in the ambient air. [Systems with heaters must be operated ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.]~~ The 31 day Frequency is based on known reliability of equipment and the two train redundancy available. 41

(continued)

ANO-290 **<INSERT B3.7-61A>**

B.1

If the PRVS negative pressure boundary is inoperable, the PRVS trains cannot perform their intended functions. Actions must be taken to restore an OPERABLE PRVS negative pressure boundary within 24 hours. During the period that the PRVS negative pressure boundary is inoperable, appropriate compensatory measures (consistent with the intent, as applicable, of GDC 64 and 10 CFR Part 100) should be utilized to control and minimize the release of radioactive materials from the reactor building to the environment in post accident conditions. Preplanned measures should be available to address these concerns for intentional and unintentional entry into the Condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possible repair, and test most problems with the PRVS negative pressure boundary.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.~~12~~ 11.2

This SR verifie~~s~~ s that the required ~~E~~ PR VS testing is performed in accordance with the [Ventilation Filter Testing Program (VFTP)]. ~~The EVS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 5).~~ The [VFTP] includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal ~~(general use and following specific operations)~~. Specific test frequencies and additional information are discussed in detail in the [VFTP].

41

SR 3.7.~~12~~ 11.3

This SR verifies that each ~~E~~ PR VS train starts and operates on an actual or simulated actuation signal. The [18] month Frequency is consistent with ~~that specified~~ the guidance provided in ~~Reference 5~~ Regulatory Guide 1.52 (Ref. 4).

edit

~~SR 3.7.12.4~~

~~This SR verifies the integrity of the negative pressure boundary area. The ability of the EVS to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper functioning of the EVS. During the [post accident] mode of operation, the EVS is designed to maintain a slight negative pressure in the negative pressure boundary area with respect to adjacent areas to prevent unfiltered LEAKAGE. The EVS is designed to maintain this negative pressure at a flow rate of [3000] cfm from the negative pressure boundary area. The Frequency of [18] months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration SRs.~~

36

~~SR 3.7.12.5~~

~~Operating the EVS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the EVS filter bypass damper is verified if it can be closed. An [18] month Frequency is consistent with that specified in Reference 5.~~

33

(continued)

BASES (continued)

REFERENCES

1. ~~F~~ SAR, Section ~~[6.2.3]~~ 6.5. edit
2. ~~F~~ SAR, Sections ~~[9.4.2]~~ 14.2.2.5 and 14.2.2.6 edit
~~3. FSAR, Section [15.4.6].~~ edit 6
~~4. 10 CFR 100.11.~~
3. 10 CFR 50.36. edit
4. ~~5.~~ Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants," Rev. 2, March 1978.

B 3.7 PLANT SYSTEMS

B 3.7.~~13~~ 12 Fuel ~~Storage Pool~~ Handling Area Ventilation System ~~(FSPVS)~~ (FHAVS)

BASES

BACKGROUND

The ~~FSPVS~~ FHAVS ~~provides negative pressure in the fuel storage area, and~~ filters airborne radioactive ~~particulates~~ from the area of the spent fuel pool following a fuel ~~handling~~ material accident.

edit

The ~~FSPVS~~ FHAVS consists of ~~portions of the normal Fuel Handling Area Ventilation System (FHAVS), the station Emergency Ventilation System (EVS), ductwork bypasses, and dampers. The portion of the normal FHAVS used by the FSPVS consists of ducting between the spent fuel pool and the normal FHAVS exhaust fans or dampers, and redundant radiation detectors installed close to the suction end of the FHAVS exhaust fan ducting. The portion of the EVS used by the FSPVS consists of two independent, redundant trains. Each train consists of a heater,~~ Auxiliary Building Heating, Ventilation, and Air Conditioning System. The FHAVS single train which includes a supply fan, prefilter, ~~or~~ high efficiency particulate air (HEPA) filter, activated charcoal adsorber section for removal of gaseous activity (principally iodines), and two exhaust fans. Ductwork, ~~valves or~~ dampers, and instrumentation also form part of the system. ~~Two isolation valves are installed in series in the ductwork between the FHAVS and the EVS to provide isolation of the EVS from the FHAVS on an Engineered Safety Feature actuation signal. These valves are opened prior to fuel handling operations. The EVS is the subject of LCO 3.7.12, "Emergency Ventilation System (EVS)," and is fully described in the FSAR, Section [6.2.3], Reference 12. A ductwork bypass with redundant dampers connects the FHAVS to the EVS.~~

42

During ~~normal~~ operation, the exhaust from the fuel handling area is passed through the FHAVS exhaust filter and is discharged through the station vent stack. ~~In the event of a fuel handling accident, the radiation detectors (one per EVS train), located at the suction of the FHAVS exhaust fan ducting, send signals to isolate the FHAVS supply and exhaust fans and ductwork, open the redundant dampers in the bypass ductwork, and start the EVS fans. The EVS fans pull the air from the fuel handling area, creating a negative pressure, and discharge the filtered air to the station vent.~~

The FHAVS is discussed in the ~~F~~SAR, Sections ~~[6.2.3]~~, 9.7, ~~[9.4.2]~~, and ~~[15.4.7]~~ 14.2.2 (Refs. 1, ~~2~~, and ~~3~~ 2, respectively).

edit

(continued)

BASES

BACKGROUND (continued)

~~because it may be used for normal as well as post accident, atmospheric cleanup functions.~~ 42

APPLICABLE SAFETY ANALYSES

The FHAVS ~~FSPVS~~ design basis is established by the ~~consequences of the limiting Design Basis Accident (DBA), which is a~~ fuel handling accident. The analysis of the fuel handling accident, given in Reference ~~3~~ 2, ~~assumes that a certain number of fuel rods in an assembly are damaged. The DBA analysis of the fuel handling accident assumes that only one train of the FSPVS is functional due to a single failure that disables the other train. The accident analysis~~ credits the FHAVS for a ~~accounts for the~~ reduction in the amount of airborne radioactive material released to the environment. ~~provided by the remaining one train of this filtration system.~~ The~~se~~ assumptions and the analysis ~~follow the guidance provided in Regulatory Guide 1.25 (Ref. 4).~~ are further discussed in Reference 2. 42

The FHAVS ~~FSPVS~~ satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 3). 6

LCO

During movement of irradiated fuel, ~~[Two] independent and redundant trains of~~ the FHAVS is ~~FSPVS are~~ required to be OPERABLE and operating ~~to ensure that at least one is available, assuming a single failure that disables the other train coincident with a loss of offsite power. Total system failure could result in the atmospheric release from the fuel handling area exceeding 10 CFR 100 (Ref. 5) limits in the event of a fuel handling accident.~~ 35 42

~~The FSPVS is considered OPERABLE when the individual components necessary to control operator exposure in the fuel handling building are OPERABLE in both trains. An FSPVS train is~~ For The FHAVS to be considered OPERABLE~~, when its associated~~:

1. ~~Fan is~~ One exhaust fan must be OPERABLE;

2. HEPA filter and charcoal adsorber ~~are~~ must not be excessively restricting flow, and ~~are~~ must be capable of performing their filtration functions; and

3. ~~[Heater, demister,]~~ ductwork, ~~valves,~~ and dampers ~~are~~ must be OPERABLE~~, and air circulation can be maintained~~.

The FHAVS must be operating since it does not automatically start following a fuel handling accident. A supply fan may be operating, but is not required for FHAVS OPERABILITY. 35

(continued)

BASES (continued)

APPLICABILITY

~~In [MODES 1, 2, 3, and 4] the FSPVS is required to be OPERABLE to provide fission product removal associated with ECCS leaks due to a loss of coolant accident (refer to LCO 3.7.12) for units that use this system as part of their EVSs.~~ 34

During movement of FHAVS irradiated fuel assemblies in the fuel handling area, the ~~FSPVS~~ is always required to be OPERABLE and operating to mitigate the consequences of a fuel handling accident. 35

~~In MODES 5 and 6, the FSPVS is not required to be OPERABLE since the ECCS is not required to be OPERABLE.~~ 34

ACTIONS

(INSERT B 3.7-66A) 34

A.1

~~With one FSPVS train inoperable, action must be taken to restore OPERABLE status within 7 days. During this time period, the remaining OPERABLE train is adequate to perform the FSPVS function. However, the overall reliability is reduced because a single failure in the OPERABLE FSPVS train could result in a loss of FSPVS functioning. The 7 day Completion Time is based on the risk from an event occurring requiring the inoperable FSPVS train, and ability of the remaining FSPVS train to provide the required protection.~~ 34

~~B.1 and B.2~~

~~In MODE 1, 2, 3, or 4, when Required Action A.1 cannot be completed within the associated Completion Time, or when both FSPVS trains are inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The 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 and C.2~~

~~If the inoperable FSPVS train cannot be restored to OPERABLE status within the required Completion Time, during movement~~

(continued)

<INSERT B3.7-66A>

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note which states that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

BASES

ACTIONS

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

~~of irradiated fuel assemblies in the fuel handling area, the OPERABLE FSPVS train must be started immediately or fuel movement suspended. This action ensures that the remaining train is OPERABLE, that no undetected failures preventing system operation will occur, and that any active failures will be readily detected.~~

~~If the system is not placed in operation, this action requires suspension of fuel movement, which precludes a fuel handling accident. This action does not preclude the movement of fuel assemblies to a safe position.~~

34

~~D.~~1

35

When ~~two trains of~~ the ~~FSPVS are~~ FHAVS is inoperable or not in operation during movement of irradiated fuel assemblies in the fuel handling area, ~~the unit must be placed in a condition in which the LCO does not apply. This LCO involves~~ immediately suspending movement of irradiated fuel assemblies in the fuel handling area. This does not preclude the movement of fuel to a safe position.

INSERT B3.7-67A

edit

35

SURVEILLANCE REQUIREMENTS

INSERT B3.7-67B

~~SR 3.7.13.1~~

~~Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not severe, testing each train once every month provides an adequate check on this system. Monthly heater operation dries out any moisture accumulated in the charcoal from humidity in the ambient air. [Systems with heaters must be operated for ≥ 10 continuous hours with the heaters energized. Systems without heaters need only be operated for ≥ 15 minutes to demonstrate the function of the system.] The 31 day Frequency is based on the known reliability of the equipment, and the two train redundancy available.~~

9

SR 3.7.~~13~~ 12.2

9

This SR verifies that the required ~~FSPVS~~ FHAVS testing is performed in accordance with the ~~[~~Ventilation Filter Testing

6

(continued)

<INSERT B3.7-67A>

immediate action must be taken to preclude the occurrence of an accident. This is achieved by

<INSERT B3.7-67B>

SR 3.7.12.1

Periodic verification of the operation of the FHAVS assures immediate availability of filtration following a fuel handling accident. A 12 hour Frequency is sufficient, considering the system indications and alarms available to the operator for monitoring the FHAVS in the control room.

FHAVS
~~FSPVS~~

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.~~13~~12.2 (continued)

Program (VFTP)~~]~~. ~~The FSPVS filter tests are in accordance with Regulatory Guide 1.52 (Ref. 6).~~ The ~~[~~VFTP~~]~~ includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal ~~(general use and following specific operations)~~. Specific test frequencies and additional information are discussed in detail in the ~~[~~VFTP~~]~~. 42

edit

~~SR 3.7.13.3~~

~~This SR verifies that each FSPVS train starts and operates on an actual or simulated actuation signal. The 18 month Frequency is consistent with that specified in Reference 6.~~ 35

~~SR 3.7.13.4~~

~~This SR verifies the integrity of the fuel handling area. The ability of the fuel handling area to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper function of the FSPVS. During the [post accident] mode of operation, the FSPVS is designed to maintain a slight negative pressure in the fuel handling area to prevent unfiltered LEAKAGE. The FSPVS is designed to maintain this negative pressure at a flow rate of ≤ [3000] cfm to the fuel handling area. The Frequency of [18] months on a STAGGERED TEST BASIS is consistent with industry practice.~~ 36

~~SR 3.7.13.5~~

~~Operating the FSPVS filter bypass damper is necessary to ensure that the system functions properly. The OPERABILITY of the FSPVS filter bypass damper is verified if it can be opened. A Frequency of [18] months is specified in Reference 6.~~ 33

(continued)

BASES (continued)

REFERENCES

1. ~~F~~SAR, Section ~~6.2.3.~~ 9.7.

~~2. FSAR, Section 9.4.21.~~

2 ~~3~~. ~~F~~SAR, Section ~~15.4.7.~~ 14.2.2.

3. 10 CFR 50.36.

~~4. Regulatory Guide 1.25.~~

~~5. 10 CFR 100.11.~~

~~6. Regulatory Guide 1.52.~~

edit

6

edit

 H10

B 3.7 PLANT SYSTEMS

B 3.7.1~~0~~ 3 Spent Fuel ~~Storage~~ Pool Water Level H10

BASES

BACKGROUND

The minimum water level in the Spent fuel ~~storage~~ pool meets the assumption of iodine decontamination factors following a fuel handling accident. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel. H10

A general description of the Spent fuel ~~storage~~ pool design is given in the FSAR, Section ~~[9.1.2]~~ 9.6.1.3, Reference 1. The Spent Fuel Pool Cooling and Cleanup System is given in the FSAR, Section ~~[9.1.3]~~ 9.4 (Ref. 2). The assumptions of the fuel handling accident are given in the FSAR, Section ~~[15.4.7]~~ 14.2.2.3 (Ref. 3). H10

APPLICABLE SAFETY ANALYSES

<INSERT B3.7-70A>

~~The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Regulatory Guide 1.25 (Ref. 4). The resultant 2 hour thyroid dose to a person at the exclusion area boundary is below 10 CFR 100 (Ref. 5) guidelines.~~

~~According to Reference 4, there is 23 ft of water between the top of the damaged fuel bundle and the fuel pool surface for a fuel handling accident. With 23 ft, the assumptions of Reference 4 can be used directly. In practice, the LCO preserves this assumption for the bulk of the fuel in the storage racks. In the case of a single bundle dropped and lying horizontally on top of the spent fuel rack, however, there may be < 23 ft above the top of the fuel bundle and the surface, by the width of the bundle. To offset this small nonconservatism, the analysis assumes that all fuel rods fail, although the analysis shows that only the first [few] rows fail from a hypothetical maximum drop.~~ 58

The Spent fuel ~~storage~~ pool water level satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref.). H10 H6

(continued)

3.7-14

<INSERT B3.7-70A>

During movement of irradiated fuel assemblies, the water level in the spent fuel pool is an initial condition design parameter in the analysis of the fuel handling accident in the fuel handling building postulated by Regulatory Guide 1.25 (Ref. 4). A minimum water level of 23 feet above the top of the irradiated fuel assemblies seated in the storage racks (Regulatory Position C.1.c of Ref. 4) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 4) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the spent fuel pool water. The fuel pellet to cladding gap is assumed to contain 12% of the total fuel rod iodine inventory (Ref. 3).

The fuel handling accident analysis inside the fuel handling building is described in Reference 3. With a minimum water level of 23 feet above the top of the irradiated fuel assemblies seated in the storage racks, and a minimum decay time of 100 hours prior to fuel handling, the analysis demonstrates that the iodine release due to a postulated fuel handling accident is adequately captured by the water, and offsite doses are maintained within allowable limits (Ref. 5).

3.7-14

Spent

BASES (continued)

LCO

The specified water level preserves the assumptions of the fuel handling accident analysis (Ref. 3). As such, it is the minimum required for fuel storage and movement within the Spent fuel ~~storage~~ pool. (10)

APPLICABILITY

This LCO applies during movement of irradiated fuel assemblies in the Spent fuel ~~storage~~ pool since the potential for a release of fission products exists. (10)

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

When the initial conditions for an accident cannot be met, immediate action must be taken to preclude the occurrence of an accident. With the Spent fuel ~~storage~~ pool at less than the required level, the movement of fuel assemblies in the Spent fuel ~~storage~~ pool is immediately suspended. This effectively precludes the occurrence of a fuel handling accident. In such a case, unit procedures control the movement of loads over the spent fuel. This does not preclude movement of a fuel assembly to a safe position. (10)

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODES 1, 2, 3, and 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.13.1

This SR verifies that sufficient Spent fuel ~~storage~~ pool water is available in the event of a fuel handling accident. The water level in the Spent fuel ~~storage~~ pool must be checked periodically. The 7 day Frequency is appropriate because the volume in the pool is normally stable. Water level (10)

(continued)

3.7-14

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1~~0~~3.1 (continued)

changes are controlled by unit procedures and are acceptable, based on operating experience.

During refueling operations, the level in the Spent fuel ~~storage~~ pool is at equilibrium with that in the refueling canal, and the level in the refueling canal is checked daily in accordance with SR 3.9.6.1. ⊢10

REFERENCES

1. FSAR, Section ~~[9.1.2]~~ 9.6.1.3.
2. FSAR, Section ~~[9.1.3]~~ 9.4.
3. FSAR, Section ~~[15.4.7]~~ 14.2.2.3.
4. Regulatory Guide 1.25.
5. 10 CFR 100.11.
6. 10 CFR 50.36. ⊢6

3.7-14

44

B 3.7 PLANT SYSTEMS

B 3.7.~~15~~ 14 Spent Fuel Pool Boron Concentration

BASES

BACKGROUND

As described in the Bases for ~~following~~ LCO 3.7.~~16~~ 15, "Spent Fuel Pool ~~Assembly~~ Storage," fuel assemblies are stored in the spent fuel pool racks ~~[in a "checkerboard" pattern]~~ in accordance with criteria based on initial enrichment and discharge burnup. Although the water in the spent fuel pool is normally borated to ≥ ~~[500]~~ 1600 ppm, the criteria that limit the storage of a fuel assembly to specific rack locations are conservatively developed without taking credit for boron. edit

INSERT B3.7-73A

APPLICABLE SAFETY ANALYSES

~~A fuel assembly could be inadvertently loaded into a spent fuel rack location not allowed by LCO 3.7.16 (e.g., an unirradiated fuel assembly or an insufficiently depleted fuel assembly). This accident is analyzed assuming the extreme case of completely loading the spent fuel pool racks with unirradiated assemblies of maximum enrichment. Another type of postulated accident is associated with a fuel assembly that is dropped onto the fully loaded spent fuel pool storage rack. Either incident could have a positive reactivity effect, decreasing the margin to criticality. However, the negative reactivity effect of the soluble boron compensates for the increased reactivity caused by either one of the two postulated accident scenarios.~~

INSERT B3.7-73B

The concentration of dissolved boron in the fuel storage pool satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 4).

6

LCO

The specified concentration ~~[≥ 500]~~ ≥ 1600 ppm of dissolved boron in the spent fuel ~~storage~~ pool conservatively preserves the assumption used in the analyses of the potential accident scenarios ~~described above~~. This concentration of dissolved boron is the minimum required concentration for fuel assembly storage and movement within the spent fuel ~~storage~~ pool.

APPLICABILITY

This LCO applies whenever fuel assemblies are stored in the spent fuel pool, until a complete spent fuel pool

(continued)

<INSERT B3.7-73A>

in the spent fuel pool water.

The spent fuel storage pool is divided into two separate and distinct regions as shown in SAR Figure 9-53 which, for the purpose of criticality considerations, are considered as separate pools. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.10 wt% U-235, or spent (irradiated) fuel regardless of the discharge fuel burnup. Region 2 is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within

3.7-14 the acceptable domain according to Figure 3.7.15-1. Fuel assemblies not meeting the criteria of Figure 3.7.15-1 shall be stored in accordance with paragraph 4.3.1.1.e in SAR Section 4.3, Fuel Storage.

The water in the spent fuel storage pool normally contains soluble boron, which results in large subcriticality margins under actual operating conditions. However, the NRC guidelines specify that the limiting k_{eff} of 0.95 be evaluated in the absence of soluble boron. Hence, the design of both regions is based on the use of unborated water, which maintains each region in a subcritical condition during normal operation with the regions fully loaded. The double contingency principle discussed in ANSI N-16.1-1975 and the April 1978 NRC letter (Ref. 1) allows credit for soluble boron under other abnormal or accident conditions, since only a single accident need be considered at one time. Thus, for accident conditions, the presence of soluble boron in the spent fuel pool water can be assumed as a realistic condition. For example, accident scenarios are postulated which could potentially increase the reactivity and reduce the margin to criticality. To mitigate these postulated criticality related accidents, boron is dissolved in the pool water. Safe operation of the high density storage racks with no movement of assemblies may therefore be achieved by controlling the location

3.7-14 of each assembly in accordance with LCO 3.7.15, "Spent Fuel Pool Storage." Prior to movement of an assembly, it is necessary to perform SR 3.7.15.1.

<INSERT B3.7-73B>

Most accident conditions will not result in an increase in K_{eff} of the rack. Examples are the loss of cooling systems (reactivity decreases with decreasing water density) and dropping a fuel assembly on top of the rack (the rack structure pertinent for criticality is not deformed and the assembly has more that eight inches of water separating it from the active fuel in the rest of the rack which precludes interaction). However, accidents can be postulated which would increase reactivity such as inadvertent drop of an assembly between the outside periphery of the rack and the pool wall. Thus, for accident conditions, the presence of soluble boron in the storage pool water is assumed as a realistic initial condition.

The presence of 1600 ppm boron in the pool water will decrease reactivity by approximately 30% ΔK. Thus $K_{eff} \leq 0.95$ can be easily met for postulated accidents, since any reactivity increase will be much less than the negative worth of the dissolved boron.

BASES

APPLICABILITY (continued)

verification has been performed following the last movement of fuel assemblies in the spent fuel pool. This LCO does not apply following the verification since the verification would confirm that there are no misloaded fuel assemblies. With no further fuel assembly movement in progress, there is no potential for a misloaded fuel assembly or a dropped fuel assembly.

ACTIONS

A.1, A.2.1, and A.2.2

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply.

When the concentration of boron in the Spent fuel ~~storage~~ pool is less than required, immediate action must be taken to preclude the occurrence of an accident or to mitigate the consequences of an accident in progress. This is most efficiently achieved by immediately suspending the movement of the fuel assemblies. This does not preclude movement of a fuel assembly to a safe position.

INSERT B3.7-74A

edit

30

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, inability to suspend movement of fuel assemblies is not a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.14.1

edit

This SR verifies that the concentration of boron in the Spent fuel ~~storage~~ pool is within the required limit. As long as this SR is met, the analyzed incidents are fully addressed. The 7 day Frequency is appropriate because no major replenishment of pool water is expected to take place over a short period of time.

REFERENCES

~~None.~~ INSERT B3.7-74B

<INSERT B3.7-74A>

In addition, action must be immediately initiated to restore the spent fuel pool boron concentration to within its limit. An acceptable alternative is to immediately initiate performance of a spent fuel pool verification to ensure proper locations of the fuel since the last movement of fuel assemblies in the spent fuel pool. However, prior to resuming movement of fuel assemblies, the concentration of boron must be restored. Either of these actions are acceptable, and once initiated must be continued until the action is completed. The immediate Completion Time for initiation of these actions reflects the importance of maintaining a controlled environment for irradiated fuel.

<INSERT B3.7-74B>

1. Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978, NRC letter (Section 1.2) and implied in the proposed revision to Regulatory Guide 1.13 (Section 1.4, Appendix A).

2. SAR, Section 14.2.2.3.

3. Safety Evaluation Report, Section 2.1.3, License Amendment No. 76, April 15, 1983.

4. 10 CFR 50.36.

49

B 3.7 PLANT SYSTEMS

B 3.7.~~16~~ 15 Spent Fuel ~~Assembly~~ Pool Storage

49

BASES

BACKGROUND

The spent fuel assembly storage facility is designed to store either new (nonirradiated) nuclear fuel assemblies, or burned (irradiated) fuel assemblies in a vertical configuration underwater. The ~~storage~~ spent fuel pool is sized to store ~~[735]~~ 968 ~~irradiated~~ fuel assemblies. ~~which includes storage for [15] failed fuel containers.~~ The spent fuel storage cells are installed in parallel rows with center to center spacing of ~~[12 31/32]~~ 10.65 inches in ~~one direction, and [13 3/16] inches in the other orthogonal~~ each direction. ~~This spacing and "flux trap" construction, whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans,~~ is sufficient to maintain a k_{eff} of < 0.95 for spent fuel of original enrichment of up to ~~[3.5]~~ 4.10%. However, ~~as higher initial enrichment~~ fuel assemblies ~~are~~ to be stored in the spent fuel pool~~, they~~ Region 2 which do not meet enrichment and burnup criterion must be stored in a checkerboard pattern ~~taking into account fuel burnup~~ to maintain a k_{eff} of 0.95 or less. <INSERT B3.7-75B>

<INSERT B3.7-75A>

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53

APPLICABLE SAFETY ANALYSES

<INSERT B3.7-75C>

~~The spent fuel storage facility is designed for noncriticality by use of adequate spacing, and "flux trap" construction whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans.~~

The spent fuel ~~assembly~~ Pool storage satisfies Criterion 2 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 3).

edit 49

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LCO

The restrictions on the placement of fuel assemblies within the fuel pool, according to Figure~~s~~ [3.7.~~16~~15-1] ~~in the accompanying LCO,~~ ensure that the k_{eff} of the spent fuel pool will always remain ≤ 0.95 assuming the pool to be flooded with unborated water. The restrictions are consistent with the criticality safety analysis performed for the spent ~~[ ]~~ fuel pool~~, according to Figure [3.7.16-1]~~. Fuel assemblies not meeting the enrichment and burnup criteria ~~of Figure [3.7.16-1]~~ shall be stored in accordance with Specification 4.3.1.1.

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edit

<INSERT B 3.7-75D>

3.7-15

(continued)

<INSERT B3.7-75A>

The spent fuel storage pool is divided into two separate and distinct regions as shown in SAR Figure 9-53 which, for the purpose of criticality considerations, are considered as separate pools. Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.10 wt% U-235, or spent (irradiated) fuel regardless of the discharge fuel burnup. Region 2 is designed to accommodate fuel of various initial enrichments which have accumulated minimum burnups within 3.7-14 the acceptable domain according to Figure 3.7.15-1. Fuel assemblies not meeting the criteria of Figure 3.7.15-1 shall be stored in accordance with paragraph 4.3.1.1.e in SAR Section 4.3, Fuel Storage.

<INSERT B3.7-75B>

In order to prevent inadvertent fuel assembly insertion into two adjacent storage locations, vacant spaces adjacent to the faces of any fuel assembly which does not meet the Region 2 burnup criteria (unrestricted) are physically blocked before any such fuel assembly is placed in Region 2 (Ref. 1). In addition, the area designated for checkerboard arrangement is divided from the normal storage in Region 2 by a row of vacant storage spaces (Ref. 2).

<INSERT B3.7-75C>

Criticality of fuel assemblies in the spent fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between assemblies and inserting neutron poison between assemblies in Region 1. Region 2 controls fuel assembly interaction by fixing the minimum separation between assemblies and by setting enrichment and burnup criterion to limit fissile materials. This

3.7-15 **<INSERT B3.7-75D>**

In the event a checkerboard storage configuration is deemed necessary for a portion of Region 2, vacant spaces adjacent to the faces of any fuel assembly which does not meet the Region 2 burnup criteria (non-restricted) shall be physically blocked before any such fuel assembly may be placed in Region 2. This will prevent inadvertent fuel assembly insertion into two adjacent storage locations.

BASES (continued)

APPLICABILITY

This LCO applies whenever any fuel assembly is stored in [Region 2] of the spent fuel pool.

ACTIONS

A.1

Required Action A.1 is modified by a Note indicating that LCO 3.0.3 does not apply.

14

When the configuration of fuel assemblies stored in the spent fuel pool is not in accordance with Figure [3.7.18-1], immediate action must be taken to make the necessary fuel assembly movement(s) to bring the configuration into compliance with Figure [3.7.18-1] 15 or Specification 4.3.1.1

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edit

If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operation. Therefore, in either case, inability to move fuel assemblies is not sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.7.~~18~~ 15.1

This SR verifies by administrative means that the initial enrichment and burnup of the fuel assembly is in accordance with Figure [3.7.18-1] 15 in the accompanying LCO.

edit

REFERENCES

~~None.~~

1. SAR, Section 9.6.2.
2. Safety Evaluation Report for ANO-1 License Amendment No. 76, Section 2.1 (OCNA 048314) dated April 15, 1983.
3. 10 CFR 50.36.

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or Specification 4.3.1.1.
For fuel assemblies in the unacceptable range of Figure 3.7.15-1, performance of the SR will ensure compliance with Specification 4.3.1.1.

45

49

B 3.7 PLANT SYSTEMS

B 3.7.17 {4} Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube out-LEAKAGE from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicative of current conditions. During transients, I-131 spikes have been observed, as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products, in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, ~~anticipated operational occurrences~~ abnormalities, and accidents. (51)

(12) This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational Leakage") of primary coolant at the limit of (3.5) ~~1.0~~ µCi/gm (LCO 3.4.16, "RCS Specific Activity"). ~~The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant leakage. Most of the iodine isotopes have short half lives (i.e., < 20 hours). I-131, with a half life of 8.04 days, concentrates faster than it decays, but does not reach equilibrium because of blowdown and other losses.~~ (44)

~~With the specified activity limit, the resultant 2 hour thyroid dose to a person at the exclusion area boundary (EAB) would be about 0.79 rem if the main steam safety valves (MSSVs) are open for the 2 hours following a trip from full power.~~ INSERT B 3.7-77A (24)

Operating a unit at the allowable limits could result in a 2 hour ~~EAB~~ exclusion area boundary (EAB) exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits~~, or the limits established as the NRC staff approved licensing basis~~. (44)

(continued)

<INSERT B3.7-77A>

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

BASES (continued)

APPLICABLE SAFETY ANALYSES

INSERT B 3.7-78A

~~The accident analysis of the main steam line break, as discussed in the FSAR, Chapter [15] (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.1 µCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed established limits, (Ref. 1) for whole body and thyroid dose rates.~~

~~With a loss of offsite power, the remaining steam generator is available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric dump valves (ADVs). The Emergency Feedwater System supplies the necessary makeup to the steam generator. Venting continues until the reactor coolant temperature and pressure has decreased sufficiently for the Shutdown Cooling System to complete the cooldown.~~

~~In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and ADVs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.~~ 44

In MODES 1 and 2,

In MODES 3 and 4, the secondary specific activity limits satisfy Criterion 4 of 10 CFR 50.36.

Secondary specific activity limits satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 3). 6

LCO

As indicated in the Applicable Safety Analyses, the specific activity limit in the secondary coolant system of ≤ ~~[0.10]~~ 0.17 µCi/gm DOSE EQUIVALENT I-131 maintains the radiological consequences of a Design Basis Accident (DBA) ~~to a small fraction of~~ significantly less than the Reference 1 ~~limits~~ guideline doses. 44

Monitoring the specific activity of the secondary coolant ensures that, when secondary specific activity limits are exceeded, appropriate actions are taken, in a timely manner,

(continued)

3.7-37

<INSERT B3.7-78A>

For the purpose of determining a maximum allowable secondary coolant activity, the activity contained in the mass released following the rupture of a steam generator tube, a steam line break outside the reactor building and a loss of load incident were considered (Safety Evaluation Report for ANO-1 License Amendment No. 2, 1CNA057502, dated May 9, 1975 (Ref. 2)).

The whole body dose is negligible since any noble gases entering the secondary coolant system are continuously vented to the atmosphere by the condenser vacuum pumps. Thus, in the event of a loss of load incident or steam line break, there are only small quantities of these gases which would be released (Ref. 2).

The dose analysis performed to determine the maximum allowable reactor coolant activity assuming the maximum allowable primary to secondary leakage of 1 gpm as given in the Bases for LCO 3.4.13 indicated that the controlling accident to determine the allowable secondary coolant activity would be the rupture of a steam generator tube. For the loss of load incident with a loss of 205,000 pounds of water released to the atmosphere via the relief valves, the resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 μCi/gm would be 0.6 Rem with the same meteorological and iodine release assumptions used for the steam generator tube rupture as given in the Bases for LCO 3.4.13. For the less probable accident of a steam line break, the assumption is made that a loss of 10^6 pounds of water or the contents of one loop in the secondary coolant system occurs and is released directly to the atmosphere. Since the water will flash to steam, the total radioiodine activity is assumed to be released to the atmosphere. The resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 μCi/gm would be less than 28 Rem with the same meteorological assumptions used for the steam generator tube rupture and loss of load incident (Ref. 2).

3.7-37

BASES

LCO (continued)	to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.
APPLICABILITY	In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere. In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are at low pressure and primary to secondary LEAKAGE is minimal. Therefore, ~~monitoring of~~ secondary specific activity is not ~~required.~~ a concern. edit
ACTIONS	A.1 and A.2 DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant contributes to increased post accident doses. If secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.
SURVEILLANCE REQUIREMENTS	SR 3.7.~~17~~ 4 .1 This SR verifies that the secondary specific activity is within the limits of the accident analysis assumptions. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the ~~validity of the safety~~ analysis assumptions ~~as releases.~~ are met. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit. edit edit

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 100.11.
2. ~~FSAR, Chapter [15]~~ Safety Evaluation Report for ANO-1 License Amendment No. 2, 1CNA057502, dated May 9, 1975. 44
3. 10 CFR 50.36. 6

38

B 3.7 PLANT SYSTEMS

B 3.7.18 Steam Generator Level

BASES

BACKGROUND

A principal function of the steam generators is to provide superheated steam at a constant pressure (900 psia) over the power range. Steam generator water inventory is maintained large enough to provide adequate primary to secondary heat transfer. Mass inventory and indicated water level in the steam generator increases with load as the length of the four heat transfer regions within the steam generator vary. Inventory is controlled indirectly as a function of power and maintenance of a constant average primary system temperature by the feedwater controls in the Integrated Control System.

The maximum operating steam generator level is based primarily on preserving the initial condition assumptions for steam generator inventory used in the FSAR steam line break (SLB) analysis (Ref. 1). An inventory of 62,600 lb was used in this analysis. The 62,600 lb must not be exceeded due to the concerns of a possible return to criticality because of primary side cooling following an SLB and the maximum pressure in the reactor building.

For a clean once through steam generator, the mass inventory in a steam generator for operating at 100% power is approximately 39,000 lb to 40,000 lb.

As a steam generator becomes fouled and the operating level approaches the limit of 96%, the mass inventory in the downcomer region increases approximately 10,000 lb, and adds to the total mass inventory of the steam generator. In matching unit data of startup level versus power, the steam generator performance codes have shown that fouling of the lower tube support plates does not significantly change the heat transfer characteristics of the steam generator. Thus, the steam temperature, or superheat, is not degraded due to the fouling of the tube support plates, and mass inventory changes are mainly due to the added level in the downcomer.

Analytically, increasing the fouling of the steam generator tube surfaces degrades the heat transfer capability of the steam generator, increases the mass inventory, and decreases the steam superheat at 100% power (2544 MW). The results

(continued)

38

BASES

BACKGROUND (continued)

were presented as the amount of mass inventory in each steam generator versus operating range level and steam superheat.

The limiting curve, which was determined from several steam generator performance code runs at a power level of 100%, conservatively bounds steam generator mass inventory value, when operating at power levels < 100%.

The points displayed in Figure 3.7.18-1, in the accompanying LCO, are the intercept points of the 57,000 lb mass value, and the operating range level x and steam superheat values.

The steam generator performance analysis also indicated that startup and full range level instruments are inadequate indicators of steam generator mass inventory at high power levels due to the combination of static and dynamic pressure losses. If the water level should rise above the 96% upper limit, the steam superheat would tend to decrease due to reduced feedwater heating through the aspirator ports. Normally, a reduction in water level is manually initiated to maintain steam flow through the aspirator port by reducing the power level. Thus, the superheat versus level limitation also tends to ensure that, in normal operation, water level will remain clear of the aspirator ports.

Feedwater nozzle flooding would impair feedwater heating, and could result in excessive tube to shell temperature differentials, excessive tubesheet temperature differentials, and large variations in pressurizer level.

APPLICABLE SAFETY ANALYSES

The most limiting Design Basis Accident that would be affected by steam generator operating level is a steam line failure. This accident is evaluated in Reference 1. The parameter of interest is the mass of water, or inventory, contained in the steam generator due to its role in lowering Reactor Coolant System (RCS) temperature (return to criticality concern), and in raising containment pressure during an SLB accident. A higher inventory causes the effects of the accident to be more severe. Figure 3.7.18-1, in the accompanying LCO, is based upon maintaining inventory < 57,000 lb, which is 10% less than the inventory used in the FSAR accident analysis, and therefore is conservative.

(continued)

38

BASES

APPLICABLE SAFETY ANALYSES (continued)	The steam generator level satisfies Criterion 2 of the NRC Policy Statement.
LCO	This LCO is required to preserve the initial condition assumptions of the accident analyses. Failure to meet the maximum steam generator level LCO requirements can result in additional mass and energy released to containment, and excessive cooling (and related core reactivity effects) following an SLB. In addition, feedwater nozzle flooding would impair feedwater heating, and could result in excessive tube to shell temperature differentials and excessive tubesheet temperature gradients.
APPLICABILITY	In MODES 1 and 2, a maximum steam generator water level is required to preserve the initial condition assumption for steam generator inventory used in the steam line failure accident analysis (Ref. 1). In MODE 3, limits on RCS boron concentrations will prevent a return to criticality in the event of an SLB. In MODES 4, 5, and 6, the water in the steam generator has a low specific enthalpy; therefore, there is no need to limit the steam generator inventory when the unit is in this condition.
ACTIONS	<u>A.1</u> With the steam generator level in excess of the maximum limit, action must be taken to restore the level to within the bounds assumed in the analysis. To achieve this status, the water level is restored to within the limit. The 15 minute Completion Time is considered to be a reasonable time to perform this evolution. <u>B.1</u> If the water level in one or more steam generators cannot be restored to less than or equal to the maximum level in

(continued)

BASES

ACTIONS	B.1 (continued) Figure 3.7.18-1, the unit must be placed in a MODE that minimizes the accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours. The allowed Completion Time is 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 3.7.18.1 This SR verifies the steam generator level to be within acceptable limits. The 12 hour Frequency is adequate because the operator will be aware of unit evolutions that can affect the steam generator level between checks. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to steam generator level status.
REFERENCES	1. FSAR, Section [15.4.4].