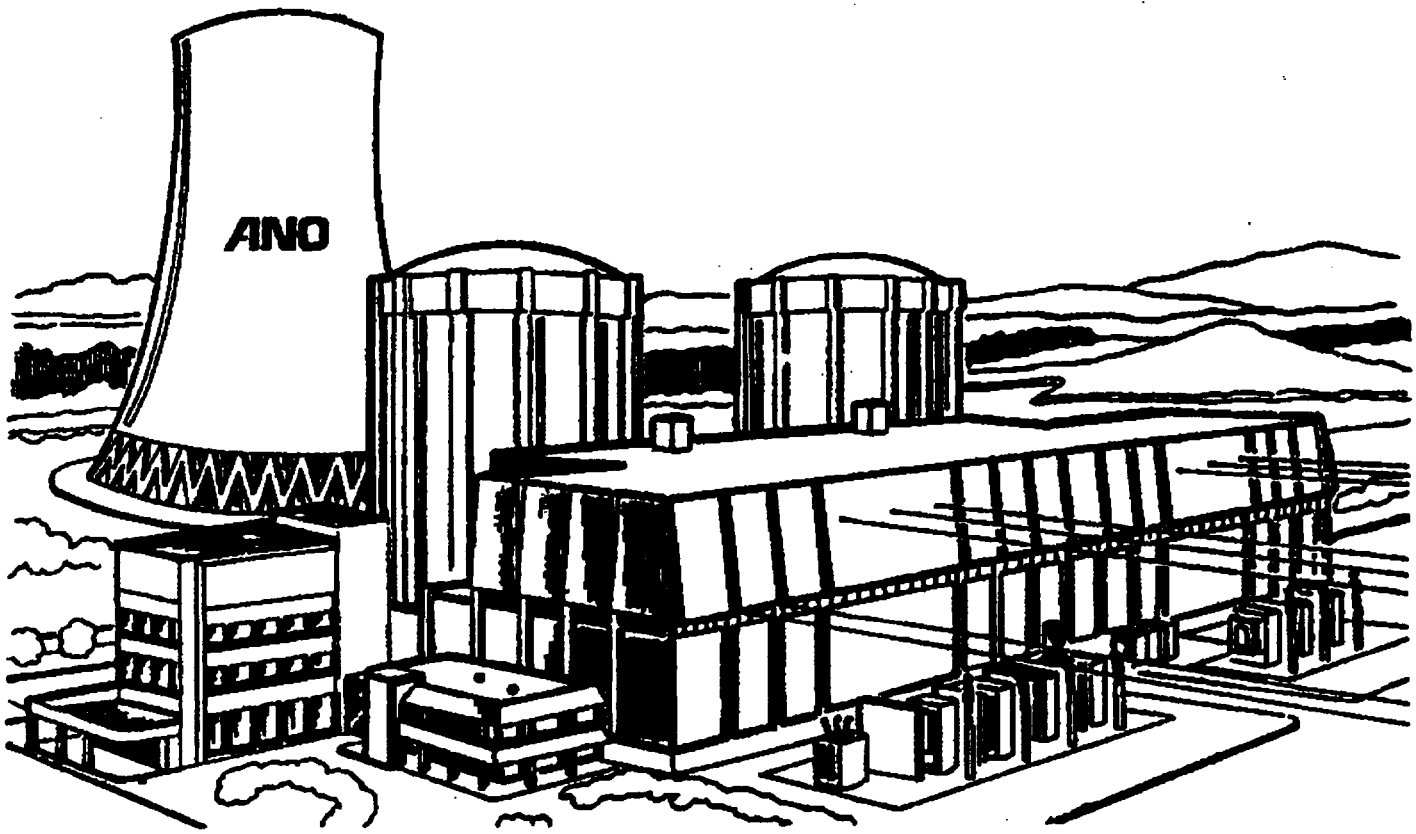
ARKANSAS NUCLEAR ONE - UNIT 1

IMPROVED TECHNICAL SPECIFICATIONS SUBMITTAL



Section 3.6, "Reactor Building Systems" Supplement 1

Entergy

September 28, 2000

Entergy

Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72801
Tel 501-858-4888

Craig Anderson
Vice President
Operations ANO

September 28, 2000

1CAN090007

U. S. Nuclear Regulatory Commission
Document Control Desk
Mail Station OP1-17
Washington, DC 20555

Subject: Arkansas Nuclear One - Unit 1
Docket No. 50-313
License No. DPR-51
Arkansas Nuclear One - Unit 1 - Reply To Request For Additional Information (RAI) RE: Improved Technical Specification Section 3.6, "Reactor Building Systems" (TAC No. MA8082)

Gentlemen:

By letter dated January 28, 2000 (1CAN010007), Entergy Operations submitted a license amendment request to convert the Arkansas Nuclear One - Unit 1 (ANO-1) current Technical Specifications (CTS) to an improved Technical Specification (ITS) format similar to NUREG-1430, "Standard Technical Specifications - Babcock & Wilcox Plants," Revision 1, dated April 1995. During a meeting on August 31, 2000, members of the ANO-1 staff and the NRC Technical Specifications Branch discussed the NRC comments on ITS Section 3.6, "Reactor Building Systems," and the ANO-1 resolutions of these comments.

This submittal contains the Entergy Operations responses to the RAIs discussed at the August 31, 2000 meeting. Attachment 1 contains a description of the contents and format of the ITS Section 3.6 supplement package. Attachments 2 and 3 delineate those comments received from the NRC Staff and ANO-1 personnel, respectively, and the associated resolutions of those comments.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on September 28, 2000.

Very truly yours,

CGA/cws
Attachments

A001

cc: Mr. Ellis W. Merschoff (w/o attachments)
Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

NRC Senior Resident Inspector (w/o attachments)
Arkansas Nuclear One
P.O. Box 310
London, AR 72847

Mr. William Reckley (2 copies)
NRR Project Manager Region IV/ANO-1
U. S. Nuclear Regulatory Commission
NRR Mail Stop O-7 D1
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Mr. David D. Snellings (w/o attachments)
Director, Division of Radiation
Control and Emergency Management
Arkansas Department of Health
4815 West Markham Street
Little Rock, AR 72205

Format of Supplement Package

The improved Technical Specification (ITS) supplement package is organized as described below:

TAB ITS

Contains the proposed ITS Limiting Conditions for Operation (LCOs).

TAB ITS Bases

Contains the proposed ITS Bases

TAB Current Technical Specification (CTS) Markup

Contains annotated copies of the CTS pages which show the disposition of existing requirements into the proposed ITS. The pages are arranged in ITS order. The upper right hand corner of the CTS page is annotated with the ITS Specification number to which the CTS page applies. Items on the CTS page that are addressed in other proposed ITS Sections (or Specifications within the Section) are annotated with the appropriate location.

Where a proposed ITS requirement differs from a CTS requirement, individual details of the CTS revision are annotated with alpha-numeric designators which relate to the appropriate Discussion of Change (DOC). The DOC provides a concise justification for the change. The DOCs are located directly preceding the CTS Markup in each Section or sub-Section. The alpha-numeric designators also relate to the evaluations supporting a finding of No Significant Hazard Consideration (NSHC).

The CTS pages in the Section packages reflect License Amendments issued as of the date of the submittal letter, and License Amendment Requests described in Attachment 2 to the submittal letter.

The DOCs are numbered sequentially within each letter category for each ITS Section or sub-Section. The proposed changes for each CTS requirement are separated into the following categories:

Designator	Category
A	ADMINISTRATIVE - changes to the CTS that result in no additional or reduced restrictions or flexibility. These changes are supported in aggregate by a single NSHC.

M TECHNICAL CHANGES - MORE RESTRICTIVE - changes to the CTS that result in added restrictions or reduced flexibility. These changes are supported in aggregate by a single NSHC.

L TECHNICAL CHANGES - LESS RESTRICTIVE - changes to the CTS that result in reduced restrictions or added flexibility. Each corresponding evaluation is supported by a corresponding evaluation supporting a finding of NSHC.

LA TECHNICAL CHANGES - REMOVAL OF DETAIL - changes to the CTS that eliminate detail and relocate the detail to a licensee controlled document. Typically, this involves details of system design and function, or procedural detail on methods of conducting a surveillance. These changes are supported in aggregate by a single NSHC.

R RELOCATED SPECIFICATIONS - changes to the CTS that encompass the requirements that do not meet the selection criteria of 10 CFR 50.36(c)(2)(ii). These changes are supported in aggregate by a single NSHC.

The CTS Bases pages are replaced in their entirety. A single DOC justifies the replacement.

TAB <u>NSHC</u>

Contains evaluations required by 10 CFR 50.91(a) supporting a finding of No Significant Hazard Consideration (NSHC). Generic evaluations for a finding of NSHC have been written for each category of changes except Category "L." The evaluations supporting a finding of NSHC are ordered as follows: A, M, LA, R, and L. Each evaluation is annotated to correspond to the DOC discussed in the NSHC. The generic NSHC evaluations for Category A, M, and R changes are located in the Split Report section.

TAB <u>NUREG Markup</u>

Contains annotated copies of the applicable NUREG-1430, Revision 1, LCOs which show how the proposed ITS LCO differs from the NUREG LCO. Where a proposed ITS LCO differs from the NUREG LCO, individual details of the change are annotated with numeric designators which relate to the appropriate Discussion of Difference (DOD). The DOD provides a concise justification for the change. The LCO DODs are located directly preceding the associated markup for each Section or sub-Section.

TAB Bases Markup

Contains annotated copies of the applicable NUREG-1430, Revision 1, Bases which show how the proposed ITS Bases differ from the NUREG Bases. Where a proposed ITS Bases requirement differs from the NUREG Bases, individual details of the change are annotated with numeric designators which relate to the appropriate DOD. The DOD provides a justification for the change. The DODs are located directly preceding the associated markup of the NUREG Limiting Conditions for Operation for each Section or sub-Section.

Existing ANO-1 License Amendment Requests (LARs) Incorporated in this supplement

There are no new LARs incorporated in this ITS Section 3.6 supplement. Our letter dated January 28, 2000, showed an LAR dated August 6, 1998 as affecting CTS pages 37 and 39. These CTS pages appear among the CTS markup pages for Section 3.6. This LAR was approved as Amendment 205 to the ANO-1 CTS. Discussions of this LAR have been deleted from the package, as described by comment ANO-71.

Disposition of Generic Changes

In addition to those generic changes shown as incorporated in our letter dated January 28, 2000, one additional generic change has been incorporated in this supplement. TSTF-52, Revision 3 was approved too late in the ITS development cycle for inclusion, but is now shown as incorporated by DOD-2.

List of Beyond Scope Items

In addition to the beyond scope issues discussed in our letter dated January 28, 2000, this supplement presents two additional issues.

1) Reactor Building Purge Valves

ANO-1 does not utilize resilient seated purge valves in its reactor building purge isolation system. However, the CTS 4.26 Bases indicate that the test frequency for the installed valves is based on Generic Issue B-20, "Containment Leakage Due to Seal Deterioration." Since the installed purge isolation valves are not resilient seated, ANO-1 has not incorporated NUREG-1430 SR 3.6.3.6 in the proposed ITS, proposing that this testing be controlled under the provisions of ITS 5.5.16, " Reactor Building Leak Rate Testing Program." Since the CTS Bases conflict with the installed equipment, and since the change is not in accordance with NUREG-1430, this item has been determined to be beyond the scope of ITS conversion. This change is discussed in DOC-L24 and DOD-7.

2) Reactor Building Lower Pressure Limit

ANO-1 CTS 3.6.4 provides a lower limit for reactor building pressure of 5.5 inches Hg. This corresponds to a pressure of approximately -2.7 psig. ANO proposes to incorporate a value of -1.0 psig as the lower reactor building pressure limit in ITS 3.6.4. NUREG-1430 shows this value as a bracketed value, indicating that the plant specific value should be used and the associated Bases describe this value as preserving "the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break." The Bases also state that the limits "prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of an inadvertent actuation of the Containment Spray System." Since an input to the ANO-1 LOCA analysis assumes that reactor building pressure is at 13.7 psia (-1.0 psig), this change is necessary to ensure the safety analysis is protected. Further justification can be found in DOC M14.

Resolution of NRC Comments and ANO-1 Initiated Changes

Attachment 2 provides a listing of all comments on ITS Section 3.6 received as a result of NRC review and the ANO-1 resolutions of these comments. Attachment 3 provides a list of changes to ITS Section 3.6 as a result of the incorporation of comments received from the ANO-1 staff. In both cases, each comment is assigned a unique identifying number such as 3.6.1-1, for an NRC generated comment, or ANO-71, for an ANO-1 generated comment. This identifying number also appears in the left hand margin on each page of the submittal package that was revised as a result of the comment. Each comment response details the location of the necessary changes.

NRC Comment Resolution
ITS Section 3.6: "Reactor Building Systems"

Comment **3.6. 1-1**
DOC A1
DOC A18
DOC A19
DOC A20
DOC LA 1
CTS 1.7
ITS 3.6.1, 3.6.2, 3.6.3 and Associated Bases

CTS 1.7 defines REACTOR BUILDING integrity. A markup of CTS 1.7 shows that portions of CTS 1.7.a, 1.7.b, 1.7.c, and 1.7.e are relocated to ITS B.3.6.1 Bases - LCO, ITS 3.6.2 Bases - LCO, and ITS B3.6.3 Bases - LCO and the relocation is justified by DOC LA1. The rest of CTS 1.7 is covered by DOCs A1, A18, A19, and A20. These DOCs state that the balance of CTS 1.7 is deleted, is incorporated into ITS 3.6.1 or relocated to other LCOs in ITS 3.6. These changes are not entirely correct with regards to ITS 3.6.1. CTS 1.7 is relocated virtually intact to ITS 3.6.1 Bases BACKGROUND which makes the whole change with regards to ITS 3.6.1 a Less Restrictive (LA) change. See Comment Numbers 3.6.1-2. Comment: Revise the CTS markup and the discussions and justifications associated with DOC LA1 to show that CTS 1.7 is relocated to ITS B3.6.1 Bases - BACKGROUND. See Comment Numbers 3.6.1-2.

Response
1) CTS page 5 m/u revised to show 1.7.d also relocated to ITS 3.6.1 Bases Background discussion.
2) DOC LA1 revised to show 1.7a, b, c, & d relocated to ITS 3.6.1 Bases Background.

Comment **3.6.1-2**
DOC A18
DOC LA 1
JFD 2
CTS 1.7.a
ITS B3.6.1 Bases - BACKGROUND and LCO

CTS 1.7 defines REACTOR BUILDING integrity. A markup of CTS 1.7 is provided in the CTS markup of CTS 3.6. DOC LA 1 states that the definition requirements have been relocated to ITS B3.6.1 Bases - LCO. This justification is not entirely correct. CTS 1.7.a states that "The equipment hatch is closed and sealed..." ITS B3.6.1 Bases BACKGROUND states the following: "To maintain this leak tight barrier :c. The equipment hatch is closed; and ..." The requirement for sealing the equipment hatch has not been included here, yet it is included in ITS B3.6.1 Bases - LCO. No justification is provided for this difference. Comment: Correct this discrepancy.

Response
1) ITS m/u page B3.6-2 revised to include "and sealed" in the Background discussion.
2) Proposed ITS B3.6.1 Background revised to include "and sealed."

Note: Justification for Difference (JFD) and Discussion of Difference (DOD) are used interchangeably.

Comment **3.6.1-3**
DOC A20
DOC LA1
JFD 2
JFD 7
CTS 1.7.e
CTS 4.4.1
CTS 4.26.2
STS SR 3.6.1.1 and Associated Bases
ITS SR 3.6.1.1 and Associated Bases

CTS 1.7.e, 4.4.1 and 4.26.2 require leak rate testing in accordance with the Reactor Building Leakage Rate Testing Program which is based on the requirements of 10 CFR 50 Appendix J, Option B. STS SR 3.6.1.1 requires the visual examination and leakage rate testing be performed in accordance with 10 CFR 50 Appendix J as modified by approved exemptions. ITS SR 3.6.1.1 modifies STS SR 3.6.1.1 to conform to CTS 1.7.e and 4.4.1 as modified in the CTS markup. See Comment Numbers 3.6.1-4 and 3.6.3-4 for concerns with the leak rate testing of reactor building purge valves. The STS is based on Appendix J, Option A while the CTS and ITS are based on Appendix J, Option B. Changes to the STS with regards to Option A versus Option B are covered by a letter from Mr. Christopher 1. Grimes to Mr. David J. Modeen, NEI, dated 11/2/95 and TSTF-52 as modified by staff Comments of 10/96,12/98 and 1/2000. While the ITS SR 3.6.1.1 differences from STS SR 3.6. 1.1 are in conformance with the letter and TSTF-52 as modified by staff Comments, the changes to the ITS Bases as well as ITS 3.6.2 and ITS 3.6.3 and their associated Bases are not in conformance with the letter, TSTF-52 as modified by the staff and the CTS. See Comment Numbers 3.6.2-2. and 3.6.3-3 Comment: Licensee should revise its submittal to conform to the 11/2/95 letter and TSTF-52 modified by the staff. See Comment Numbers 3.6.1-4, 3.6.2-2, 3.6.3-3 and 3.6.3-4.

Response The following changes have been made to reflect TSTF-52, Rev 3 incorporation

1) DOD-2 revised .
2) ITS m/u page B3.6-2 Applicable Safety Analyses discussion revised.
3) Proposed ITS B3.6.1 Applicable Safety Analyses discussion revised.
4) ITS m/u page B3.6-1 Background discussion revised.
5) Proposed ITS B3.6.1 Background discussion revised.
6) Insert B3.6-3A revised.
7) ITS m/u page B3.6-4 SR 3.6.1.1 revised.
8) Inserts B3.6-4A, B, C, and D deleted.
9) Proposed ITS B3.6.1 SR 3.6.1.1 revised.
10) Insert B3.6-3A added Reference 5 for B3.6.1 LCO discussion.
11) ITS m/u page B3.6-5 revised Reference 1 to include Option B and inserted Reference 5 for 10CFR 50.55a(g)(6)(ii)(B).
12) ITS m/u page B3.6-7 Applicable Safety Analyses discussion revised to show DOD-43 applicable.
13) Proposed ITS B3.6.1 LCO discussion and References revised to reflect changes.

Note: Justification for Difference (JFD) and Discussion of Difference (DOD) are used interchangeably.

Comment **3.6.1-4**
DOC A20
DOC LA 1
JFD 2
JFD 7
CTS 1.7.e
CTS 4.26.2
STS B3.6.1 Bases - LCO and SR 3.6.1.1
ITS B3.6.1 Bases - LCO and SR 3.6-1.1

ITS B3.6.1 Bases - LCO and SR 3.6.1.1 makes a number of changes to the wording of the corresponding STS B3.6.1 Bases Sections with regards to the applicability of the reactor building purge valves leakage testing. In light of Comment Number 3.6.3-4. which requires the retention of CTS 4.26.2 in ITS 3.6.3, ITS B3.6.1 Bases - LCO and SR 3.6.1.1 need to be revised to reflect this change to the ITS. Comment: Revise the ITS Bases markup to reflect the changes associated with Comment Number 3.6.3-4.

Response
1) DOC L24 has been written to provide details on the ANO-1 installed reactor building purge isolation valves.
2) CTS m/u page 110z revised to show DOC L24 applicable to CTS 4.26.2.
3) NSHC 3.6 L24 written to address DOC L24.

Comment **3.6.1-5**
CTS 1.7.d
STS B3.6.1 Bases - BACKGROUND
ITS B3.6.1 Bases - BACKGROUND
ITS B3.6.3 Bases - BACKGROUND

STS, B3.6.1 Bases - BACKGROUND states the following: "To maintain this leak tight barrier: "a. All penetrations ... either: 2. Closed by... de-activated automatic valves secured in their closed position." ITS B3.6.1 Bases - BACKGROUND changes this sentence to delete the word "secured" and justifies the change as an editorial change. While the change brings the Bases into conformance with CTS 1.7.d, it conflicts with ITS B3.6.3 Bases - BACKGROUND which states that the de-activated automatic valves are secured in their closed position. Comment: Correct this discrepancy.

Response
1) ITS m/u page B3.6-2 Background revised to remove editorial change and retain NUREG-1430 wording.
2) Proposed ITS B3.6.1 Background revised to incorporate change.

Note: Justification for Difference (JFD) and Discussion of Difference (DOD) are used interchangeably.

Comment **3.6.2-1**
DOC A8
CTS 3.6.1
ITS 3.6.2 ACTION Notes 1, 2, and 3 and Associated Bases

CTS 3.6.1 is modified to add ITS 3.6.2 ACTION Notes 1, 2, and 3. This change is characterized as an Administrative change (DOC A8). DOC A8 states that these Notes "...are interpreted to be presently available in the CTS." While the staff may agree that Note 3 may be currently interpreted from structure and wording of the CTS to be presently available in the CTS, it does not see how the same conclusion can be reached for Notes 1 and 2. The CTS requires at least one door in each airlock to be locked closed and sealed during repair. Note 1 and its associated Bases allows the OPERABLE airlock door to be open and unlocked for repair purposes which the CTS would not. Thus, this aspect of the change would be a Less Restrictive (L) change. In addition the CTS does not allow for separate condition entry. If an airlock is inoperable one enters CTS 3.6.1. if a door on the other airlock becomes inoperable before the first door is restored to OPERABLE status, one does not restart the clock as in the ITS, but continues with the ACTION as written. Thus, the addition of both ITS 3.6.2 ACTION Notes 1 and 2 is a Less Restrictive (L) change. See Comment Number 3.6.2-4. Comment: Revise the CTS markup and provide a discussion and justification for this Less Restrictive (L) change. See Comment Number 3.6.2-4.

Response
1) CTS insert page 54A revised to show addition of 3.6.2 Action Notes 1 and 2 as less restrictive.
2) DOC A-8 revised to remove Notes 1 and 2 from the discussion.
3) DOC L-21 written to address addition of 3.6.2 Action Notes 1 and 2.
4) NSHC 3.6 L21 written to address DOC L-21.

Comment **3.6.2-2**
DOC A20
DOC LA1
JFD 2
STS SR 3.6.2.1 and Associated Bases
ITS SR 3.6.2.1 and Associated Bases

See Comment Number 3.6.1-3. Comment: See Comment Number 3.6.1-3.

Response The following changes were made to incorporate TSTF-52, Rev 3:
1) ITS m/u page B3.6-7 Applicable Safety Analyses and LCO discussions revised.
2) ITS m/u page B3.6-12 SR 3.6.2.1 Bases revised.
3) Proposed ITS B3.6.2 Applicable Safety Analyses, LCO, and SR 3.6.2.1 revised.

Comment **3.6.2-3**
DOC M2
CTS 3.6.1
STS 3.6.2 ACTIONS
ITS 3.6.2 ACTIONS

CTS 3.6.1 specifies the ACTIONS to follow for an inoperable reactor building. DOC M2 provides a justification for changing "be in Hot Standby" to "be in MODE 3." However, DOC M2 only addresses this change with regards to ITS 3.6.1 Condition B. No mention is made with regards to ITS 3.6.2 Condition D. Comment: Revise DOC M2 to make it applicable to ITS 3.6.2.

Response DOC M2 revised to clearly show NUREG 3.6.1 Condition B, 3.6.2 Condition D, and 3.6.3 Condition D are addressed.

Comment **3.6.2-4**
JFD 24
STS 3.6.2 ACTION Note 1, and REQUIRED ACTION A Note 2
ITS 3.6.2 ACTION Note 1, Required Action A Notes and Associated Bases

STS/ITS 3.6.2 ACTION Note 1 allows entry and exit through an inoperable airlock to perform repairs on the affected airlock components. STS 3.6.2 Required Action A Note 2 restricts this entry and exit if both air locks are inoperable. ITS 3.6.2 Required Action A does not contain this Note. JFD 24 justifies the deletion on the premise that STS 3.6.2 Required Action A Note 2 conflicts with the allowance in STS 3-6.2 ACTION Note 1 which allows unlimited entry and exit. JFD 24 also states that Note 2 would be inconsistent with the requirements inferred by the CTS. With regards to the latter item, Note 2 is not inconsistent, the CTS would limit entry and exit if two airlock doors were inoperable as discussed in Comment Number 3.6-2.1. In the former case there is no conflict, it just limits the time that entry and exit can be done when both air locks are inoperable. In addition, based on the justification the change would be considered as a generic change. Comment: Delete this generic change. See Comment Number 3.6.2-1.

Response
1) CTS m/u insert page 54A revised to show ITS 3.6.2 Condition A Note 2 added as a less restrictive.
2) DOC L21 written to address this less restrictive change.
3) NSHC 3.6 L21 written to address DOC L21.
4) ITS m/u page 3.6-3 revised to retain Note 2.
5) Revised DOD-24 to state "Not used."
6) Proposed ITS 3.6.2 Condition A Required Action Notes revised to incorporate retention of Note 2.
7) ITS m/u page B3.6-9 Actions revised to show retention of Note 2 and deletion of DOD-24.
8) ITS m/u page B3.6-10 revised to show retention of Note 2.
9) Proposed ITS B3.6.2 Actions revised to show retention of Note 2.

Note: Justification for Difference (JFD) and Discussion of Difference (DOD) are used interchangeably.

Comment **3.6.2-5**
JFD 24
STS B3.6.2 Bases - BACKGROUND
ITS B3.6.2 Bases - BACKGROUND

The first sentence of the second paragraph in STS B3.6.2 Bases - BACKGROUND specifies the diameter of the airlock. ITS B3.6.2 Bases - BACKGROUND deletes this item by JFD 24. JFD 24 has nothing to do with this airlock description, it deals with the ACTION Notes. See Comment Number 3.6.2-4. Comment: Provide a discussion and justification for this deletion.

Response
1) ITS m/u page B3.6-6 Background revised to show DOD-42 describes the proposed change.
2) DOD-42 written to discuss deletion of details.

Comment **3.6.2-6**
STS 3.6.2 Bases - C.1, C.2 and C.3
ITS B3.6.2 Bases - C-1, C.2 and C.3

The third paragraph, first sentence in STS B3.6.2 Bases - C.1, C.2 and C.3 states the following: "Additionally, the affected air lock(s) must..." ITS B3.6.2 Bases - C.1, C.2 and C-3 modifies this sentence by deleting the "(s)" from "Air Lock(s)," and justifies the deletion as editorial. Since the Condition deals with the inoperability of one or more air locks, the term "air lock(s)" is correct. Comment: Delete this change.

Response
1) Editorial change removed from ITS m/u page B3.6-11 Required Action C.1, C.2, and C.3 discussion.
2) Proposed ITS B3.6.2 Required Actions revised to incorporate m/u.

Comment **3.6.3-1**
DOC A16
CTS Table 4.1-2. Item 8
ITS SR 3.6.3.5 and Associated Bases

CTS Table 4.1-2, Item 8 requires the automatic actuation or functioning of the RB Isolation Trip System on an 18 month frequency. The corresponding ITS SR is ITS SR 3.6.3.5. While the CTS phrase "functioning test can be interpreted to mean "an actual or simulated actuation signal", the CTS seems explicit in that all automatic RB isolation valves including locked valves must be tested. The ITS exempts valves which are locked, sealed or otherwise secured in position. Thus the ITS is less restrictive than the CTS. Comment: Revise the CTS markup and provide a discussion and justification for this Less Restrictive (L) change.

Response DOC A16 revised to provide further justification for the administrative nature of the change.

Note: Justification for Difference (JFD) and Discussion of Difference (DOD) are used interchangeably.

Comment **3.6.3-2**
DOC A19
CTS 1.7.c
STS B3.6.3 Bases - BACKGROUND and LCO
ITS B3.6.3 Bases - BACKGROUND AND LCO

CTS 1.7.c which refers to all non-automatic RB isolation valves is modified by DOC A19 to include check valves. DOC A19 states that check valves are considered as non-automatic valves. This is incorrect. The staff considers check valves when used as isolation valves as automatic valves. STS B3.6.3 Bases - BACKGROUND states this position and the discussion in STS B3.6.3 Bases - LCO reaffirms it when it differentiates between automatic power operated isolation valves and check valves. The Bases for this position can be found in 10 CFR 50 Appendix A, General Design Criteria 55, 56 and 57 which state that check valves may not be used as one of the automatic isolation valves for certain types of penetrations. Therefore the changes made to CTS 1.7.c. and the first paragraph in ITS B3.6.3 Bases - BACKGROUND with regards to check valves are incorrect. In addition the deletion of the words "power operated" from second paragraph of ITS B3.6.3 Bases - LCO is incorrect. Comment: Revise the CTS/ITS markups and provide the appropriate discussions and justifications to reflect that check valves are automatic valves.

Response
1) CTS m/u page 5 revised to remove "check valves" from text inserted into CTS 1.7.c by DOC A19.
2) DOC A19 revised to delete check valves from the discussion.
3) ITS m/u page B3.6-14 Background revised to remove editorial changes.
4) Propose ITS B3.6.3 Background revised to reflect m/u changes.

Comment **3.6.3-3**
DOC A20
JFD 2
JFD 7
CTS 1.7.e
CTS 4.4.1
CTS 4.26.2
STS SR 3.6.3.6 and Associated Bases

See Comment Numbers 3.6.1-3 and 3.6.3-4. Comment: See Comment Numbers 3.6.1-3 and
3.6.3-4.

Response
1) DOC L24 has been written to provide details on the ANO-1 installed reactor building purge isolation valves.
2) CTS m/u page 110z revised to show DOC L24 applicable to CTS 4.26.2.
3) NSHC 3.6 L24 written to address DOC L24.

Comment **3.6.3-4**
DOC A20
JFD 7
CTS 1.7.c
CTS 3.0.3
CTS 3.6.1
CTS 4.26.2
STS 3.6.3 Conditions A, B, and D, SR 3.6.3.1 SR 3.6.3.6 and Associated Bases
ITS 3.6.3 Conditions A, B, SR 3.6.3.1 and Associated Bases

CTS 4.26.2 specifies that the RB purge valves be leak rate tested at a specified frequency. The CTS markup of CTS 4.26.2 implies that this item is to be relocated to ITS 5.0. This is unacceptable. Amendment Number 185 which implemented the changes associated with 10 CFR 50 Appendix J Option B did not change the requirements or frequencies associated with CTS 4.26.2. In addition the changes to the STS with regards to Option A versus Option B covered by the letter from Mr. Christopher I. Grimes to Mr. David J. Modeen, NEI , dated 11/2/95 and TSTF 52 as modified by staff Comments of 10/96, 12/98 and 1/2000 do not relocate to the Containment Leakage Rate Testing Program the associated leak rate testing surveillance for purge valves (i.e., STS SR 3.6.3.6). The performance based allowance for extending the leakage rate testing frequencies associated with 10 CFR 50 Appendix J was determined by the staff to be not applicable to purge valve leakage testing. Therefore, this specification cannot be relocated to ITS 5.5-16 "Reactor Building Leakage Rate Testing Program". In addition, STS SR 3.6.3.6 and its associated Bases or a modification of the SR to reflect the CTS requirements which specifies that the RB purge valves must be leak tested on a specified frequency must be retained. Furthermore, since the surveillance is to be retained in the ITS, an appropriate ACTION needs to be included. Therefore, STS 3.6.3 Condition D or a modification based on the current ACTIONS for purge valve leakage (i.e., CTS 3.6.1 or CTS 3.0.3) needs to be added. Furthermore, ITS 3.6.3 Conditions A and B and SR 3.6.3.1 may need to be modified to conform to STS 3.6.3 Conditions A and B and SR 3.6.3.1 to reflect these changes. If, however, ITS 3.6.3 Conditions A and B are going to be used as the ACTIONS for a leaking purge valve, then appropriate justifications need to be provided for the change in Completion Times (CTS of 1 hour versus ITS of 4 hours). COMMENT: Revise the CTS/ITS markup and provide the appropriate discussions and justification to reflect the retention of CTS 4.26.2 in the ITS as discussed above.

Response
1) DOC L24 has been written to provide details on the ANO-1 installed reactor building purge isolation valves.
2) CTS m/u page 110z revised to show DOC L24 applicable to CTS 4.26.2.
3) NSHC 3.6 L24 written to address DOC L24.

Comment **3.6.3-5**
DOC M2
CTS 3.6.1
STS 3.6.3 ACTIONS
ITS 3.6.3 ACTIONS

CTS 3.6.1 specifies the ACTIONS to follow for an inoperable reactor building. DOC M2 provides a justification for changing "be in Hot Standby" to "be in MODE 3". However, DOC M2 only addresses this change with regards to ITS 3.6.1 CONDITION B. No mention is made with regards to ITS 3.6.3 CONDITION D. COMMENT: Revise DOC M2 to make it applicable to ITS 3.6.3.

Response DOC M2 revised to clearly show NUREG 3.6.1 Condition B, 3.6.2 Condition D, and 3.6.3 Condition D are addressed.

Comment **3.6.3-6**
DOC M4
CTS 1.7.c
CTS 3.6.5
ITS SR 3.6.3.2, SR 3.6.3.3 and Associated Bases

CTS 3.6.5 verifies all manual RB isolation valves are locked closed prior to criticality following a refueling shutdown. CTS 3.6.5 is modified by DOC M4 to reflect the frequencies specified in ITS SR 3.6.3.2 and SR 3.6.3.3. See Comment Number 3.6.3-7 for concern with regards to the frequency changes. DOC M4 and CTS 3.6.5 address only manual RB isolation valves. ITS SR 3.6.3.2 and SR 3.6.3.3 applies to manual RB isolation valves and blind flanges. Even though CTS 1.7.c addresses blind flanges, the CTS markup does not show the addition of STS SR 3.6.3.2 and SR 3.6.3.3 as it applies to blind flanges. COMMENT: Revise the CTS markup and provide an appropriate discussion and justification for the More Restrictive change of adding the surveillances for blind flanges.

Response
1) CTS m/u page 54 revised to show "blind flanges" inserted into CTS 3.6.5.
2) DOC M4 revised to include discussion of blind flanges.

Comment **3.6.3-7**
DOC M4
JFD 19
CTS 3.6.5
STS SR 3.6.3.4 and Associated Bases
ITS SR 3.6.3.3 and Associated Bases

STS SR 3.6.3.4 specifies a frequency of "Prior to entering MODE 4 from MODE 5 if ... days." ITS SR 3.6.3.3 modifies this frequency to "Once prior to entering...". The addition of the word "once" is considered as a generic change. This generic change was submitted generically before, but was rejected by the staff. COMMENT: Delete this generic change.

Response
1) ITS m/u page 3.6-12 revised to delete "once" from SR 3.6.3.3 Frequency and deleted DOD-19 from m/u.
2) DOD-19 revised to show "Not used."
3) Proposed ITS 3.6.3.3 revised to remove "once" from Frequency.
4) ITS m/u page B3.6-25 revised to delete "once" from SR 3.6.3.3 Frequency and deleted DOD-19 from m/u.
5) Proposed ITS B 3.6.3 SR 3.6.3.3 revised to remove "once" from Frequency.

Comment **3.6.3-8**
DOC M5
CTS 3.0.3
CTS 3.6.1
CTS 3.6.6
ITS 3.6.3 ACTIONS B. C, and D, and Associated Bases

CTS 3.6.1 and 3.6.6 are modified by DOC M5 to add ITS 3.6.3 ACTIONS B and C. DOC M5 states that the change is More Restrictive even though the CTS has these implied conditions. The staff disagrees. With the implied CTS conditions the changes are Administrative and Less Restrictive (L). The addition of ACTION B is an Administrative change. With two valves in a penetration inoperable, CTS 3.6.6 cannot be met (other valve not OPERABLE) therefore either CTS 3.0.3 or 3.6.1 is entered. This would be the equivalent of ITS 3.6.3 ACTIONS B and D. Thus the change is an Administrative change. The addition of ITS 3.6.3 ACTION C is a Less Restrictive (L) change. Using the same argument CTS 3.6.6 cannot be met (No second valve OPERABLE), therefore CTS 3.0.3 or 3.6.1 is entered. The change is a Less Restrictive (L) change because the Completion Time goes from 1 hour to 72 hours. COMMENT: Revise the CTS markup and provide the appropriate discussions and justifications for this Administrative and Less Restrictive (L) changes.

Response
1) CTS m/u page 55 revised to reflect ITS 3.6.3 Condition B and Notes addition as A5 and ITS 3.6.3 Action C and Notes addition as L22.
2) Revised DOC A5 to include discussion of ITS 3.6.3 Condition B and Notes.
3) DOC L22 written to address addition of ITS 3.6.3 Action C and Notes as a less restrictive change.
4) NSHC written to address L22.

Note: Justification for Difference (JFD) and Discussion of Difference (DOD) are used interchangeably.

Comment **3.6.3-9**
DOC M5
JFD 6
CTS 3.6
STS 3.6.3 ACTION C and Associated Bases
ITS 3.6.3 ACTION C and Associated Bases

STS 3.6.3 ACTION C specifies the required ACTIONS to be taken for an inoperable containment isolation valve in a penetration flow path with only one containment isolation valve and a closed system. STS 3.6.3 ACTION C has been modified by TSTF 30 Rev. 3 to extend the Completion Time from 4 hours to 72 hours. This modification in the ITS is in accordance with TSTF-30 which is acceptable. However, the Bases changes are not in accordance with TSTF-30 Rev. 3. COMMENT: Licensee to update submittal to be in accordance with TSTF-30 Rev. 3 or provide additional justification for the deviations.

Response
1) ITS m/u insert B3.6-21A written to provide details of the design of the only closed system to which the actions of 3.6.3 Condition C apply.
2) ITS m/u page B3.6-21 revised to show INSERT B3.6-21A.
3) DOD 6 revised to reflect the content of ITS m/u INSERT B3.6-21A.
4) Proposed ITS B3.6.3 Required Action C.1 and C.2, 2nd paragraph, revised to reflect change.

Comment **3.6.3-10**
DOC L12
CTS 3.6
ITS SR 3.6.3.1

CTS 3.6 is modified by CTS Insert 54A. CTS Insert 54A consists of two pages which affect the CTS markup for ITS 3.6.3. One page shows DOC L12 as applying to ITS SR 3.6.3.1 and SR 3.6.3.2 Notes. The other page shows DOC L12 as applying to ITS SR. 3.6.3.2 and SR 3.6.3.3 Notes. ITS SR 3.6.3.1 does not contain any Notes. COMMENT: Correct this discrepancy.

Response CTS insert page 54a dated 7/8/98 was a previous version that was included in error. The correct page was dated 1/28/00. The outdated page has been removed from the package.

Comment **3.6.3-11**
DOC LA1
JFD 35
CTS 1.7.c, 1.7d and 3.6.6
ITS 3.6.3 ACTION Note 1 and Associated Bases

The CTS markup of CTS 1.7 shows that CTS 1.7.c is the Bases for ITS 3.6.3 ACTION Note 1. This is incorrect. It is stated in JFD 35 that the phrase "closed as required" is the basis for this addition. The staff disagrees. The phrase "closed as required" does not imply that the manual valves may be opened intermittently under administrative controls. The staff interprets the phrase to mean that only those manual valves that are required to be closed are closed and are to remain closed. In addition, based on CTS 1.7.d and 3.6.6 it does not seem that closed deactivated automatic valves can be opened. The ITS through ITS 3.6.3 ACTION Note 1 would allow this. COMMENT: Revise the CTS markup and provide a discussion and justification for the Less Restrictive (L) change of adding ITS 3.6.3 ACTION Note 1.

Response
1) DOC A22 written to address opening manual valves under administrative controls.
2) CTS m/u page 5 revised to show DOC A22 applicable to CTS 1.7.c.

Comment **3.6.3-12**
JFD 7
STS 3.6.3 ACTION Note 4

ITS 3.6.3 does not include STS 3.6.3 ACTION Note 4. The justification used for deleting this Note - JFD7 - states that the Note is only applicable to purge valve leakage and since ITS 3.6.3 does not contain a purge valve leakage surveillance the Note is not needed. This is incorrect. Based on this justification alone because of Comment Number 3.6.3-4, the Note needs to be retained. However, regardless of the resolution of Comment Number 3.6.3-4, the Note still needs to be retained in the ITS. The reason for this is that the ITS does not allow for cascading through TS. Even though the ACTIONS of ITS 3.6.3 are met for inoperable isolation valve(s), there may be leakage from the inoperable valve or the valve used to isolate the penetration which causes the overall containment leakage rate to exceed acceptance criteria. This Note provides directions on how to address this situation. COMMENT: Revise the CTS/ITS markups and provide an appropriate discussion and justification for this change.

Response
1) CTS m/u insert page 54A revised to show ITS 3.6.3 Action Note 4 addition as A1.
2) ITS m/u page 3.6-7 revised to retain NUREG 3.6.3 Actions Note 4.
3) Proposed ITS 3.6.3 Actions revised to retain Note 4.
4) DOD-7 revised to delete discussion of NUREG 3.6.3 Action Note 4 deletion.

Comment **3.6.3-13**
JFD 7
STS B3.6.3 Bases - APPLICABLE SAFETY ANALYSES
ITS B3.6.3 Bases - APPLICABLE SAFETY ANALYSES

The fourth paragraph of STS B3.6.3 Bases - APPLICABLE SAFETY ANALYSES discusses the single failure criterion and common mode failures with regards to containment isolation valves. ITS B3.6.3 Bases - APPLICABLE SAFETY ANALYSES deletes this paragraph. The justification used for this deletion is JFD 7. JFD 7 only applies to and discusses changes in the ITS as they affect RB purge valves, not RB isolation valves. COMMENT: Provide a discussion and justification for this deletion.

Response The fourth paragraph of the NUREG-1430 Applicable Safety Analyses discussion provides additional details of the design of the containment purge isolation valves and does not provide a general discussion of the design of containment isolation valves.

1) DOC L24 has been written to provide details on the ANO-1 installed reactor building purge isolation valves.
2) CTS m/u page 110z revised to show DOC L24 applicable to CTS 4.26.2.
3) NSHC 3.6 L24 written to address DOC L24.

Comment **3.6.3-14**
JFD 7
STS 3.6.3 Bases - APPLICABLE SAFETY ANALYSES
ITS B3.6.3 Bases - APPLICABLE SAFETY ANALYSES

The fifth paragraph of STS B3.6.3 Bases - APPLICABLE SAFETY ANALYSES deals with why certain containment purge valves are required to be sealed closed during MODES 1 though 4 and how the single-failure criterion applies in these cases. ITS B3.6.3 Bases - APPLICABLE SAFETY ANALYSES deletes this paragraph. Based on ITS B3.6.3 Bases discussions, the SRs associated with RB purge valves, and the discussion provided in JFD 7 which is used to justify deletion of the paragraph, it would seem that the subject matter is applicable to the ANO-1 RB purge valves and that the paragraph should be retained. COMMENT: Provide a discussion and justification showing why this paragraph is not applicable to ANO-1.

Response
1) ITS m/u page B3.6-17 revised to retain an Applicable Safety Analyses discussion of single-failure with respect to purge valves.
2) Proposed ITS B3.6.3 Applicable Safety Analyses revised to reflect change.
3) DOD-7 revised to discuss changes from NUREG-1430 wording.

Comment 3.6.3-15
JFD 30
STS B3.6.3 Bases - LCO
ITS SR 3.6.3.4 and B3.6.3 Bases - BACKGROUND, APPLICABLE SAFETY ANALYSES, LCO and SR 3.6.3.4

The last sentence of the second paragraph of STS B3.6.3 Bases - LCO states the following: "The valves covered by this LCO are listed along with their associated stroke times in the FSAR (Ref. 4)." ITS B3.6.3 Bases - LCO deletes the phrase "along with their associated stroke times" from this sentence. This change is indicated in the ITS markup as an editorial correction. This is incorrect. The deletion if acceptable should be tied to JFD 30. JFD 30 deletes from the various ITS B3.6.3 Bases Sections references and discussions with regards to RB isolation valve response time. JFD 30 states that the ANO-1 analyses for RB isolation set no specific valves for RB isolation valve response times. Based on this, the various discussions on response times is deleted from the ITS B3.6.3 Bases which is acceptable for those other areas. However, JFD 30 also states that RB isolation "valve response times are maintained in accordance with industry standards for sizing valve operators." This along with ITS SR 3.6.3.4 which requires the RB isolation valve isolation time be verified to be within limits, suggests that there are documents which list the stroke time for all automatic RB isolation valves. Thus, the deleted phrase needs to be retained in ITS B3.6.3 Bases-LCO and the appropriate documents referenced so that the ITS users know where to find this information to verify compliance with the SR. COMMENT: Revise ITS B3.6.3 Bases-LCO to include the phrase "along with their associated stroke time" and the documents that contain the times.

Response
1) ITS m/u page B3.6-17 LCO discussion revised to include location of valve stroke times.
2) DOD-44 written to address location of valve stroke times.
3) Proposed ITS B3.6.3 LCO discussion revised to incorporate change.

Comment 3.6.3-16
JFD 34
STS B3.6.3 Bases-LCO
ITS B3.6.3 Bases-LCO

The second sentence of the first paragraph to STS B3.6.3 Bases-LCO states the following: "The containment isolation valve safety function is... establishing the containment boundary during a DBA." ITS B3.6.3 Bases-LCO deletes the phrase "during a DBA" from this sentence using the justification provided by JFD 34. JFD 34 states that the DBA considers certain conditions as initial conditions. The sentence has nothing to do with initial DBA conditions; it is only stating that the safety function applies during a DBA. By deleting this phrase, the sentence would imply that the safety function is applicable for all conditions. Comment: Delete this change.

Response
1) ITS m/u page B3.6-17 LCO discussion revised to retain "during a DBA."
2) Proposed ITS B3.6.3 LCO discussion revised to reflect change.

Note: Justification for Difference (JFD) and Discussion of Difference (DOD) are used interchangeably.

Comment **3.6.3-17**
JFD 40
STS B3.6.3 Bases-LCO
ITS B3.6.3 Bases-LCO

The third paragraph of STS B3.6.3 Bases-LCO describes the OPERABILITY for normally closed isolation valves. The purpose of the LCO Bases discussion is to describe what constitutes an OPERABLE system, component or structure. ITS B3.6.3 Bases-LCO modifies this paragraph to try to correct what the licensee perceives as an erroneous description. The staff disagrees. The STS paragraph deals with those RB isolation valves, manual or otherwise that during normal operation are closed. This defines for these valves their OPERABILITY. Thus the changes proposed are unacceptable. COMMENT: Delete these changes.

Response
1) ITS m/u page B3.6-17 revised to clarify LCO discussion.
2) DOD-40 revised.
3) Proposed ITS B3.6.3 LCO discussion revised to incorporate change.

Comment **3.6.3-18**
STS B3.6.3 Bases- APPLICABLE SAFETY ANALYSES
ITS B3.6.3 Bases- APPLICABLE SAFETY ANALYSES

The second sentence of the first paragraph in STS B3.6.3 Bases-APPLICABLE SAFETY ANALYSES states the following: "As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment." ITS B3.6.3 Bases - APPLICABLE SAFETY ANALYSES modifies this sentence by substituting "isolation" for "leak tightness." Isolation of the containment does not connote leak tightness; however, containment isolation valve OPERABILITY requires a certain degree of leak tightness in order to meet the requirements of 10 CFR 50 Appendix J. Thus the STS words are correct. COMMENT: Delete this change.

Response
1) ITS m/u page B3.6-16 Applicable Safety Analyses discussion revised to remove editorial change.
2) Proposed ITS B 3.6.3 Applicable Safety Analyses discussion revised to incorporate change.

Comment 3.6.3-19
STS 3.6.3 ACTION Note 1 and Associated Bases
ITS B3.6.3 ACTION Note 1 and Associated Bases

The first paragraph, first sentence of STS B3.6.3 Bases - ACTIONS state the following: "The ACTIONS are modified by a Note allowing penetration flow paths, except-.." ITS B3.6.3 Bases - ACTIONS modifies this sentence by adding "with inoperable reactor building isolation valves" between "path" and ", except...". The intent of the STS Note is to allow any closed containment isolation valve except certain purge valves to be opened under administrative controls and not restrict it to just those valves closed as a result of Required Actions. It also prevents unnecessary entry into the ACTIONS. The proposed change would limit the STS intent. COMMENT: Delete this change.

Response
1) ITS m/u page B3.6-18 ACTIONS discussion revised to delete insertion of "with inoperable reactor building isolation valves."
2) Proposed ITS B3.6.3 Actions discussion revised to reflect change.

Comment 3.6.3-20
STS B3.6.3 Bases - ACTIONS
ITS B3.6.3 Bases - ACTIONS

The second sentence of the first paragraph in STS B3.6.3 Bases - ACTIONS states the following: "The administrative controls consist of stationing a dedicated operator at the valve controls, who is in continuous communication with the control room." ITS B3.6.3 Bases-ACTIONS modifies this sentence by substituting "dedicated individual" for "dedicated operator at the valve controls". This is unacceptable since it changes the intent of the administrative controls. The intent is that the dedicated individual/operator can respond quickly to a condition which requires the valve to be closed. The ITS change would allow the individual/operator to be anywhere in the plant, thus he would not be able to respond quickly to the condition requiring valve isolation- COMMENT: Delete this change.

Response
1) ITS m/u page B3.6-18 Actions revised to retain "at the valve controls," insert a statement on ALARA considerations and show DOD-41 as applicable.
2) DOD-41 written to justify allowing the dedicated individual to stand by in a lower dose area.
3) Proposed ITS B3.6.3 Actions revised to reflect changes.

Comment **3.6.4-1**
DOC M13
JFD 20
CTS 3.6.
STS SR 3.6.4.1 and SR 3.6.5.1
ITS SR 3.6.4.1 and Associated Bases

STS SR 3.6.4.1 verifies the containment pressure is within limits on a frequency of every 12 hours. ITS SR 3.6.4.1 changes the frequency to every 24 hours and is justified in DOC M13 and JFD 20. The justification for the change is based on ANO-1 operating experience related to trending and consistency with the frequency specified in STS SR 3.6.5.5 for containment temperature. With regards to the former justification no data or sufficient information was provided in DOC M13 or JFD 20 to lead to the conclusion that once per 24 hours is an acceptable plant specific change for the SR. As to the latter justification consistency with another STS specification is not adequate justification for a change, and would lead the staff to conclude that the change is generic. COMMENT: Revise the ITS markup to delete this generic frequency change or provide adequate discussion and justification based on current licensing basis, system design, or current operating practice to show that a frequency of 24 hours is plant specific to ANO-1.

Response
1) ITS m/u page 3.6-14 revised to delete 24 hour Frequency and retain NUREG Frequency of 12 hours. DOD-20 also deleted.
2) Proposed ITS 3.6.4.1 revised to reflect change in Frequency.
3) DOD-20 revised to show "Not used."
4) ITS m/u page B3.6-31 revised to delete 24 hour Frequency, Retain NUREG Frequency of 12 hours, and delete DOD-20.
5) Proposed ITS B3.6.4.1 revised to reflect change in Frequency.

Comment **3.6.4-2**
DOC M13
DOC M14
JFD 14
CTS 3.6.4
ITS LCO 3.6.4, SR 3.6.4.1 and Associated Bases

CTS 3.6.4 specifies that the reactor building internal pressure limits shall be between a 5.5 inches Hg vacuum and 3.0 psig. ITS LCO 3.6.4 and SR 3.6.4.1 specify the limits as ≥ 1.0 psig and ≤ 3.0 psig. DOC M14 states that 5.5 inches Hg vacuum is equivalent to -2.7 psig, and that this value conflicts with the value assumed in the ANO-1 ECCS analysis which is -1.0 psig. Even though the change is a More Restrictive change, this is a change in the current licensing basis. The 5.5 inches Hg vacuum was part of the original licensing basis and there must have been a reason this number was selected. See Comment Number 3.6.4-3 for one possible reason. As such, the staff considers this change to be a beyond scope of review item for this conversion. COMMENT: Delete this change. See Comment Number 3.6.4-3 for one possible reason.

Response DOC M14 revised to provide additional detail on ANO-1 analytical basis associated with reactor building lower pressure limit.

Comment **3.6.4-3**
JFD 14
JFD 37
JFD 39
STS B3.6.4 Bases
ITS B3.6.4 Bases

ITS B3.6.4 Bases makes numerous changes to STS B3.6.4 Bases to remove the discussions with regards to inadvertent containment spray issues. These discussions are replaced by discussions related to ANO-1 DBA and ECCS analyses. The justification used to delete the inadvertent containment spray issues - JFD 37 - states that requirements related to containment structural integrity and internal pressure during an inadvertent containment spray actuation are not related to the criteria of 10 CFR 50.36. The inadvertent containment spray actuation as stated in STS B3.6.4 is part of the design basis for containment, and was one of the factors taken into the consideration for specification retention and satisfying the criteria of 10 CFR 50.36. Furthermore, the current pressure limit of 5.5 inches Hg vacuum may have been selected based on this requirement, which would require the retaining of this value. See Comment Numbers 3.6.4-2 and 3.6.5-6. COMMENT: Delete this change, or provide a discussion and justification for this change based on current licensing basis, system design or operating constraints.

Response DOC M14 revised to provide additional detail on ANO-1 analytical basis associated with reactor building lower pressure limit.

Comment **3.6.4-4**
JFD 14
JFD 39
ITS B3.6.4 Bases - APPLICABLE SAFETY ANALYSES

ITS B3.6.4 Bases - APPLICABLE SAFETY ANALYSES - insert B3.6-30C states the following: "The LCO limit of 2.0 psig does not consider instrument uncertainty. The LCO limit of -1.0 psig is considered to be as an indicated valve." No justification or explanation is provided for the 2.0 psig value. Where does this number come from and is it the correct value? In addition, the staff believes that these statements are more appropriate for either the LCO Bases or SR 3.6.4.1 Bases write-ups rather than the APPLICABLE SAFETY ANALYSES. COMMENT: Revise the ITS Bases markup as appropriate and provide the necessary discussions and justification for this change.

Response
1) Insert B3.6-30C revised to correct typo from "2.0" to "3.0."
2) Proposed ITS B3.6.4 Applicable Safety Analyses discussion revised to incorporate change.

Comment **3.6.5-1**
DOC A10
DOC M19
JFD 21
CTS 3.3.1 and 3.3.6
ITS 3.6.5 ACTIONS E, F and Associated Bases

CTS 3.3.6 specifies that if the requirements of CTS 3.3.1 for one OPERABLE RB spray train and one OPERABLE RB cooling train are not met in MODES 3 and 4, then the plant be in Cold Shutdown (MODE 5) within 72 hours. The CTS markup modifies this ACTION to allow 36 hours to restore the inoperable component to OPERABLE status, before a shutdown is initiated, and requires that the plant be in COLD SHUTDOWN (MODE 5) within the following 36 hours. The CTS markup designates the change of restore within 36 hours as an Administrative change and justifies the change in DOC A10. This is incorrect. Even though the overall time to reach MODE 5 does not change (72 hours), the specifying of time to restore OPERABILITY where none was specified before makes the change a More Restrictive change. COMMENT: Revise the CTS markup and provide a discussion and justification for this More Restrictive change.

Response
1) CTS m/u page 38-2 revised to delete DOC A10 and show change as DOC M19.
2) DOC A10 revised to show "Not used."
3) DOC M19 revised to discuss ITS 3.6.5 Condition E Completion Time.

Comment **3.6.5-2**
DOC L19
CTS 3.3.1.(I), and 3.3.4.(D)
ITS LCO 3.6.5. SR 3.6.5.1, and Associated Bases

CTS 3.3.1.(I) and 3.3.4.(D) requires the valves associated with the RB Spray System and RB Cooling System to be OPERABLE or locked in the engineered safety position. ITS LCO 3.6.5 and 3.6.6 will retain these requirements as a condition of system OPERABILITY. The ITS will allow these valves to be verified OPERABLE by actuation to the correct position or by being locked sealed or otherwise secured in position. According to DOC L19 the expanded options for controlling valve position makes the change a Less Restrictive (L) change. This is not totally correct. As currently written, the CTS would require that all manual valves be locked in position. The ITS based on the wording of ITS SR 3.6.5.1 and SR 3.6.6.1 does not require all manual valves to be locked, sealed or otherwise secured in position. This aspect of the change has not been addressed or justified. See Comment Number 3.6.6-2. COMMENT: Revise the CTS markup to reflect the above discussion and provide additional discussion and justification for this Less Restrictive (L) change. See Comment Number 3.6.6-2.

Response
1) CTS m/u page 37 revised to show incorporation of ITS SR 3.6.5.1 as a more restrictive change.
2) DOC M10 revised to discuss incorporation of ITS SR 3.6.5.1.
3) DOC L19 revised to discuss manual valves not included in ES valve requirements.

Comment **3.6.5-3**
DOC LA3
JFD 21
CTS 4.5.2.2.1
STS B3.6.6 Bases SR 3.6.6.4
ITS SR 3.6.5.4 and Associated Bases

CTS 4.5.2.2.1 verifies RB spray pump OPERABILITY. The CTS markup shows the entire specification as being relocated to the IST program. This is incorrect. The first part of the specification is the basis for ITS SR 3.6.5.4. and should be marked accordingly. In addition, no justification is provided to show that the pump start frequency of "not to exceed 3 months" is the current frequency in the IST program for these pumps, If not, then the ITS frequency of "In accordance with the Inservice Testing Program" is either a More Restrictive or Less Restrictive (L) change depending on whether the current IST frequency is greater than or less than the CTS frequency. In addition the markup of ITS B3.6.5 Bases - SR 3.6.5.4 shows that the STS words "confirms one point on the pump design curve" as being deleted, and that the change is editorial. This is incorrect. The phrase should either be retained or modified to reflect the requirement in CTS 4.5.2.2.1 that "the discharge pressure and flow are within +/-10% of a point on the pump head curve." See Comment Number 3.6.5-4 for additional concerns on ITS SR 3.6.5.4. Comment. Revise the CTS/ITS markups to reflect the above discussion and provide additional discussion and justification for the More or Less Restrictive (L and LA) changes.

Response
1) CTS m/u page 96 revised to show details of test relocated to Bases by DOC LA1.
2) LA1 revised to show CTS 4.5.2.2.1 information on test acceptance relocated to Bases for ITS SR 3.6.5.4.
3) LA3 revised to show current IST frequencies.
4) ITS m/u pages B3.6-42 and B3.6-43 revised to incorporate details of test performance.
5) DOD-45 written to discuss inclusion of test performance details in Bases of ITS SR 3.6.5.4.
6) Proposed ITS B3.6.5.4 revised to incorporate changes.

Comment **3.6.5-4**
DOC LA3
JFD 21
JFD 31
CTS 3.3.1 and 3.3.4
CTS 4.5.2.1 and 4.5.2.2
ITS SR 3.6.5.1 through SR 3.6.5.7 and Associated Bases

CTS 4.5.2.1 and 4.5.2.2 specify the surveillance to be performed on the components associated with the RB Spray and RB Cooling Systems. Based on the wording and structure of the CTS surveillances, the CTS requires that all components - pumps, valves, fans, coolers, etc.. be tested regardless of whether the component is required to be OPERABLE in accordance with CTS 3.3.1 or 3.3.4. The corresponding ITS SRs - SR 3.6.5.1 through SR 3.6.5.7 limit the applicability of the SRs to those systems and components that are required by the ITS LCO 3.6.5. i.e., in MODES 3 and 4 only the components in the required train of RB spray and RB cooling are to be tested. No justification is provided for this Less Restrictive (L) change. COMMENT: Revise the CTS markup and provide a discussion and justification for this Less Restrictive (L) change.

Response
1) DOC A23 written to address performance of SRs when not in the Mode of Applicability.
2) CTS m/u pages 95 and 96 revised by the addition of DOC A23.

Comment **3.6.5-5**
JFD 21
STS 3.7.2 ACTIONS A, B and Associated Bases
ITS 3.6.5 ACTION A, B, C. and D and Associated Bases

ITS B3.6.5 Bases - D.1 states the following: "if the Required Actions and associated Completion Times are not met, the unit must be brought to a MODE in which the LCO, as modified by the Note does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours." These sentences are slightly confusing, particularly when one considers the phrase "as modified by the Note" and whether it applies here or not. It is recommended that since the STS 3.7.2 ACTIONS A and B are similar in structure and intent to ITS 3.6.5 ACTIONS A, B, C, and D, that wording similar to STS B3.7.2 Bases - B.1 be used for the above sentences in ITS B3.6.5 Bases-D.1. COMMENT: Revise the ITS markup to reflect the above discussion.

Response
1) ITS m/u page B3.6-40 Required Action D.1 discussion revised to delete insertion of "as modified by the Note."
2) Proposed ITS B3.6.5 Required Action D.1 discussion revised to reflect change.

Comment **3.6.5-6**
JFD 37
STS B3.6.6. Bases - APPLICABLE SAFETY ANALYSES
ITS B3.6.5 Bases - APPLICABLE SAFETY ANALYSES

ITS B3.6.5 Bases - APPLICABLE SAFETY ANALYSES modifies the wording in STS B3.6.6 Bases - APPLICABLE SAFETY ANALYSES to remove the discussion with regards to inadvertent containment spray issues. The same discussion and requirements imposed by Comment Numbers 3.6.4-2 and 3.6.4-3 are applicable here. COMMENT: See Comment Numbers 3.6.4-2 and 3.6.4-3.

Response DOC M14 revised to provide additional detail on ANO-1 analytical basis associated with reactor building lower pressure limit.

Comment **3.6.5-7**
TS 4.5.2.1.1.a and Associated Bases
ITS SR 3.6.5.5, SR 3.6.5.6 and Associated Bases

CTS 4.5.2.1.1.a specifies that the RB Spray System, system test shall be performed except that the RB inlet valves shall not receive the Actuation Signal to prevent water entering the nozzles. The ITS breaks this CTS surveillance into two surveillances - ITS SR 3.6.5.5 and SR 3.6.5.6. ITS SR 3.6.5.5 verifies that each automatic containment spray valve that is not locked, sealed or otherwise secured in position actuates to its correct position on an actuation signal. (See Comment Number 3.6.5-8 for additional concerns with regards to actuation signal). It is implied from CTS 4.5.2.1.1.a and its associated Bases that these RB inlet valves are automatic valves, but it is not clear. If they are manual valves then there is no problem. However, if these valves are automatic, then there is the concern as to when these valves will be tested per ITS SR 3.6.5.5 since the locked, sealed, and secured exception in the SR could result in the valves never being tested for this SR. The exception from testing of locked, sealed or otherwise secured valves was only intended to apply to those valves that during normal operating conditions are locked, sealed, or otherwise secured in position. COMMENT: Specify whether the RB inlet valve is manual or automatic. If automatic, discuss when and how this valve will be tested in accordance with ITS SR 3.6.5.5. See Comment Number 3.6.5-8.

Response
1) ITS m/u page B3.6-43 SR 3.6.5.5 and SR 3.6.5.6 discussion revised to show INSERT B3.6-43B and DOD-46.
2) ITS INSERT page B3.6-43B written to provide details on performance of SR.
3) DOD-46 written to address insertion of test details.
4) Proposed ITS B3.6.5 SR 3.6.5.5 and SR 3.6.5.6 discussion revised to reflect change.

Comment 3.6.5-8
CTS 4.5.2.1.1.a and 4.5.2.1.2.c.1
ITS SR 3.6.5.5, SR 3.6.5.6, 3.6.5.7 and Associated Bases

CTS 4.5.2.1.1.a and 4.5.2.1.2.c.1 require a system test of the RB Spray System and RB Cooling System, respectively. These specifications specify that a test signal shall be used to demonstrate system actuation. The ITS breaks these CTS surveillances up into three surveillances - ITS SR 3.6-5.5, SR 3.6-5.6, and SR 3.6.5.7, however the ITS tests may be initiated by either an actual or simulated actuation signal. The CTS markup does not show this change "test signal" to "actual or simulated actuation" Using a test signal connotes only a simulated actuation. By adding the words "actual actuation" the change becomes a Less Restrictive (L) change. COMMENT: Revise the CTS markup and provide a discussion and justification for this Less Restrictive (L) change.

Response
1) CTS m/u pages 95 and 96 revised to show incorporation of actual actuation signal as a less restrictive (L23).
2) DOC L23 written to address incorporation of actual actuation signal.
3) NSHC for 3.6 L23 written.

Comment 3.6.5-9
CTS 4.5.2.1.2.b.
ITS SR 3.6.5.2 and Associated Bases

CTS 4.5.2.1.2.b specifies that at least once per 31 days each RB cooling train fan shall be started and operated for at least 15 minutes. The corresponding ITS SR is ITS SR 3.6.5.2. CTS 4.5.2.1.2.b. (1) specifies that for this test, the fan if not already operating shall be started from the control room. ITS SR 3.6.5.2 and its associated Bases does not specify this requirement. COMMENT: Revise the CTS/ITS markup to specify the location of the requirements in CTS 4.5.2.1.2.b.(1). Provide the appropriate discussions and justifications associated with this change.

Response
1) ITS m/u page B3.6-42 SR 3.6.5.2 discussion revised to show insertion of CTS details on performance of SR.
2) CTS m/u page 95 revised to show CTS 4.5.1.2(b)(1) relocated to ITS SR 3.6.5.2 Bases.
3) LA1 revised to show CTS 4.5.1.2(b)(1) relocated to ITS SR 3.6.5.2 Bases.
4) DOD-26 revised to discuss insertion of CTS details on performance of SR.
5) Proposed ITS B3.6.5 SR 3.6.5.2 discussion revised to reflect changes.

Comment **3.6.5-10**
STS B3.6.5 Bases- A, 1 and C.1
ITS B3.6.5 Bases-A.1 and B.1

The last sentence in both STS B3.6.5 Bases A.1 and B.1 states the following: "Refer to Section 1.3 for a more detailed discussion on the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time." ITS B3.6.5 Bases- A.1 and B.1 deletes this statement and justifies it as an editorial correction. This is incorrect. The purpose of the Bases ACTIONS discussions is to provide a justification and purpose for the ACTION, its remedial measures and Completion Times. This sentence by referring to Section 1.3 provides the justification and purpose for this unique Completion Time and thus should be retained. COMMENT: Delete this change.

Response
1) ITS m/u pages B3.6-39 and B3.6-40 Required Action A.1 and B.1 discussions revised to retain Section 1.3 discussion.
2) Proposed ITS B3.6.5 Required Actions A.1 and B.1 revised to reflect changes.

Comment **3.6.6-1**
JFD 16
ITS B3.6.6 Bases - BACKGROUND

The second sentence in ITS B3.6.6 Bases - BACKGROUND, Insert B3.6-45B states the following: "When the valves are open, the sodium hydroxide is ready to be into the RB Spray System headers." The sentence does not make sense and seems to be incomplete. COMMENT: Correct this discrepancy.

Response INSERT B3.6-45B correct by inserting "introduced" in the second sentence.

Comment **3.6.6-2**
DOC 1-19
CTS 3.3.4.(D)
ITS LCO 3.6.6, SR 3.6.6.1, and Associated Bases

See Comment Number 3.6.5-2. COMMENT: See Comment Number 3.6.5-2

Response DOC L19 revised to discuss manual valves not included in ES valve requirements.

Note: Justification for Difference (JFD) and Discussion of Difference (DOD) are used interchangeably.

Comment **3.6.7-1**
DOC LA1
CTS 4.12.1.a and 4.12.1.b.3
ITS B3.6.7 Bases - SR 3.6.7.3

CTS 4.12.1.b.3 verifies the integrity of the electrical heater circuits by performing a resistance to ground test. The surveillance also states that this test is to be performed following the performance of CTS 4.12.1.a. The CTS markup shows this portion (test performance following CTS 4.12.1.a) as being relocated by DOC LA1. DOC LA1 states that this statement is relocated to ITS B3.6.7 Bases - SR 3.6.7.3. ITS B3.6.7 Bases -SR 3.6.7.3 does not contain this statement, and DOC LA1 does not justify the Less Restrictive change associated with the deletion of the phrase. COMMENT: Revise the ITS Bases markup to include this statement or provide a discussion and justification for its deletion.

Response
1) ITS m/u page B3.6-54 revised to show insertion of "following the performance of SR 3.6.7.2" in the SR 3.6.7.3 discussion.
2) Proposed ITS B 3.6.7 SR 3.6.7.3 discussion revised to reflect change.

ANO Comment Resolution
ITS Section 3.6: Reactor Building Systems

Comment **ANO-71**
DOC A21 is no longer required since Amendment 205 was approved by the NRC.

Response
1) CTS m/u pages 37 and 39 revised to delete DOC A21 and to show Amendment number.
2) DOC A21 shown as "Not used."

Comment **ANO-72**
Develop a DOD to discuss the deletion of rod ejection accident from the DBAs that challenge RB Operability.

Response
1) Developed DOD-43.
2) Revised ITS M/U page B 3.6-2 to show inclusion of DOD-43.

Comment **ANO-73**
DOC LA2 shows CTS 4.5.1.1.c being relocated to the IST program. This is inappropriate. Suggest that this level of detail be relocated to the affected SR Bases.

Response
1) Revised CTS m/u page 95 to show relocateion of CTS 4.5.1.1.c to SR 3.6.5.5 and 3.6.5.6 Bases as LA1.
2) Revised ITS m/u page B3.6-43 to show new Insert B3.6-43A and DOD-46.
3) Added ITS m/u insert B3.6-43A with appropriate wording.
4) Revised proposed ITS B3.6.5.5 and B3.6.5.6 to incorporate change.
5) Added DOD-46 to discuss inserted text.

Comment **ANO-74**
DOC LA 3 relocates the CTS 4.4.1.4 allowance to stroke test valves once every 18 months if stroke operation of that valve is not practicable during plant operation to the IST program. This is actually a more restrictive requirement. The ANO-1 IST program requires those valves that cannot be full stroked during plant operation to be full stroked during cold shutdowns. Since the base frequency is once every three months, this type of valve could be required to be stroked once per quarter if cold shutdown were entered that frequently.

Response
1) CTS m/u page 79 has been revised to show portion of CTS 4.4.1.4 deleted by DOC M23.
2) DOC M23 has been written to discuss this deletion of details.
3) DOC LA3 has been revised to delete reference to CTS 4.4.1.4.

3.6 REACTOR BUILDING SYSTEMS

3.6.1 Reactor Building

LCO 3.6.1 The reactor building shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor building inoperable.	A.1 Restore reactor building to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	Perform required visual examinations and leakage rate testing except for reactor building air lock testing, in accordance with the Reactor Building Leakage Rate Testing Program.	In accordance with the Reactor Building Leakage Rate Testing Program

3.6 REACTOR BUILDING SYSTEMS

3.6.2 Reactor Building Air Locks

LCO 3.6.2 Two reactor building air locks shall be OPERABLE.

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

ACTIONS

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

1. Entry and exit is permissible to perform repairs on the affected air lock components.
2. Separate Condition entry is allowed for each air lock.
3. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Reactor Building," when air lock leakage results in exceeding the overall reactor building leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more reactor building air locks with one reactor building air lock door inoperable.	----------NOTES---------- 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls if both air locks are inoperable. ---------- A.1 Verify the OPERABLE door is closed in the affected air lock. <u>AND</u>	1 hour

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Lock the OPERABLE door closed in the affected air lock. AND A.3 ----------NOTE---------- Air lock doors in high radiation areas may be verified locked closed by administrative means. ---------- Verify the OPERABLE door is locked closed in the affected air lock.	24 hours Once per 31 days
B. One or more reactor building air locks with reactor building air lock interlock mechanism inoperable.	----------NOTES---------- 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit of the reactor building is permissible under the control of a dedicated individual. ---------- B.1 Verify an OPERABLE door is closed in the affected air lock. AND	 1 hour

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Lock an OPERABLE door closed in the affected air lock.	24 hours
	AND		
	B.3	----------NOTE---------- Air lock doors in high radiation areas may be verified locked closed by administrative means. ---------- Verify an OPERABLE door is locked closed in the affected air lock.	Once per 31 days
C. One or more reactor building air locks inoperable for reasons other than Condition A or B.	C.1	Initiate action to evaluate overall reactor building leakage rate per LCO 3.6.1.	Immediately
	AND		
	C.2	Verify a door is closed in the affected air lock.	1 hour
	AND		
	C.3	Restore air lock to OPERABLE status.	24 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	----------NOTE---------- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1. ---------- Perform required air lock leakage rate testing in accordance with the Reactor Building Leakage Rate Testing Program.	In accordance with the Reactor Building Leakage Rate Testing Program
SR 3.6.2.2	Verify only one door in the air lock can be opened at a time.	18 months

3.6 REACTOR BUILDING SYSTEMS

3.6.3 Reactor Building Isolation Valves

LCO 3.6.3 Each reactor building isolation valve shall be OPERABLE.

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

ACTIONS

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

1. Penetration flow paths, except for purge valve penetration flow paths, may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for system(s) made inoperable by reactor building isolation valves.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Reactor Building," when isolation valve leakage results in exceeding the overall reactor building leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ----------NOTE---------- Only applicable to penetration flow paths with two reactor building isolation valves. ---------- One or more penetration flow paths with one reactor building isolation valve inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. AND	48 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 ----------NOTES---------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ---------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside the reactor building <u>AND</u> Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside the reactor building
B. ----------NOTE---------- Only applicable to penetration flow paths with two reactor building isolation valves. ---------- One or more penetration flow paths with two reactor building isolation valves inoperable.	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ----------NOTE---------- Only applicable to penetration flow paths with only one reactor building isolation valve and a closed system. ---------- One or more penetration flow paths with one reactor building isolation valve inoperable.	C.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	72 hours
	AND	
	C.2 ----------NOTES---------- 1. Isolation devices in high radiation areas may be verified by use of administrative means. 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means. ---------- Verify the affected penetration flow path is isolated.	Once per 31 days
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours
	AND	
	D.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1	Verify each reactor building purge isolation valve is closed.	31 days

SURVEILLANCE		FREQUENCY
SR 3.6.3.2	----------NOTE---------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ---------- Verify each reactor building isolation manual valve and blind flange that is located outside the reactor building and not locked, sealed, or otherwise secured, and is required to be closed during accident conditions is closed, except for reactor building isolation valves that are open under administrative controls.	31 days
SR 3.6.3.3	----------NOTE---------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ---------- Verify each reactor building isolation manual valve and blind flange that is located inside the reactor building and not locked, sealed, or otherwise secured, and required to be closed during accident conditions is closed, except for reactor building isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days
SR 3.6.3.4	Verify the isolation time of each automatic power operated reactor building isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.5	Verify each automatic reactor building isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	18 months

3.6 REACTOR BUILDING SYSTEMS

3.6.4 Reactor Building Pressure

LCO 3.6.4 Reactor building pressure shall be ≥ -1.0 psig and ≤ +3.0 psig.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor building pressure not within limits.	A.1 Restore reactor building pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Verify reactor building pressure is ≥ -1.0 psig and ≤ +3.0 psig.	12 hours

3.6 REACTOR BUILDING SYSTEMS

3.6.5 Reactor Building Spray and Cooling Systems

LCO 3.6.5 Two reactor building spray trains and two reactor building cooling trains shall be OPERABLE.

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

Only one train of reactor building spray and one train of reactor building cooling are required to be OPERABLE during MODES 3 and 4.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One reactor building spray train inoperable in MODE 1 or 2.	A.1 Restore reactor building spray train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO
B. One reactor building cooling train inoperable in MODE 1 or 2.	B.1 Restore reactor building cooling train to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
C. Two reactor building cooling trains inoperable in MODE 1 or 2.	C.1 Restore one reactor building cooling train to OPERABLE status.	72 hours

D.	Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Be in MODE 3.	6 hours
E.	One required reactor building spray train inoperable in MODE 3 or 4. OR One required reactor building cooling train inoperable in MODE 3 or 4.	E.1	Restore required inoperable train to OPERABLE status.	36 hours
F.	Required Action and associated Completion Time of Condition E not met.	F.1	Be in MODE 5.	36 hours
G.	Two reactor building spray trains inoperable in MODE 1 or 2. OR Any combination of three or more trains inoperable in MODE 1 or 2. OR One required reactor building spray train and one required reactor building cooling train inoperable in MODE 3 or 4.	G.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify each reactor building spray manual, power operated, and automatic valve in each required flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.5.2	Operate each required reactor building cooling train fan unit for ≥ 15 minutes.	31 days
SR 3.6.5.3	Verify each required reactor building cooling train cooling water flow rate is ≥ 1200 gpm.	31 days
SR 3.6.5.4	Verify each required reactor building spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.6.5.5	Verify each automatic reactor building spray valve in each required 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 3.6.5.6	Verify each required reactor building spray pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.6.5.7	Verify each required reactor building cooling train starts automatically on an actual or simulated actuation signal.	18 months
SR 3.6.5.8	Verify each required train spray nozzle is unobstructed.	10 years

3.6 REACTOR BUILDING SYSTEMS

3.6.6 Spray Additive System

LCO 3.6.6 The Spray Additive System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spray Additive System inoperable.	A.1 Restore Spray Additive System to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify each Spray Additive System manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.6.2	Verify sodium hydroxide tank solution volume is ≥ 9000 gallons.	184 days
SR 3.6.6.3	Verify sodium hydroxide tank solution concentration is > 5.0 wt% and < 16.5 wt.% NaOH.	184 days

SURVEILLANCE		FREQUENCY
SR 3.6.6.4	Verify each Spray Additive System automatic valve in the flow path actuates to the correct position on an actual or simulated actuation signal.	18 months

3.6 REACTOR BUILDING SYSTEMS

3.6.7 Hydrogen Recombiners

LCO 3.6.7 Two hydrogen recombiners shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One hydrogen recombiner inoperable.	A.1 ----------NOTE---------- LCO 3.0.4 is not applicable. ---------- Restore hydrogen recombiner to OPERABLE status.	30 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.7.1	Perform a system functional test for each hydrogen recombiner.	18 months
SR 3.6.7.2	Visually examine each hydrogen recombiner enclosure and verify there is no evidence of abnormal conditions.	18 months

SURVEILLANCE		FREQUENCY
SR 3.6.7.3	Perform a resistance to ground test for each heater phase.	18 months

B 3.6 REACTOR BUILDING SYSTEMS

B 3.6.1 Reactor Building

BASES

BACKGROUND

The reactor building consists of the reactor building (RB) structure, its steel liner, and the penetrations of this liner and structure. The reactor building is designed to contain radioactive material that may be released from the reactor core following a design basis loss of coolant accident (LOCA). Additionally, the reactor building provides shielding from the fission product radioactivity that may be present in the reactor building atmosphere following an accident.

The reactor building is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. The reactor building design includes ungrouted tendons where the cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions, and the dome roof is prestressed using a three way post tensioning system. The inside surface of the reactor building is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

The reinforced concrete structure is required for structural integrity of the reactor building under Design Basis Accident (DBA) conditions. The steel liner and its penetrations establish the leakage limiting boundary of the reactor building. Maintaining the reactor building OPERABLE limits the leakage of fission product radioactivity from the reactor building to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 1), as modified by approved exemptions.

The isolation devices for the penetrations in the reactor building boundary are a part of the reactor building leak tight barrier. To maintain this leak tight barrier:

a. All penetrations required to be closed during accident conditions are either.

   1. capable of being closed by an OPERABLE automatic reactor building isolation system except as provided in LCO 3.3.5, "Engineered Safeguards Actuation System (ESAS) Instrumentation," LCO 3.3.6, "ESAS Manual Initiation," and LCO 3.3.7, "ESAS Actuation Logic," or

   2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Reactor Building Isolation Valves";

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Reactor Building Air Locks"; and

c. The equipment hatch is closed and sealed.

APPLICABLE SAFETY ANALYSES

The design basis for the reactor building is that the reactor building must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a challenge to reactor building OPERABILITY from high pressures and temperatures are a LOCA and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within the reactor building can occur from a LOCA. In the DBA analyses, it is assumed that the reactor building is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of reactor building leakage. The reactor building was designed with an allowable leakage rate of 0.2% of reactor building air weight per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as La: the maximum allowable leakage rate at the calculated maximum peak reactor building pressure (Pa) resulting from the limiting design basis LOCA. The allowable leakage rate represented by La forms the basis for the acceptance criteria imposed on all reactor building leakage rate testing. La is assumed to be 0.2% per day in the safety analysis at Pa = 54.0 psig (Refs. 2 and 3).

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

LCO

Reactor building OPERABILITY is maintained by limiting leakage to ≤ 1.0 La, except prior to the first startup after performing a required Reactor Building Leakage Rate Testing Program leakage test. At this time, the applicable leakage limits must be met. Reactor building OPERABILITY for leakage is attained by ensuring that the equipment hatch and both doors of the personnel and emergency air locks are closed and sealed, except as appropriate for maintenance activities, and that the other isolation devices are closed, deactivated in the closed position, or OPERABLE as required. Reactor building OPERABILITY is also maintained by monitoring the deviation of key design parameters of the RB structure from the original design configuration and ensuring that structural limits are not exceeded. Visual and other required examinations of tendons, anchorages and surfaces are performed periodically in accordance with station procedures. These procedures embody applicable requirements of the 1992 Edition with the 1992 Addenda of Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code, as set forth by 10CFR50.55a(g)(6)(ii)(B) (Ref. 5). Any degradations exceeding the Containment Inspection Program acceptance criteria during inspection surveillances will be reviewed under an engineering evaluation within 60 days of the completion

of the inspection to determine what impact the degradation has on overall reactor building OPERABILITY, is if any. Compliance with this LCO, in conjunction with LCO 3.6.2, will ensure a reactor building configuration that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the reactor building air lock (LCO 3.6.2) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the reactor building being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 La.

APPLICABILITY

In MODES 1, 2, 3 and 4, the reactor building OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES. In MODES 5 and 6, the probability and consequences of events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the reactor building is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from the reactor building. The requirements for the reactor building during MODE 6 are addressed in LCO 3.9.3, "Reactor Building Penetrations."

ACTIONS

A.1

In the event the reactor building is inoperable, the reactor building must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining reactor building OPERABILITY during MODES 1, 2, 3, and 4. This time period also ensures the probability of an accident (requiring reactor building OPERABILITY) occurring during periods when reactor building is inoperable is minimal.

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the reactor building OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Reactor Building Leakage Rate Testing Program. Failure to meet air lock leakage limits specified in LCO 3.6.2 does not invalidate the acceptability of these overall leakage determinations unless its contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Reactor Building Leakage Rate Testing Program leakage test is required to be ≤ 0.6 La for combined Type B and C leakage, and ≤ 0.75 La for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 La. At ≤ 1.0 La the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Reactor Building Leakage Rate Testing Program. These periodic testing requirements verify that the reactor building leakage rate does not exceed the leakage rate assumed in the safety analysis.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. SAR, Chapter 14.
3. SAR, Section 5.2.
4. 10 CFR 50.36.
5. 10CFR50.55a(g)(6)(ii)(B)

B 3.6 REACTOR BUILDING SYSTEMS

B 3.6.2 Reactor Building Air Locks

BASES

BACKGROUND

Reactor building air locks, also known as the personnel air lock and the emergency (or escape) air lock, form part of the reactor building pressure boundary and provide a means for personnel access during all MODES of operation.

Each air lock is nominally a right circular cylinder with a door at each end. The doors are interlocked to prevent simultaneous opening. Each air lock door has been designed and is tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in the reactor building. As such, closure of a single door supports the reactor building OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in the reactor building internal pressure results in increased sealing force on each door).

The reactor building air locks form part of the reactor building pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining the reactor building leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within the reactor building are a loss of coolant accident (LOCA) and a steam line break (Ref. 2). In the analysis of each of these accidents, it is assumed that the reactor building is OPERABLE such that release of fission products to the environment is controlled by the rate of the reactor building leakage. The reactor building was designed with an allowable leakage rate of $\leq$ 0.2% of the reactor building air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as La: the maximum allowable reactor building leakage rate at the calculated maximum peak reactor building pressure, (Pa), following a design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

In MODES 1 and 2, the reactor building air locks satisfy Criterion 3 of 10 CFR 50.36 (Ref. 4). In MODES 3 and 4, the reactor building air locks satisfy Criterion 4 of 10 CFR 50.36.

LCO

Each reactor building air lock forms part of the reactor building pressure boundary. As a part of the reactor building pressure boundary, the air lock safety function is related to control of the reactor building leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test (i.e., closed and sealed), and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of the reactor building does not exist when the reactor building is required to be OPERABLE. Closure and sealing of a single door in each air lock provides sufficient leakage barrier following postulated events. Nevertheless, both doors are normally closed and sealed when the air lock is not being used for normal entry into or exit from the reactor building.

APPLICABILITY

In MODES 1, 2, 3, and 4, the reactor building air lock OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES. In MODES 5 and 6, the probability and consequences of events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the reactor building air locks are not required in MODE 5 to prevent leakage of radioactive material from the reactor building. The requirements for the reactor building air locks during MODE 6 are addressed in LCO 3.9.3, "Reactor Building Penetrations."

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable but capable of being swung, then it and the air lock barrel may be easily accessed for most repairs. It is preferred that an inoperable inner door be accessed from inside the reactor building by entering through the other OPERABLE air lock. However, if this not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the reactor building boundary is not intact (during access through the OPERABLE door). Opening the OPERABLE door, even if it means the reactor building boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the reactor building during the short time in which the OPERABLE door is expected to be open.

After each entry and exit the OPERABLE door must be immediately closed. If conditions permit, entry and exit should be via an OPERABLE air lock.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

In the event the air lock leakage results in exceeding the overall reactor building leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Reactor Building."

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

With one air lock door inoperable in one or more reactor building air locks, the OPERABLE door must be verified closed (Required Action A.1) in each affected reactor building air lock.

This ensures that a leak tight reactor building barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires the reactor building be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the remaining OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is considered reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable reactor building leakage boundary is maintained. The Completion Time of once per 31 days is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

The Required Actions have been modified by two Notes. Note 1 clarifies that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the Completion Time from the initial entry into Condition A;

only the requirement to comply with the Required Actions. Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. Reactor building entry may be required to perform Technical Specification (TS) Surveillances and Required Actions, as well as other activities on equipment inside the reactor building that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS-required activities) if the reactor building was entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

B.1, B.2, and B.3

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A.

The Required Actions have been modified by two Notes. Note 1 clarifies that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from the reactor building under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small.

C.1, C.2, and C.3

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be immediately initiated to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable since it is overly conservative to immediately declare the reactor building inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), the reactor building remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a unit shutdown. In addition, even with both doors failing the seal test, the overall reactor building leakage rate can still be within limits.

Required Action C.2 requires that one door in the affected reactor building air lock must be verified to be closed. This action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that the reactor building be restored to OPERABLE status within 1 hour.

Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status assuming that at least one door is maintained closed in each affected air lock.

D.1 and D.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining the reactor building air locks OPERABLE requires compliance with the leakage rate test requirements of the Reactor Building Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and reactor building OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall reactor building leakage rate. The Frequency is required by the Reactor Building Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable, since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria which is applicable to SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C reactor building leakage rate.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident reactor building pressure, closure of either door will support the reactor building OPERABILITY. Thus, the door

interlock feature supports the reactor building OPERABILITY while the air lock is being used for personnel transit in and out of the reactor building. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is not normally challenged when the reactor building air lock door is used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed every 18 months. 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 loss of reactor building OPERABILITY if the Surveillance were performed with the reactor at power. The 18 month Frequency for the interlock is justified based on generic operating experience. The 18 month Frequency is based on engineering judgment and is considered adequate given that the interlock is not expected to be challenged during use of the airlock.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.

2. SAR, Chapter 14.

3. SAR, Chapter 5.

4. 10 CFR 50.36.

B 3.6 REACTOR BUILDING SYSTEMS

B 3.6.3 Reactor Building Isolation Valves

BASES

BACKGROUND

The reactor building isolation valves form part of the reactor building pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on an automatic isolation signal. These isolation devices consist of either passive devices or active (automatic) devices. Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close following an accident without operator action, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers (typically reactor building isolation valves) make up the Reactor Building Isolation System.

Reactor building isolation occurs upon receipt of a high reactor building pressure signal. The reactor building isolation signal closes automatic reactor building isolation valves in fluid penetrations not required for operation of engineered safeguard systems to prevent leakage of radioactive material. Also, upon receipt of a low RCS pressure signal, certain automatic reactor building isolation valves isolate. Other penetrations are isolated by the use of valves in the closed position or blind flanges. As a result, the reactor building isolation valves (and blind flanges) help ensure that the reactor building atmosphere will be isolated in the event of a release of radioactive material to the reactor building atmosphere from the RCS following a Design Basis Accident (DBA).

OPERABILITY of the reactor building isolation valves (and blind flanges) supports the reactor building OPERABILITY during accident conditions.

The OPERABILITY requirements for the reactor building isolation valves help ensure that the reactor building is isolated. Therefore, the OPERABILITY requirements provide assurance that the reactor building function assumed in the safety analysis will be maintained.

The Reactor Building Purge System is part of the Reactor Building Ventilation System. The Reactor Building Purge System was designed for intermittent operation, providing a means of removing airborne radioactivity caused by minor leakage from the RCS prior to personnel entry into the reactor building. The Reactor Building Purge System consists of one 24 inch line for exhaust and one 24 inch line for supply, with supply and exhaust fans. The reactor building purge

supply and exhaust lines each contain two isolation valves that receive a reactor building isolation signal.

Failure of the purge valves to close following a design basis event would cause a significant increase in the radioactive release because of the large reactor building leakage path introduced by these 24 inch purge lines. Failure of the purge valves to close would result in leakage considerably in excess of the reactor building design leakage rate of ≤ 0.2% of reactor building air weight per day (La) (Ref. 1). The 24 inch purge valves are not tested for automatic closure from their open position under DBA conditions. Therefore, the 24 inch supply and exhaust purge valves are maintained closed with the handswitch keys removed (SR 3.6.3.1) in MODES 1, 2, 3, and 4 to ensure the reactor building boundary is maintained.

APPLICABLE SAFETY ANALYSES

The reactor building isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the reactor building boundary during major accidents. As part of the reactor building boundary, the reactor building isolation valve OPERABILITY supports leak tightness of the reactor building. Therefore, the safety analysis of any event requiring isolation of the reactor building is applicable to this LCO.

The DBAs that result in a release of radioactive material within the reactor building are a loss of coolant accident (LOCA) and a main steam line break (Ref. 2). In the analysis for each of these accidents, it is assumed that the reactor building isolation valves are either closed or function to close. This ensures that potential paths to the environment through the reactor building isolation valves (including reactor building purge valves) are minimized. The safety analysis assumes that the 24 inch purge valves are closed at event initiation.

The LOCA analysis assumes a fixed amount of core inventory escapes. No mechanistic scenario is evaluated to determine what portion of the inventory is released prior to closure of the reactor building isolation valves. Industry standards for sizing valve operators govern the closure times of the reactor building isolation valves.

ANO-1 does not currently allow reactor building purging in MODES 1, 2, 3, and 4. Therefore, each of the reactor building purge valves is required to remain closed with its handswitch key removed during MODES 1, 2, 3, and 4. This prevents inadvertent actuation of the reactor building purge valves while in MODES 1, 2, 3, and 4. The purge system valve design prevents a single failure from compromising the reactor building boundary as long as the system is operated in accordance with the subject LCO.

In MODES 1 and 2, the reactor building isolation valves satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3). In MODES 3 and 4, the reactor building isolation valves satisfy Criterion 4 of 10 CFR 50.36.

LCO

Reactor Building isolation valves form a part of the reactor building boundary. The reactor building isolation valve safety function is related to minimizing the loss of reactor coolant inventory and establishing the reactor building boundary during a DBA.

The automatic power operated reactor building isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The 24 inch purge valves must be maintained closed. The valves covered by this LCO are listed in the SAR (Ref. 4). Their associated stroke times are contained in the Inservice Testing Program. The normally closed manual reactor building isolation valves are considered OPERABLE when the valves are closed, blind flanges are in place, or open under administrative controls. These passive isolation valves/devices are listed in Reference 4.

The reactor building isolation valve leakage rates are addressed by LCO 3.6.1, "Reactor Building," as Type C testing.

This LCO provides assurance that the reactor building isolation valves will perform their designated safety functions to minimize the loss of reactor coolant inventory and establish the reactor building boundary during accidents.

APPLICABILITY

In MODES 1, 2, 3 and 4, the reactor building isolation valves OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES. In MODES 5 and 6, the probability and consequences of events are reduced due to the pressure and temperature limitations of these MODES. The requirements for reactor building isolation valves during MODE 5 and 6, primarily related to movement of irradiated fuel in the reactor building, are addressed in LCO 3.9.3, "Reactor Building Penetrations."

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths, except for purge valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated individual at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for reactor building isolation is indicated. Due to ALARA concerns, it is permissible for this dedicated individual to be stationed in a nearby lower dose area provided the intent of rapidly isolating the penetration is retained. Due to the size of the reactor building purge line penetration and the fact that those penetrations exhaust directly from the reactor building atmosphere to the environment, the penetration

flow paths containing these valves may not be opened under administrative controls.

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable reactor building isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable reactor building isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are further modified by a third Note, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable reactor building isolation valve.

A.1 and A.2

In the event one reactor building isolation valve in one or more penetration flow paths is inoperable, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic reactor building isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to the reactor building. Required Action A.1 must be completed within the 48 hour Completion Time. The specified time period is reasonable, considering the time required to isolate the penetration and the relative importance of supporting reactor building OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the 48 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This periodic verification is necessary to ensure that the reactor building penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside the reactor building and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside the reactor building" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside the reactor building, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

Condition A has been modified by a Note indicating this Condition is only applicable to those penetration flow paths with two reactor building isolation valves. For penetration flow paths in closed systems with only one reactor building isolation valve, Condition C provides appropriate actions.

Required Action A.2 is modified by two Notes. Note 1 applies to isolation devices located in high radiation areas and allows the devices to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small.

B.1

With two reactor building isolation valves in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of the reactor building and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two reactor building isolation valves. Condition A of this LCO addresses the condition of one reactor building isolation valve inoperable in this type of penetration flow path.

C.1 and C.2

With one or more penetration flow paths with one reactor building isolation valve inoperable, the inoperable valve must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration. Required Action C.1 must be completed within the 72 hour Completion Time. The specified time period

is reasonable, considering the relative structural integrity of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of supporting reactor building OPERABILITY during MODES 1, 2, 3, and 4. In the event the affected penetration is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure that reactor building penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one reactor building isolation valve and a closed system. The service water system is the only closed system within the reactor building to which Specification 3.6.3 Condition C applies. The service water system within the reactor building is designed to seismic category 1 standards. Because the system is located outside of the secondary shield walls, it is protected from missiles and pipe whip from reactor coolant system components. The service water system is capable of withstanding reactor building design pressure and temperature and is designed to withstand the LOCA accident transient and environment. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system.

Required Action C.2 is modified by two Notes. Note 1 applies to valves and blind flanges located in high radiation areas and allows these devices to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these devices, once verified to be in the proper position, is small.

D.1 and D.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

Each 24 inch reactor building purge isolation valve in the purge system supply and exhaust is required to be verified closed at 31 day intervals. This Surveillance is designed to ensure that a gross breach of the reactor building is not caused by an inadvertent opening of a reactor building purge valve. Detailed analysis of the purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Therefore, these valves are required to be in the closed position during MODES 1, 2, 3, and 4. A reactor building purge valve that is closed must have motive power to the valve operator removed. This can be accomplished by removing the valve handswitch key. The Frequency is consistent with other reactor building isolation valves discussed in SR 3.6.3.2.

SR 3.6.3.2

This SR requires verification that each reactor building isolation manual valve and blind flange located outside the reactor building and not locked, sealed, or otherwise secured, and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the reactor building boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those reactor building isolation valves outside the reactor building and capable of being mispositioned are in the correct position. Since verification of valve position for the reactor building isolation valves outside the reactor building is relatively easy, the 31 day Frequency was chosen to provide added assurance of the correct positions. The SR specifies that the reactor building isolation valves open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these reactor building isolation valves, once they have been verified to be in the proper position, is low.

SR 3.6.3.3

This SR requires verification that each reactor building isolation manual valve and blind flange that is located inside the reactor building and not locked, sealed, or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the reactor building boundary is within design limits. For reactor building isolation valves inside reactor building, the Frequency of "prior to entering MODE 4

from MODE 5 if not performed within the previous 92 days" is appropriate, since these reactor building isolation valves are operated under administrative controls and the probability of their misalignment is low. The SR specifies that reactor building isolation valves open under administrative controls are not required to meet the SR during the time they are open. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

The Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these reactor building isolation valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.4

Verifying that the isolation time of each automatic power operated reactor building isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period consistent with the industry standards for sizing valve operators. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

SR 3.6.3.5

Automatic reactor building isolation valves close on a reactor building isolation signal to prevent leakage of radioactive material from the reactor building following a DBA. This SR ensures that each automatic reactor building isolation valve will actuate to its isolation position on a reactor building isolation signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. SAR, Chapter 5.

2. SAR, Chapter 14.

3. 10 CFR 50.36.

4. SAR, Table 5-1.

B 3.6 REACTOR BUILDING SYSTEMS

B 3.6.4 Reactor Building Pressure

BASES

BACKGROUND

The reactor building pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). Additionally, keeping the reactor building pressure within the limits maintains the initial conditions assumed for the reactor building design basis accident (DBA) and Emergency Core Cooling System (ECCS) analyses.

The reactor building pressure is a process variable that is monitored and controlled. The reactor building pressure limits are derived from the input conditions used in the reactor building DBA and ECCS analyses. Should operation occur outside these limits coincident with a DBA, post accident reactor building pressures and ECCS performance could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Reactor building internal pressure is an initial condition used in the DBA analyses to establish the maximum peak reactor building internal pressure. The limiting DBAs considered, relative to reactor building pressure, are the LOCA and SLB. The worst-case LOCA generates larger mass and energy release than the worst-case SLB. Thus, the LOCA event bounds the SLB event from the reactor building peak pressure standpoint (Ref. 1).

The initial pressure condition used in the reactor building analysis was 14.7 psia. The LCO limit of 3.0 psig ensures that, in the event of an accident, the design pressure of 59 psig for the reactor building is not exceeded. The LCO limit of -1.0 psig ensures that operation within the design assumptions for ECCS is maintained (Ref. 2). The LCO limit of 3.0 psig does not consider instrument uncertainty. The LCO limit of -1.0 psig is considered to be an as-indicated value.

For certain aspects of transient accident analyses, maximizing the calculated reactor building pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling Systems during the core reflood phase of a LOCA analysis increases with increasing the reactor building backpressure. Therefore, for the reflood phase, the reactor building backpressure is assumed in a manner designed to conservatively minimize, rather than maximize, the reactor building pressure response in accordance with 10 CFR 50, Appendix K (Ref. 3).

In MODES 1 and 2, the reactor building pressure satisfies Criterion 2 of 10 CFR 50.36 (Ref. 4). In MODES 3 and 4, the reactor building pressure satisfies Criterion 4 of 10 CFR 50.36.

LCO

Maintaining the reactor building pressure less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak reactor building accident pressure will remain below the reactor building design pressure.

Additionally, keeping the reactor building pressure within the limits maintains the initial conditions assumed for the ECCS analyses.

APPLICABILITY

In MODES 1, 2, 3 and 4, the reactor building OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES. Since maintaining reactor building pressure within design basis limits is essential to ensure that the peak reactor building pressure from an accident does not exceed the reactor building design pressure, the LCO is applicable in MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining the reactor building pressure within the limits of the LCO is not required in MODES 5 and 6.

ACTIONS

A.1

When the reactor building pressure is not within the limits of the LCO, the reactor building pressure must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the reactor building analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Reactor Building," which requires that the reactor building be restored to OPERABLE status within 1 hour.

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that the reactor building pressure is within limits ensures that operation remains within the limits assumed in the ECCS and the reactor building analyses. The 12 hour Frequency of this SR was developed after taking into consideration operating experience related to trending of the reactor building pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to an abnormal reactor building pressure condition.

REFERENCES

1. SAR, Chapter 14.
2. SAR, Chapter 5.
3. 10 CFR 50, Appendix K.
4. 10 CFR 50.36.

B 3.6 REACTOR BUILDING SYSTEMS

B 3.6.5 Reactor Building Spray and Cooling Systems

BASES

BACKGROUND

The Reactor Building Spray and Reactor Building Cooling systems provide reactor building atmosphere cooling to limit post accident pressure and temperature in the reactor building to less than the design values. In the event of a Design Basis Accident (DBA), reduction of reactor building pressure reduces the release of fission products from the reactor building to the environment. The Reactor Building Spray and Reactor Building Cooling systems are designed to meet the requirements as discussed in the Safety Analysis Report (SAR), specifically, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal System," GDC 40, "Testing of Containment Heat Removal System," GDC 41, "Containment Atmosphere Cleanup," GDC 42, "Inspection of Containment Atmosphere Cleanup," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems" (Ref. 1).

The Reactor Building Cooling System and Reactor Building Spray System are Engineered Safeguards (ES) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained. The Reactor Building Spray System and Reactor Building Cooling System provide redundant reactor building heat removal operation. The Reactor Building Spray System and Reactor Building Cooling System provide redundant methods to limit and maintain post accident conditions to less than the reactor building design values.

Reactor Building Spray System

The Reactor Building Spray System consists of two separate trains of equal capacity, each capable of meeting the design basis. Each train includes a reactor building spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ES bus. The borated water storage tank (BWST) supplies borated water to the Reactor Building Spray System during the injection phase of operation. In the recirculation mode of operation, Reactor Building Spray System pump suction is manually transferred to the reactor building sump.

The Reactor Building Spray System provides a spray of borated water into the upper regions of the reactor building to reduce the reactor building pressure and temperature during a DBA. During MODE 1 or 2, the Reactor Building Spray System supports the Spray Additive System function of iodine removal by providing the distribution mechanism. In MODES 3 and 4, sodium hydroxide is not mixed with the spray flow. In the recirculation mode of operation, heat is removed from the reactor building sump water by the decay heat removal coolers. Each train of the

Reactor Building Spray System provides adequate spray coverage to meet the system design requirements for reactor building heat removal.

The Reactor Building Spray System is actuated automatically by a reactor building High-High pressure signal. An automatic actuation opens the Reactor Building Spray System pump discharge valves and starts the Reactor Building Spray System pumps.

Reactor Building Cooling System

The Reactor Building Cooling System during normal operations consists of five (5) chilled water supplied cooling coils each in-line with a fan. Four (4) of these fan and chiller coil circuits have in-line service water cooling coils. During normal operations the service water to these coils is isolated. The post accident configuration of the Reactor Building Cooling System consists of the four service water cooling coils and their respective axial flow fans and dampers arranged as two independent trains.

Upon receipt of an Engineered Safeguards Actuation System (ESAS) RB high pressure signal, the four (4) fans associated with the service water coils receive a start signal, the chilled water is isolated, the service water supply and discharge valves open, the RB cooler bypass dampers open (which causes the return air to bypass the chilled water coils) and the RB cooler backdraft dampers receive an open signal. This equipment is powered from class 1E electrical power.

Each of the four (4) service water coil and fan air paths receives return air separately and directly from the RB atmosphere and discharges through ducting to a common plenum for distribution to the various reactor building spaces. The four (4) fans are mounted vertically on the ventilation units and are axial-flow type. The fan motors are single speed and operate in post-accident conditions at the same speed as normal conditions. Reducing fan motor speed during accident conditions is not required due to the reduced suction pressure drop (and hence fan load relative to normal conditions) created by bypassing the chilled water coils. An RB cooling train consists of two coolers and their associated fans which have sufficient capacity to meet post accident heat removal requirements. Conservatively each reactor building emergency cooling train consists of two fans powered from the same emergency bus and their associated coils, but other combinations may be justified by an engineering evaluation (Ref. 2). The continuous availability of appropriate service water flow to the RB Cooling System is assured by the periodic addition of a biocide to the Service Water System.

APPLICABLE SAFETY ANALYSES

The Reactor Building Spray System and the Reactor Building Cooling System reduce the temperature and pressure following a DBA. The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break. The postulated DBAs are analyzed, with regard to the reactor building ES systems, assuming the loss of one ES bus. This is the worst-case single active failure,

resulting in one train of the Reactor Building Spray System and one train of the Reactor Building Cooling System being inoperable.

The analysis and evaluation show that, under the worst-case scenario, the highest peak reactor building pressure is 53.96 psig (experienced during a LOCA). The analysis shows that the peak reactor building temperature is 283.9°F (experienced during a LOCA). Both results are conservatively reported as 54 psig and 284°F, respectively, and are less than the design values. The analyses and evaluations assume a power level of 2568 MWt, one reactor building spray train and one reactor building cooling train operating, and initial (pre-accident) conditions of 140°F and 14.7 psia. The analyses also assume a delayed initiation to provide conservative peak calculated reactor building pressure and temperature responses.

The assumed Reactor Building Spray System total delay time of 300 seconds conservatively envelopes diesel generator (DG) startup (for loss of offsite power), block loading of equipment, the reactor building spray pump startup, and spray line filling (Ref. 3).

The reactor building cooling train performance for post accident conditions is given in Reference 2. The result of the analysis is that each train can provide 100% of the required cooling capacity during the post accident condition. The train post accident cooling capacity under varying reactor building ambient conditions, is also shown in Reference 2.

The assumed Reactor Building Cooling System total delay time of 300 seconds conservatively envelopes signal delay, DG startup, block loading of equipment, fan startup, and service water pump startup times (Ref. 3).

In MODES 1 and 2, the Reactor Building Spray System and the Reactor Building Cooling System satisfy Criterion 3 of 10 CFR 50.36 (Ref. 4). In MODES 3 and 4, the Reactor Building Spray System and the Reactor Building Cooling System satisfy Criterion 4 of 10 CFR 50.36.

LCO

During a DBA, the combination of one reactor building cooling train and one reactor building spray train is sufficient to reduce the reactor building pressure and temperature. One reactor building spray train is required, during MODE 1 or 2, to support the Spray Additive System in the removal of iodine from the reactor building atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, in MODES 1 and 2, two reactor building spray trains and two reactor building cooling units must be OPERABLE. Therefore, in the event of an accident, the minimum requirements are met, assuming the worst-case single active failure occurs. In MODE 3 or 4, one reactor building spray train and one reactor building cooling train are required to be operable. The LCO is provided with a Note which clarifies this requirement.

The Reactor Building Spray System includes spray pumps, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an Engineered Safeguards Actuation System signal and manually transferring suction to the reactor building sump.

The Reactor Building Cooling System includes cooling coils, dampers, axial flow fans, single speed fan motors, instruments, and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1, 2, 3, and 4, the reactor building OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES. Since an event could cause a release of radioactive material in the reactor building as well as a temperature and pressure rise, the Reactor Building Spray System and the Reactor Building Cooling System are required to be OPERABLE in MODES 1, 2, 3, and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the Reactor Building Spray System and the Reactor Building Cooling System are not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

With one reactor building spray train inoperable in MODE 1 or 2, the inoperable reactor building spray train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to support the iodine removal and perform the reactor building cooling functions. The 72 hour Completion Time takes into account the redundant heat and iodine removal capability afforded by the OPERABLE reactor building train, reasonable time for repairs, and the low probability of a DBA occurring during this period.

The 10 day portion of the Completion Time for Required Action A.1 is based on the low probability of coincident entry into two Conditions in this LCO coupled with the low probability of an accident occurring during this time. Refer to Section 1.3, Completion Times for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time

B.1

With one of the reactor building cooling trains inoperable in MODE 1 or 2, the inoperable reactor building cooling train must be restored to OPERABLE status within 7 days. The remaining OPERABLE components are capable of providing at least 100% of the heat removal needs after an accident. The 7 day Completion Time takes into account the redundant heat removal capabilities afforded by combinations of the Reactor Building Spray System and Reactor Building Cooling System and the low probability of a DBA occurring during this period.

The 10 day portion of the Completion Time for Required Action B.1 is based on the low probability of coincident entry into two Conditions in this LCO coupled with the low probability of an accident occurring during this time. Refer to Section 1.3 for a more detailed discussion of the "from discovery of failure to meet the LCO" portion of the Completion Time.

C.1

With two of the reactor building cooling trains inoperable in MODE 1 or 2, one of the reactor building cooling trains must be restored to OPERABLE status within 72 hours. The remaining spray system components (both spray trains are OPERABLE or else Condition G is entered) support iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour Completion Time takes into account the redundant heat removal capabilities afforded by the Reactor Building Spray System and the low probability of a DBA occurring during this period.

D.1

If the Required Actions and associated Completion Times are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours. 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.

E.1

With either one required reactor building (RB) spray train or one required reactor building cooling train inoperable in MODE 3 or 4, the inoperable train must be restored to OPERABLE status in 36 hours. The 36 hour Completion Time is reasonable based on consideration of the cooling capacity of the remaining required train of RB cooling or RB spray, the reduced reactor coolant energy in these MODES, and the short time spent in these MODES.

F.1

If the Required Action and associated Completion Time of Condition E are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to MODE 5 within 36 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required unit conditions in an orderly manner and without challenging unit systems.

G.1

With two reactor building spray trains inoperable in MODE 1 or 2, or any combination of three or more reactor building spray and reactor building cooling trains inoperable in MODE 1 or 2, or one required reactor building spray train and one required reactor building cooling train inoperable in MODE 3 or 4, then LCO 3.0.3 must be entered immediately. The first part of this Condition addresses the loss of Spray Additive System support which would result from two inoperable reactor building spray trains in MODE 1 or 2. The second part of this Condition considers the loss of adequate reactor building cooling capacity in MODE 1 or 2 which would result from the loss of three or more of the four RB spray and RB cooling trains. Finally, the third part of this Condition addresses loss of reactor building cooling capability in MODES 3 and 4 when only one train of RB spray and one train of RB cooling are required.

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the reactor building spray flow path provides assurance that the proper flow paths will exist for the Reactor Building Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were 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 SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown or control room indication, that those valves outside the reactor building and capable of potentially being mispositioned are in the correct position.

SR 3.6.5.2

Operating each required reactor building cooling train fan unit for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. This SR is performed by starting (unless operating) each operational cooling fan from the control room. The 31 day Frequency was developed considering the known reliability of the fan units and controls, the redundancy available, and the low probability of a significant degradation of the

reactor building cooling trains occurring between surveillances and has been shown to be acceptable through operating experience.

SR 3.6.5.3

Verifying that a service water flow rate of 1200 gpm is provided to each required reactor building cooling train provides assurance that the original design flow rate is being achieved and that the service water flow rate is not degrading (Ref. 3). Assurance that the flow doesn't degrade by biological fouling between surveillances is provided by the addition of a biocide to the Service Water System whenever the service water temperature is between 60°F and 80°F. The Frequency was developed considering the known reliability of the system, the redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

SR 3.6.5.4

Verifying that each required reactor building spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Acceptable performance will be indicated if the pump starts, operates for fifteen minutes, and the discharge pressure and flow rate are within ± 10 % of a point on the pump head curve. Flow and differential pressure are measured during normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 5). Since the Reactor Building Spray System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms the discharge pressure and flow rate are within ± 10 % of a point on the pump head curve and is indicative of overall pump performance. Such inservice tests confirm component OPERABILITY, trend performance, and may detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.5.5 and SR 3.6.5.6

These SRs require verification that each automatic reactor building spray valve actuates to its correct position and that each reactor building spray pump starts upon receipt of an actual or simulated actuation signal. The SRs are considered satisfactory if visual observation and control board indication verifies that all components have responded to the actuation signal properly. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. During testing of the spray pump, the reactor building isolation valve in the spray line is closed with its breaker open to prevent spraying the reactor building. After spray pump performance is verified, the pump is stopped. Its breaker is racked down to prevent restart. Power is then restored to the reactor building isolation valve for valve testing. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a unit outage and on the potential for an unplanned transient if the

Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.5.7

This SR requires verification by control board indication that each required reactor building cooling train actuates upon receipt of an actual or simulated actuation signal. The 18 month Frequency has been shown to be acceptable through operating experience. See SR 3.6.5.5 and SR 3.6.5.6, above, for further discussion of the basis for the 18 month Frequency.

SR 3.6.5.8

With the reactor building spray header isolated and drained of any solution, low pressure air or smoke can be blown through test connections. Performance of this Surveillance demonstrates that each spray nozzle is unobstructed and provides assurance that spray coverage of the reactor building during an accident is not degraded. Due to the passive nature of the design of the nozzles, a test at 10 year intervals is considered adequate to detect obstruction of the spray nozzles.

REFERENCES

1. SAR, Section 1.4.
2. SAR, Chapter 6.
3. SAR, Chapter 14.
4. 10 CFR 50.36.
5. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.6 REACTOR BUILDING SYSTEMS

B 3.6.6 Spray Additive System

BASES

BACKGROUND

The Spray Additive System reduces the iodine fission product inventory in the reactor building atmosphere resulting from a Design Basis Accident (DBA). The Reactor Building Spray System supports the Spray Additive System iodine removal function by providing a distribution mechanism for the solution.

The Reactor Building Spray System and Spray Additive System perform no function during normal operations. In the event of a loss of coolant accident (LOCA), the Spray Additive System will be automatically actuated upon a reactor building high-high pressure signal by the Engineered Safeguards Actuation System. Actuation of the Spray Additive System opens the sodium hydroxide isolation valves, which are powered from independent buses. When the valves are open, the sodium hydroxide solution is ready to be introduced into the RB Spray System headers.

Radioiodine in its various forms is the fission product of primary concern in the evaluation of the dose consequences of an accident. It is absorbed by a sprayed solution from the reactor building atmosphere. The spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms. Sodium hydroxide (NaOH), because of its stability when exposed to radiation and elevated temperature, is the spray additive utilized.

The NaOH tank is designed and located to permit gravity draining into the Reactor Building Spray System. The sodium hydroxide volume requirement is given in gallons for compatability with the design analyses. The minimum NaOH tank volume of 9000 gallons preserves the required NaOH solution contribution from the tank to the post-LOCA minimum sump level. Both the Reactor Building Spray System pumps initially take suction from the borated water storage tank (BWST) via two independent flow paths. The NaOH tank has a common outlet that splits and feeds each of the Reactor Building Spray System suction lines. The system is designed to discharge at a rate commensurate with the draining rate of the BWST so that all borated water injected is mixed with sodium hydroxide.

The flow rate is proportioned to provide a spray solution with a pH which is alkaline (Ref. 1). The range of alkalinity was established not only to aid in removal of airborne iodine, but also to minimize the corrosion of mechanical system components that would occur if the acidic borated water were not buffered. The pH range also considers the environmental qualification of equipment in the reactor building that may be subjected to the spray.

APPLICABLE SAFETY ANALYSES

The reactor building Spray Additive System provides for the effective removal of airborne iodine within the reactor building following a DBA.

Following the assumed release of radioactive materials into the reactor building, the reactor building is assumed to leak at its design value following the accident. The analysis assumes that most of the reactor building volume is covered by the spray.

The delay time assumed for the Spray Additive System is the same as for the Reactor Building Spray System and is discussed in the Bases for LCO 3.6.5, "Reactor Building Spray and Cooling Systems."

The LOCA analyses assume that one train of the Reactor Building Spray System/Spray Additive System is inoperable and that sufficient NaOH volume is added to the remaining BWST by the Reactor Building Spray System flow path.

In the evaluation of the worst-case LOCA, the safety analysis assumed that an alkaline reactor building spray effectively reduced the airborne iodine.

Each Reactor Building Spray System suction line is equipped with its own gravity feed from the NaOH tank. Therefore, in the event of a single failure within the Spray Additive System (i.e., NaOH isolation valve failure), NaOH will still be mixed with the borated water, establishing the alkalinity to provide effective iodine removal.

The Spray Additive System satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

The Spray Additive System is necessary to reduce the release of radioactive material to the environment in the event of a DBA. To be considered OPERABLE, the volume and concentration of the NaOH solution must be sufficient to provide NaOH into the spray flow until the Reactor Building Spray System suction path is switched from the BWST to the reactor building sump and to raise the long term sump solution pH to a level conducive to iodine removal. The long term sump solution pH is in the alkaline range. This pH maximizes the effectiveness of the iodine retention mechanism without introducing conditions that may induce caustic stress corrosion cracking of mechanical system components. In addition, it is essential that valves in the Spray Additive System flow paths are properly positioned and that automatic valves are capable of activating to their correct positions.

APPLICABILITY

In MODES 1 and 2, the reactor building OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in a lower

MODE would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in a lower MODE. Although the core is designed to retain structural integrity during an accident, fuel failure with resultant radioactive material release is postulated and the Spray Additive System is required OPERABLE in MODES 1 and 2.

In MODES 3 and 4, there is no postulated fuel failure contribution to radioactive material release and significantly less need for iodine removal capacity. Also, because of the limited time spent in these MODES, the probability of an event requiring use of the Spray Additive System is low. Therefore, the Spray Additive System is not required to be OPERABLE in MODE 3 or 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Thus, the Spray Additive System is not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

With the Reactor Building Spray Additive System inoperable, the system must be restored to OPERABLE status within 72 hours. The pH adjustment capability of the spray solution for corrosion protection and iodine removal enhancement is reduced or non-existent in this Condition. The Reactor Building Spray System would still be available and would remove some iodine from the reactor building atmosphere in the event of a LOCA. The 72 hour Completion Time takes into account the redundant flow path capabilities and the low probability of the worst-case DBA occurring during this period.

B.1

If the Required Actions and associated Completion Times are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to 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 3.6.6.1

Verifying the correct alignment of NaOH manual, power operated, and automatic valves in the Spray Additive System flow path provides assurance that the system is able to provide NaOH to the Reactor Building Spray System in the event of a DBA. This SR does not apply to valves that are locked, sealed, or otherwise

secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown or control room indication, that those valves outside the reactor building capable of potentially being mispositioned are in the correct position.

SR 3.6.6.2

To provide the most effective iodine removal, the reactor building spray should be an alkaline solution. Since the BWST contents are normally acidic, the NaOH tank must provide a sufficient volume of NaOH to adjust pH for all water injected. This SR is performed to verify the availability of sufficient NaOH solution in the Spray Additive System. The NaOH tank solution minimum volume of 9000 gallons corresponds to a tank level of approximately 26 feet at a temperature of 77°F and a NaOH concentration of 5.0 wt%. This parameter does not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures. The minimum NaOH tank volume preserves the required NaOH solution contribution from the tank to the post-LOCA minimum sump level. The 184 day Frequency is based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Tank level is also indicated and alarmed in the control room, such that there is a high confidence that a substantial change in level would be detected.

SR 3.6.6.3

This SR provides verification of the NaOH concentration in the NaOH tank and is sufficient to ensure that the spray solution being injected into the reactor building is at the correct pH level. The concentration of NaOH in the NaOH tank must be determined by chemical analysis. There is no instrument uncertainty included in the surveillance limit values. Additional allowances for instrument uncertainty are contained in the implementing procedures. The 184 day Frequency is sufficient to ensure that the concentration of NaOH in the tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

SR 3.6.6.4

This SR provides verification that each automatic valve in the Spray Additive System flow path actuates to its correct position. 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 was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. SAR, Chapter 6.

2. 10 CFR 50.36.

B 3.6 REACTOR BUILDING SYSTEMS

B 3.6.7 Hydrogen Recombiners

BASES

BACKGROUND

Permanently installed hydrogen recombiners are required to reduce the hydrogen concentration in the reactor building following a loss of coolant accident (LOCA). The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor remains in the reactor building, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammability limits would not be reached until several days after a LOCA.

Two 100% capacity independent hydrogen recombiners are provided. Each consists of controls located in the control room, a power supply located in the auxiliary building, and a recombiner located in the reactor building. The recombiners have no moving parts. Recombination is accomplished by heating a hydrogen air mixture above 1150°F. The resulting water vapor and discharge gases are cooled prior to discharge from the recombiner. Air flows through the unit at approximately 100 scfm. A single recombiner is capable of maintaining the hydrogen concentration in the reactor building below the 4 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safeguards (ES) bus and is provided with a separate power panel and control panel.

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners provide for the capability of controlling the bulk hydrogen concentration in the reactor building to less than a concentration of 4 v/o following a DBA. This would prevent a hydrogen burn inside the reactor building, thus ensuring the reactor building pressure and temperature assumed in the accident analysis are not exceeded. The limiting DBA relative to hydrogen generation is a LOCA.

Hydrogen may accumulate within the reactor building following a LOCA as a result of:

a. A metal steam reaction between the zirconium fuel rod cladding and the reactor coolant;

b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the reactor building sump;

c. Hydrogen in the RCS at the time of the LOCA (i.e., hydrogen dissolved in the reactor coolant and hydrogen gas in the pressurizer vapor space); or

d. Corrosion of metals exposed to Reactor Building Spray System and Emergency Core Cooling Systems solutions.

To evaluate the potential for hydrogen accumulation in the reactor building following a LOCA, the hydrogen generation as a function of time following the initiation of the accident has been evaluated. Conservative assumptions presented by References 1 and 2 are used to maximize the amount of hydrogen calculated. These evaluations demonstrate approximately 8.9 days are needed for hydrogen concentration to increase to 4 v/o post LOCA without recombiner operation.

The hydrogen recombiners satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

Two hydrogen recombiners must be OPERABLE. This ensures operation of at least one hydrogen recombiner in the event of a worst-case single active failure.

Operation with at least one hydrogen recombiner ensures that the post LOCA hydrogen concentration can be prevented from exceeding the flammability limit.

APPLICABILITY

In MODES 1 and 2, the hydrogen recombiner OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in a lower MODE would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in a lower MODE. Two hydrogen recombiners are required OPERABLE in MODES 1 and 2 to assure control of hydrogen concentration within the reactor building to less than the flammability limit of 4 v/o.

In MODES 3 and 4, both the hydrogen production rate and the total hydrogen produced would be less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an event requiring the hydrogen recombiners is low. Therefore, the hydrogen recombiners are not required in MODE 3 or 4.

In MODES 5 and 6, the probability and consequences of a LOCA are low, due to the pressure and temperature limitations. Therefore, hydrogen recombiners are not required in these MODES.

ACTIONS

A.1

With one hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE recombiner is adequate to perform the hydrogen control function. However, the overall reliability is reduced because a single failure in the OPERABLE recombiner could result in a reduced hydrogen control capability. The 30 day Completion Time is based on the availability of the other hydrogen recombiner, the small probability of a LOCA occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

Required Action A.1 has been modified by a Note stating that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one hydrogen recombiner is inoperable. This allowance is based on the availability of the other hydrogen recombiner, the small probability of a LOCA occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit.

B.1

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

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1

Performance of a system functional test for each hydrogen recombiner ensures that the recombiners are operational and can obtain and sustain the temperature necessary for hydrogen recombination. In particular, this SR requires verification that the minimum heater sheath temperature increases to ≥ 700°F in ≤ 90 minutes. After reaching 700°F, the power is increased to maximum for approximately 2 minutes and power verified to be ≥ 60 kW. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.7.2

This SR ensures that there are no physical problems that could affect recombiner operation. Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failures involve loss of power, blockage of the internal flow path, missile impact, etc. A visual inspection is sufficient to determine abnormal conditions that could cause such failures. The 18 month Frequency for this SR was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

SR 3.6.7.3

This SR requires performance of a resistance to ground test for each heater phase, following the performance of SR 3.6.7.2, to ensure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance to ground for any heater phase is ≥ 10,000 ohms. The 18 month Frequency for this SR was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

REFERENCES

1. SAR, Section 6.6.

2. Regulatory Guide 1.7, Revision 2.

3. 10 CFR 50.36.

ADMINISTRATIVE

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

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

A3 CTS 3.6.3 prohibits changing unit status where the reactor is made subcritical by less than 1% Δk/k without reactor building (RB) integrity. This unit status is identified as ITS MODE 2 and is a Condition for which the RB is required OPERABLE.
ITS LCO 3.0.4 adequately controls compliance with conditions required to be met for MODE changes. Therefore, CTS 3.6.3 is redundant and may be administratively deleted.

A4 Not used.

A5 CTS 3.6.6 is not specifically identified as applicable to reactor building (RB) penetrations with two (2) valves; however, testing and closure of the 'other' valve is discussed which implies that this was the intent. The NUREG 3.6.3 Condition A Note about applicability to systems with two RB isolation valves is explicit and is adopted as an administrative change in the ITS consistent with NUREG-1430.

3.6.3-8 With 2 valves in one penetration inoperable, CTS 3.6.6 cannot be met (other valve not operable). Therefore, CTS 3.6.1 is entered. This is equivalent to NUREG 3.6.3 Conditions B and D. Therefore, these requirements are adopted in the ITS as an administrative change consistent with NUREG-1430.

A6 CTS 3.3.7(C) and (D) define Conditions where the requirements of CTS 3.3.4 (A) cannot be met because one or two trains of reactor building (RB) cooling are not OPERABLE while both trains of RB spray are OPERABLE. The NUREG 3.6.6 Conditions describe what is not OPERABLE but do not include what is OPERABLE since the LCO defines this. The requirements of ITS 3.6.5 Condition B are administratively equal to CTS 3.3.7(C). The requirements of ITS 3.6.5 Condition C are administratively equal to CTS 3.3.7(D). The CTS 3.3.7(C) and (D) statement that "but both reactor building spray systems are operable" is administratively deleted from the CTS to make the CTS Condition statement consistent with NUREG-1430.

A7 CTS 3.3.6 provides actions if the requirements of CTS 3.3.1 for one reactor building (RB) cooling train and one RB spray train are not met during MODES 3 and 4.

CTS 3.3.7 (C), (D), and (E) are applicable for CTS 3.3.4 which is applicable during MODES 1 and 2. The references to CTS 3.3.7 and to reactor shutdown are administratively deleted from CTS 3.3.6 in accordance with the Applicability of ITS 3.6.5 Condition E.

3.6.2-1 A8 The NUREG 3.6.2 ACTIONS Note 3 is adopted by the ITS as an appropriate statement of modification to Conditions which is interpreted to be presently available in the CTS. Since this Note is implied to be already available it is adopted as an administrative change.

A9 The NUREG 3.6.2 Condition A Note 1 is adopted administratively since the requirements of NUREG 3.6.2 Condition C are adopted for when two airlock doors are inoperable. This change is administrative because the present practice in implementing CTS 1.7a and b and CTS 3.6.1 for the personnel hatch and emergency hatch is equivalent.

3.6.5-1 A10 Not Used.

A11 Not used.

A12 CTS 4.5.2.1.2 (a)(1) Note 1 is administratively deleted since the "effective until" date of July 14, 1995, has been passed.

A13 CTS 4.12.1 b.2. indicates the visual examination of the hydrogen recombiners is looking for evidence of abnormal conditions "within" the recombiner enclosure. NUREG SR 3.6.8.2 (adopted as ITS SR 3.6.7.2) is worded slightly different in that it does not include the word "within" but still describes a visual examination which is considered administratively equivalent to the CTS examination. Since the two examinations are considered equivalent the ITS will adopt the NUREG SR 3.6.8.2 wording and remain consistent with NUREG-1430.

A14 CTS 3.3.7(E) describes a condition where one of two trains of reactor building (RB) spray and one of two trains of RB cooling are inoperable while meeting CTS 3.3.4(A), that is, during MODE 1 or 2. Since in the ITS, multiple Conditions of an LCO may be entered, the Condition of CTS 3.3.7(E) is equivalent to entering NUREG 3.6.6 Condition A (ITS 3.6.5 Condition A) concurrently with NUREG 3.6.6 Condition C (ITS 3.6.5 Conditon B). The equivalencies extend to the Required Actions and the Completion Times required in the CTS and the NUREG. The requirements of NUREG 3.6.6 Condition A and NUREG 3.6.6 Condition C are therefore adopted administratively as modified for ITS LCO 3.6.5 to retain consistency with NUREG-1430.

A15 Not used.

A16 CTS Table 4.1-2 Item #8 requires that the Reactor Building Isolation Trip be tested for "Functioning" every 18 months. The CTS functioning test is considered

administratively equal to the NUREG SR 3.6.3.7 requirements to verify that each automatic reactor building isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal. NUREG 3.6.3.7 exempts those automatic valves that are locked, sealed, or otherwise secured in position from the required test. Although this exception is not explicitly presented in the CTS, this change is considered administrative due to the interactions of CTS 4.0.3, CTS 1.7.d, GDC 55, and GDC 56. CTS 4.0.3 states: "Surveillance Requirements do not have to be performed on inoperable equipment." CTS 1.7.d requires all automatic reactor building isolation valves to be Operable or deactivated in the closed position. GDCs 55 and 56 would require an inoperable automatic valve to be locked closed. Therefore, a locked closed automatic valve would not require functional testing. In addition, if an automatic isolation valve were to be locked in the closed position, its associated system would be inoperable, and the operation would be limited by the associated Required Action. NUREG SR 3.6.3.7 will be renumbered and adopted as ITS SR 3.6.3.5.

3.6.3-1

A17 CTS 4.4.1.4 does not specifically identify that isolation valves undergoing functional test are timed while being stroked. However, the CTS functional test is done in accordance with ASME Section XI which includes timing when testing valves. Therefore, NUREG SR 3.6.3.5, which is renumbered and adopted as ITS SR 3.6.3.4, is considered as administratively equivalent.

A18 CTS 1.7 conditions a., b., and c. provide configuration details concerning the OPERABILITY of the equipment hatch, the personnel and emergency hatches and non-automatic reactor building isolation devices. Since these details describe the OPERABLE configuration, they are administratively equivalent to stating that the equipment is OPERABLE. ITS LCOs 3.6.1, 3.6.2, and 3.6.3 will adopt the NUREG convention that the equipment be OPERABLE.

3.6.3-2

A19 The CTS 1.7 condition c. reference to "non-automatic reactor building isolation valves" is considered to envelope manual valves used as reactor building isolation devices. NUREG-1430 LCO 3.6.3 refers to these valve types individually. The CTS 1.7 condition c. definition is revised so that it is consistent with the NUREG and the administrative equivalence is evident.

A20 Fulfilling the requirements of the Reactor Building Leak Rate Testing Program (RBLRTP) is the equivalent of fulfilling the requirements of CTS 4.4.1 as required by CTS 1.7 condition e. The CTS 1.7 definition is revised to remain consistent with the NUREG for the ITS.

ANO 71

A21 Not used.

3.6.3-11

A22 ANO-1 interprets the CTS 1.7.c requirement that: "All non-automatic reactor building isolation valves and blind flanges are closed as required" to allow manual isolation valves to be opened under administrative controls due to the presence of " as required." The definition of containment integrity for plants of more recent vintage, like ANO-2,

specifically allowed an exception to having manual valves locked closed. More specific wording for this exception was inserted into these definitions in response to Generic Letter 91-08, "Removal of Component Lists from Technical Specifications." This was not considered necessary for ANO-1 since the ANO-1 TS did not contain a list of penetrations that would be relocated, and due to the phrase "closed as required."

The alternative would result in a conflict within the CTS. For example, CTS Table 4.1-3 Item 3 requires the core flood tanks to be sampled for boron concentration monthly and after each makeup. The sample penetration is equipped with a remote manual valve inside the reactor building and a manual isolation valve outside of the reactor building. Both valves are normally closed, as required by CTS 1.7.c. However, in order to sample a core flood tank, the manual valve and the remote manual valve must be opened. This is performed under administrative controls and is described in SAR Section 5.2.2.4.1. Without the allowance to open manual valves, as provided by our interpretation of CTS 1.7.c, ANO-1 would not be able to sample the core flood tanks as required by the CTS. The ITS provides clarifications to prevent this apparent conflict. Therefore, this change is considered to be administrative in nature.

A23 CTS 4.5.2.1 and 4.5.2.2 specify the surveillances to be performed on components
3.6.5-4 associated with the reactor building spray and reactor building cooling systems. One interpretation of the requirements would require that all components be tested regardless of whether the component is required to be OPERABLE in accordance with CTS 3.3.1 and CTS 3.3.4. However, CTS 4.0.1 states: "Surveillance Requirements shall be met during the operational modes or other conditions specified for individual LCOs unless otherwise stated in an individual Surveillance Requirement." The CTS 4.0.1 Bases go on to state: "Surveillance Requirements do not have to be performed when the facility is in an operational mode for which the requirements of the associated LCO do not apply unless otherwise specified." Therefore the incorporation of the CTS requirements as ITS SRs 3.6.5.1 through SR 3.6.5.7 is considered to be administrative, from this aspect.

TECHNICAL CHANGE -- MORE RESTRICTIVE

M1 CTS 3.3.1 establishes the Applicability for a number of components, including the Reactor Building (RB) Spray System, the RB Cooling System, and the Engineered Safeguards System valves for these systems by referencing CTS 3.6.1. CTS 3.23.1 establishes Applicability for the RB purge valves similarly by referencing CTS 3.6.1. CTS 4.26.1 establishes Surveillance Frequencies for RB purge isolation valves relative to the requirement for RB Integrity (OPERABILITY) per CTS 3.6.1.

CTS 3.6.1 requires RB Integrity whenever all three following conditions exist:

a. Reactor coolant pressure is ≥ 300 psig,
b. Reactor coolant temperature is ≥ 200°F, and
c. Nuclear fuel is in the core.

With these criteria, RB Integrity would be required sometime during ITS MODE 4 but not necessarily when this MODE was entered from MODE 5.
The Applicabilities of NUREG 3.6.1 for RB OPERABILITY, NUREG 3.6.3 for the RB isolation valves and NUREG 3.6.6 for the RB Spray and Cooling Systems include MODES 1, 2, 3, and 4. This is an additional restriction on unit operation consistent with NUREG-1430.

M2 The CTS 3.6.1 progression of actions for failure to maintain reactor building (RB) integrity are to restore in 1 hour, be in hot standby (ITS MODE 2) in another 6 hours and in cold shutdown (ITS MODE 5) in a further 30 hours. These actions equate to those presented in NUREG 3.6.1 Condition B, 3.6.2 Condition D, and 3.6.3 Condition D with the following exceptions. NUREG 3.6.1 Required Action (RA) B.1, 3.6.2 RA D.1, and 3.6.3 RA D.1 require the unit to be in MODE 3 in 6 hours after entry into NUREG 3.6.1 Condition B. The NUREG requirement to be in MODE 3 (subcritical) rather than MODE 2 (critical) will be adopted in the ITS and is desirable in this instance because there is less potential energy in a non-critical reactor which could challenge RB OPERABILITY should an event occur. This is an additional restriction on unit operation consistent with NUREG-1430.

3.6.2-3

3.6.3-5

The CTS 3.6.4 progression of actions for failure to maintain RB pressure are identical to those for CTS 3.6.1 above. NUREG 3.6.4 RA B.1 requires the unit to be in MODE 3 in 6 hours after entry into NUREG 3.6.4 Condition B. The NUREG requirement to be in MODE 3 (subcritical) rather than MODE 2 (critical) will be adopted in the ITS for the reasons stated above. This is an additional restriction on unit operation consistent with NUREG-1430.

M3 CTS 3.6.4 and 3.6.6 set requirements while the reactor is critical. This unit status corresponds to ITS MODE 1 and 2. However, the CTS has an implied applicability of MODES 1, 2, 3 and 4 since these specs include a requirement to go to cold shutdown (ITS MODE 5) as part of the actions should the requirements not be met. During MODES 3 and 4, the reactor coolant is reduced to a temperature and pressure significantly below operating conditions at power. However, during these MODES, there remains sufficient stored energy within the coolant to allow any coolant released by a LOCA to flash to steam and thereby cause a release of fission products to the reactor building atmosphere. Although no core damage is anticipated due to a LOCA initiated during shutdown, the fission products present in the coolant at the time of the rupture would be available for release to the reactor building atmosphere. Therefore, maintaining reactor building OPERABILITY during MODES 3 and 4 ensures that the offsite radiation exposure of 10 CFR 100 is not exceeded. The Applicability of NUREG 3.6.4 and NUREG 3.6.6 is MODES 1, 2, 3, and 4. This Applicability will be adopted by the ITS to address the explicit and implicit requirements of the CTS. This is an additional restriction on unit operation consistent with NUREG-1430.

M4 CTS 3.6.5 requires a check of all (inside and outside) manual reactor building (RB) isolation valves "Prior to criticality following a refueling shutdown." The ITS will adopt NUREG SR 3.6.3.3 (renumbered to ITS SR 3.6.3.2) for position checks of valves outside the RB and NUREG SR 3.6.3.4 (renumbered to ITS SR 3.6.3.3) for position checks of valves inside the RB. NUREG SR 3.6.3.3 and NUREG SR 3.6.3.4 explicitly include blind flanges in these position checks. Although CTS 1.7.c includes
3.6.3-6 blind flanges, CTS 3.6.5 does not explicitly require position checks for these components. The adoption of these requirements in the ITS results in a more restrictive requirement in that blind flanges were not explicitly included in the CTS surveillance. The NUREG SR 3.6.3.3 requirements are to verify the position of appropriate valves outside the reactor building on a Frequency of 31 days. The NUREG SR 3.6.3.4 requirements place the inside valve position check Frequency as once when entering MODE 4 from MODE 5 unless done in the previous 92 days instead of when entering MODE 2 from MODE 3 as the CTS requires. The NUREG requirements are more consistent with the ITS threshold for RB OPERABILITY. These are additional restrictions on unit operation consistent with NUREG-1430.

M5 The NUREG 3.6.3 ACTIONS Notes 2 and 3 will be adopted in the ITS as appropriate clarifications of the ACTIONS for each reactor building (RB) isolation valve and its associated system. These details are not specifically addressed in the CTS.

The CTS has implied requirements, associated with requirements for RB integrity, which address two (2) inoperable valves in a penetration flow path, differentiate closed-system penetrations, or verify continued system isolation. The ITS will adopt NUREG 3.6.3 RA A.2 with Notes and both Completion Times, NUREG 3.6.3 Condition B with Note and NUREG 3.6.3 ACTION C with both Notes as appropriately specific, and therefore more restrictive, means of addressing requirements for RB isolation valves. These requirements are additional restrictions on unit operation consistent with NUREG-1430.

M6 Not used.

M7 CTS 3.3.4(B) provides the limits for volume and concentration for the sodium hydroxide (NaOH) tank. CTS Table 4.1-3 item #6 provides the surveillance Frequency for the NaOH solution concentration, however, there is no comparable NaOH tank solution volume surveillance Frequency requirement in the CTS. The ITS will adopt the NUREG SR 3.6.7.2 surveillance Frequency of 184 days for the NaOH tank solution volume to remain consistent with the surveillance Frequency of the NaOH tank solution concentration as well as the requirements of NUREG-1430.

M8 For CTS 3.3.6 and 3.3.7, in the event of concurrent reactor building (RB) spray or RB cooling train inoperability, the existing requirements allow independent application of allowed repair times without restriction. When a subsequent inoperability occurs just prior to restoration of the previous inoperability and close to the expiration of the CTS-allowed 36 hours for RB spray or 7 days for RB cooling, when taken to extreme, this independent application can provide an unlimited time of operation with an inoperable RB Spray or RB cooling train. While these simultaneous inoperabilities are expected to be rare, adoption of the maximum restoration time limit provided by NUREG 3.6.6 A.1 and C.1 is proposed to prevent extended operation in the respective Conditions. The proposed Technical Specifications format presents this as an additional Completion Time of "10 days from discovery of failure to meet the LCO" for both ITS 3.6.5 Required Action (RA) A.1 and RA B.1 and is considered to be reasonable. These additional Completion Time requirements represent additional restrictions on unit operation consistent with NUREG-1430.

M9 Not used.

M10 NUREG SR 3.6.7.1 requires surveillance of the position of the manual, power operated and automatic valves in the Spray Additive System on a frequency of 31 days. NUREG SR 3.6.7.4 requires surveillance of the actuation of the sodium hydroxide flow path automatic valves every 18 months. Neither of these surveillances, including the requirement to verify manual valve position, is in the CTS, however, they are proposed to be adopted in the ITS as adequate methods, compatible with the system design, for assuring the availability of the Spray Additive System for its safety function.

3.6.5-2 NUREG SR 3.6.6.1 requires surveillance of the position of the manual, power operated and automatic valves in the reactor building spray system on a frequency of 31 days. This surveillance, including the requirement to verify manual valve position, is not contained in the CTS, however, it is proposed to be adopted in the ITS as adequate methods, compatible with the system design, for assuring the availability of the reactor building spray system for its safety function.

These NUREG requirements, adopted by the ITS, are additional restrictions on the operation of the unit consistent with NUREG-1430.

M11 In CTS 1.7 a & b, the air lock doors are required to be "closed and sealed" to establish reactor building integrity. In the CTS context, "sealed" means meeting leakage program requirements. The NUREG 3.6.2 Required Action (RA) A.2 requirements, however, include locking the door and the NUREG 3.6.2 RA A.3 requirements include verifying the door is locked. The NUREG requirements to lock the doors when performing these Required Actions will be adopted by the ITS.

The Note on NUREG 3.6.2 RA A.3 which provides that administrative means may be used to verify that air lock doors in high radiation areas are locked closed is adopted by the ITS.

The NUREG SR 3.6.2.1 Notes 1 and 2 are adopted as appropriate modifiers of the Surveillance Requirement. Furthermore, the results of air lock leakage testing, when performed, should be compared with the overall reactor building leakage from other sources to evaluate compliance with the Reactor Building Leakage Rate Testing Program (RBLRTP).

These NUREG requirements, adopted by the ITS, are additional restrictions on the operation of the unit consistent with NUREG-1430.

M12 The CTS requirements for the reactor building (RB) air locks (alternate ANO-1 terminology is the "personnel hatch" and the "emergency hatch") are provided by the RB integrity definition: CTS 1.7 a & b and by CTS 3.6.1 requirements for OPERABILITY.

The air lock door interlock requirements of NUREG 3.6.2 Condition B, including the Condition Notes 1 and 2, and the NUREG 3.6.2 Required Action (RA) B.3 Note are adopted by the ITS to provide specific guidance for this air lock feature. The CTS doesn't provide specific guidance for verification of air lock interlock function. NUREG Surveillance Requirement (SR) 3.6.2.2 will be adopted as appropriate and consistent with the significance of maintaining RB OPERABILITY.

The "reasons other than Condition A or B" air lock inoperable requirements of NUREG 3.6.2 Condition C are adopted by the ITS to provide specific guidance if the reason for an inoperable air lock is related to other than a door or interlock.

These NUREG requirements, adopted by the ITS, are additional restrictions on the operation of the unit consistent with NUREG-1430.

M13 CTS 3.6.4 addresses the reactor building (RB) internal pressure requirements but doesn't provide for surveillance of this parameter. It is proposed that NUREG Surveillance Requirement (SR) 3.6.4.1 be adopted in the ITS to require verification of RB internal pressure on a Frequency of 24 hours. Adoption of this Surveillance Requirement will replace present administrative verification of this parameter and provide appropriate ITS verification of safety analysis assumptions. This requirement is an additional restriction on the operation of the unit consistent with NUREG-1430.

M14 CTS 3.6.4 describes the reactor building (RB) internal pressure as not to exceed 3.0 psig or 5.5 inches Hg vacuum. Verifying RB pressure in inches of mercury, vacuum, is inconsistent with the NUREG format for reactor building (RB) pressure. The equivalent of 5.5 in. Hg vacuum is -2.7 psig, however, this value conflicts with the value assumed in the ANO-1 Emergency Core Cooling System (ECCS) analysis. The ANO-1 ECCS analysis is based on BAW-10103, Rev. 3 "ECCS Analysis of B&W's 177-FA Lowered Loop NSS", dated July, 1977 and lists 13.7 psia as an input assumption. The value of 13.7 psia equates to a value of -1.0 psig. To resolve these difficulties, the ITS will adopt the NUREG format for the RB pressure limits and the values consistent with the ECCS analysis. ITS LCO 3.6.4 and ITS SR 3.6.4.1 will provide RB pressure limits of $\geq$ -1.0 psig and $\leq$ +3.0 psig which is a lesser range than the CTS and is therefore more restrictive on unit operation. This more restrictive change makes the two limits of the range compatible with each other, with the control room indication of the RB pressure, with the appropriate analyses and with NUREG-1430.

3.6.4-2

3.6.4-3

3.6.5-6

M15 CTS 4.5.2.1.1 and CTS 4.5.2.2.2 describe requirements for Reactor Building (RB) Spray System and valve testing, however, there is no requirement for periodic verification of RB Spray System valve lineup. The ITS will adopt the requirements of NUREG SR 3.6.6.1 (as ITS SR 3.6.5.1) as an adequate method of verifying that the RB Spray System will be available if required. The Frequency of 31 days is consistent with the test frequency, in the CTS, of other portions of the RB Cooling System and Spray System. This surveillance is an additional restriction on the operation of the unit consistent with NUREG-1430.

M16 CTS 3.6.5 provides that the provisions of CTS 3.0.3 are not applicable. NUREG LCO 3.6.3 doesn't include such a note, so the requirements of NUREG LCO 3.0.3 are applicable. The requirements of NUREG LCO 3.6.3 will be adopted by the ITS since this LCO provides ACTIONS which will bring the unit to MODE 5 and place the unit in a condition which is more consistent with the ITS threshold for setting RB OPERABILITY. The requirement that LCO 3.0.3 is applicable for ITS 3.6.3 is an additional restriction on the operation of the unit consistent with NUREG-1430.

M17 Not used.

M18 CTS 3.3.6 provides that if the requirements of CTS 3.3.4 for two trains of reactor building (RB) cooling and two trains of RB spray and an OPERABLE Spray Additive System in MODES 1 and 2 can not be met then the unit shall be in MODE 3 within 36 hours. The ITS 3.6.5 Condition D and ITS 3.6.6 Condition B will adopt a Completion Time of 6 hours to be in MODE 3. This is an additional restriction on the operation of the unit consistent with NUREG-1430.

M19 CTS 3.3.6 provides actions if the requirements of CTS 3.3.1 for one OPERABLE reactor building (RB) cooling train and one OPERABLE RB spray train are not met during MODES 3 and 4. However, CTS 3.3.6 does not provide an explicit time for restoration when the requirements of CTS 3.3.1 cannot be met. A restoration Completion Time of 36 hours is adopted for ITS 3.6.5 Condition E, consistent with the format of similar requirements of NUREG-1430. The specifying of a time to restore operability where none was specified before makes this change more restrictive.

3.6.5-1

CTS 3.3.6 requires that the unit be in MODE 5 in 72 hours if the conditions of CTS 3.3.1 can not be met. ITS 3.6.5 Condition F is comparable and will adopt a Completion Time of 36 hours to be in MODE 5. The response time reduction from 72 hours to 36 hours is an additional restriction on the operation of the unit which is consistent with the requirements NUREG-1430.

M20 CTS 3.3.1, CTS 3.3.4, CTS 3.3.5, CTS 3.3.6 and CTS 3.3.7 together form a matrix of requirements for the reactor building (RB) spray and RB cooling trains during MODES 1, 2, 3, and 4. This CTS matrix specifically describes some combinations of OPERABLE and inoperable trains of the two (2) trains of RB cooling and two (2) trains of RB spray during MODES 1 and 2 and the respective required trains during MODES 3 and 4. The CTS matrix, however, doesn't specifically address the combinations described in NUREG 3.6.6 Condition F or its modified version ITS 3.6.5 Condition G. Therefore, CTS 3.3.6 is considered to provide guidance for actions when conditions are not specifically described unless CTS 3.0.3 is considered appropriate. When implementing CTS 3.3.6, if the combination of inoperable trains described in ITS 3.6.5 Condition G were discovered, then CTS 3.0.3 would be considered appropriate and would be entered immediately which is the same Required Action provided by ITS 3.6.5 Condition G. However, the CTS 3.0.3 requirement to be in MODE 3 is 13 hours whereas the ITS LCO 3.0.3 requirement is 6 hours. This reduced response time is an additional restriction on the operation of the unit consistent with NUREG-1430.

M21 Not used.

M22 CTS 3.6.6 provides 24 hours to be in cold shutdown (ITS MODE 5) upon failure to restore an inoperable reactor building (RB) isolation valve or close the other valve after 48 hours while the reactor is critical (i.e., MODE 1 or 2). However, CTS 3.6.6 does not require a time period to be in an intermediate MODE on the descent to MODE 5 although the relationship of CTS 3.6.6 to CTS 3.6.1 by the reactor building integrity requirement implies MODE 3 in 6 hours. NUREG 3.6.3 Required Action (RA) E.1 requires that the unit be in MODE 3 in 6 hours on the way to MODE 5 from a similar Condition. Along with other NUREG requirements, the NUREG 3.6.3 RA E.1 requirements will be adopted as ITS 3.6.3 RA D.1, which is a more restrictive condition on unit operation consistent with NUREG-1430.

M23 ANO-74 CTS 4.4.1.4 requires each remotely operated reactor building isolation valve to be stroke tested to the position required to fulfill their safety function every three months, unless such operation is not practical during plant operation. The latter valves shall be tested once every 18 months. The ANO-1 IST program, in accordance with 1988 OM Code Part 10, requires these valves to be tested nominally every three months. The ANO-1 IST program also utilizes guidance pertaining to cold shutdown testing contained in NUREG 1482. If full stroke testing is not practicable during plant operation, the program requires partial stroke testing during operation and full stroke testing quarterly during cold shutdowns of sufficient duration. If exercising is not practicable during plant operations, the program requires full stroke testing quarterly during cold shutdowns of sufficient duration, and so on. Deleting the CTS 4.4.1.4 requirement results in a more restrictive requirement in that the IST program requires valves that cannot be stroked during power operation to be stroked quarterly during cold shutdowns of sufficient duration. If cold shutdown conditions of sufficient duration were to be entered each quarter (i.e., 92 days since the last test was performed), full stroke testing would be required during each of those cold shutdowns. Although the CTS allows an 18 month frequency for these valves, our current practices are in compliance with the IST program.

TECHNICAL CHANGE—LESS RESTRICTIVE

L1 Not used.

L2 Not used.

L3 The general CTS 3.3.5 maintenance requirements which are applicable to an inoperable reactor building (RB) spray system and the RB cooling system, are revised to be consistent with specific NUREG-1430 requirements for an inoperable RB spray train or RB cooling train. CTS 3.3.5 allows a train of these systems to be made inoperable for up to 24 hours for maintenance, but only if the redundant train is demonstrated operable within 24 hours prior to beginning the maintenance. However, the performance of maintenance on one train does not change the basis for believing that the redundant train is OPERABLE, therefore, this requirement is omitted from the ITS. This change is consistent with NUREG-1430.

L4 Not used.

L5 The requirements of CTS 3.3.7(C) for one inoperable reactor building (RB) cooling train, CTS 3.3.7(D) for two inoperable RB cooling trains, and CTS 3.3.7(E) for one inoperable RB spray and one inoperable RB cooling train correspond, respectively, to the Conditions of ITS 3.6.5 Condition B, Condition C, and a combination of Condition A and B. The Required Actions of these conditions, in the CTS, include time periods to be in hot shutdown (ITS MODE 3) and then cold shutdown (ITS MODE 5). The ITS will divide the Required Action to be in MODE 3 from the requirement to be in MODE 5 as appropriate for the Applicability of the Condition. These ITS Required Action requirements are less restrictive because the CTS required the unit to be placed in MODE 5 even though the Applicability of CTS 3.3.7(C), (D), and (E) is MODES 1 and 2.

L6 The requirements of CTS 3.3.6 (by reference to CTS 3.3.4) for one inoperable reactor building (RB) spray train or an inoperable Spray Additive System correspond, respectively, to the Conditions of ITS 3.6.5 Condition A and ITS 3.6.6 Condition A. The Required Actions of CTS 3.3.6 include time periods to be in hot shutdown (ITS MODE 3) and then cold shutdown (ITS MODE 5). The ITS will divide the Required Action to be in MODE 3 from the requirement to be in MODE 5 as appropriate for the Applicability of the Condition. These ITS Required Action requirements are less restrictive because the CTS required the unit to be placed in MODE 5 even though the Applicability of CTS 3.3.6, in this context, is MODES 1 and 2.

L7 CTS 3.14.1 requires that two hydrogen recombiners be OPERABLE whenever reactor building (RB) integrity is required, that is, during ITS MODES 1, 2, 3 and part of 4. The NUREG 3.6.8 Applicability is MODES 1 and 2. The NUREG Applicability considers that in MODES 3 and 4 both the hydrogen production rate and the total hydrogen produced after a LOCA would be less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an accident requiring the hydrogen recombiners is low. Therefore, a requirement for OPERABLE hydrogen recombiners during MODES 3 and 4 will not be adopted by the ITS. This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L8 CTS 4.12.1a. requires verifying the OPERABILITY of each hydrogen recombiner system by system functional test at least once per 6 months. NUREG SR 3.6.8.1 (ITS SR 3.6.7.1) extends this Frequency to 18 months. Experience has shown that these components usually pass the Surveillance. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. This Frequency extension is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L9 CTS Table 4.1-3 item #6 requires that the sodium hydroxide tank solution concentration be sampled "quarterly and after each makeup." NUREG SR 3.6.7.3 requires a sampling Frequency of 184 days. The 184 day Frequency provided by the NUREG is sufficient to ensure that the NaOH concentration is within the established limits. This conclusion is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated from makeup sources) and the probability that any substantial variance in tank volume will be detected. This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L10 CTS 4.5.2.1.1(b) requires that the availability of the reactor building (RB) spray headers and nozzles be verified at least every five (5) years. NUREG SR 3.6.6.8 requires that each spray nozzle be verified unobstructed on a Frequency of every ten (10) years. Due to the passive nature of the design of the nozzles, a ten (10) year Frequency is considered adequate to detect obstruction of the nozzles and will be adopted by the ITS. This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L11 CTS 4.5.2.1.2(a) requires that the service water flow rate of each reactor building (RB) cooling train be tested on a frequency of at least once per 14 days. NUREG SR 3.6.6.3 requires a Frequency of 31 days. The Frequency of 31 days for testing of the service water flow is consistent with the CTS test frequency of other portions of the RB Cooling System. Furthermore, the service water supply to the RB cooling trains is considered reliable and there is a low probability of a significant degradation of flow occurring on a frequency of 31 days. Extension of this Frequency from 14 days to 31 days is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L12 The CTS requirements for RB penetrations do not provide alternate position verification methods for valves in high radiation areas. The ITS will adopt the NUREG SR 3.6.3.3 Note and the SR 3.6.3.4 Note allowing verification by administrative means for valves and blind flanges in high radiation areas. These allowances are adopted consistent with Industry ALARA practice while adequately addressing RB OPERABILITY requirements. Their adoption is a less restrictive condition on unit operation consistent with NUREG-1430.

L13 CTS 3.14 requires two (2) hydrogen recombiners to be OPERABLE whenever reactor building (RB) integrity is required, that is, during MODES 1, 2, 3, and part of 4. This direction implies that MODE changes are not allowed with only one (1) hydrogen recombiner OPERABLE. The ITS will adopt the note on NUREG 3.6.8 Required Action A.1 which makes NUREG LCO 3.0.4 not applicable when one hydrogen recombiner is inoperable. Accepting this Note will allow for MODE changes and unit operation during the Completion Time of 30 days adopted for ITS 3.6.7 Required Action A.1. This allowance is predicated on the availability of the other, 100% capacity, hydrogen recombiner, the small probability of a LOCA occurring and the amount of time available after a LOCA for operator action to prevent hydrogen accumulation from exceeding the flammability limit. This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L14 CTS 3.3.4(C) requires that the manual valves in the sodium hydroxide tank main discharge lines shall be locked open before the reactor is made critical (i.e., ITS MODE 2). The NUREG LCO 3.6.7 requirement is an OPERABLE system, which implies that the manual valves be properly positioned but not necessarily locked. NUREG SR 3.6.7.1 requires that the manual valves (and the other valves) in the system, which are not locked or otherwise secured, be verified in their proper position. Therefore, the ITS will delete the requirement to have the Spray Additive System manual valves locked open. This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L15 CTS 3.3.6, unless modified by CTS 3.3.7, provides that if the requirements of CTS 3.3.4 for two trains of reactor building (RB) cooling and two trains of RB spray and an OPERABLE Spray Additive System in MODES 1 and 2 can not be met then the unit shall be in MODE 3 within 36 hours. This 36 hour time period, without CTS 3.3.7 modification, is applicable and includes restoration time when one RB spray train or the Spray Additive System is inoperable in MODES 1 or 2. The ITS will adopt the NUREG 3.6.6 Condition A and the NUREG 3.6.7 Condition A Completion Time of 72 hours for restoration when these conditions exist (see DOC M8 for the requirement to be in MODE 3). This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L16 CTS 3.6.6 addresses the OPERABILITY of reactor building (RB) penetrations with two valves and includes the testing of the "other" valve when one of the valves is recognized as inoperable in a position other than closed. NUREG 3.6.3 Required Action A.1 requires that the affected flow path be isolated but does not require testing of the operable valve dependant on the position of the inoperable valve. ITS 3.6.3 Required Action A.1 will delete the CTS requirement to test the other valve and to isolate using only the Operable valve, and adopt the NUREG 3.6.3 requirements to isolate the flow path by isolating the penetration. This isolation can be made with a closed and deactivated automatic valve, a closed manual valve, a blind flange, or a check valve with the flow secured. This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L17 CTS 3.6.6 provides 24 hours to be in cold shutdown (ITS MODE 5) upon failure to restore an inoperable reactor building (RB) isolation valve or close the other valve, and does not require entering MODE 3 in any specific time. NUREG 3.6.3 Required Action (RA) E.2 provides a less restrictive 36 hours to be in MODE 5 from a similar Condition. However, NUREG 3.6.3 Required Action E.1 is also included and adds a more restrictive "be in MODE 3 [in] 6 hours." These shutdown actions are consistent with the shutdown sequence and times provided throughout the ITS, and are revised here for consistency. The NUREG 3.6.3 RA E.1 and E.2 requirements will be renumbered and adopted as ITS 3.6.3 Required Actions D.1 and D.2. These Completion Times are consistent with NUREG-1430.

L18 CTS 4.4.1.4 describes Isolation Valve Function Tests as being performed on "remotely operated" reactor building (RB) isolation valves. The "remotely operated" set of isolation valves are considered administratively equivalent to the set described as "each power operated and each automatic" isolation valve. However, NUREG Surveillance Requirement (SR) 3.6.3.5, which has been renumbered for adoption as ITS SR 3.6.3.3, has been modified by TSTF-46 to remove the valves identified as power operated. This is because there are valves credited as RB isolation valves which are power operated that do not receive an RB isolation signal. These power operated valves do not have an isolation time assumed in the accident analysis since they require operator action. Therefore, deleting a reference to power operated isolation valve time testing reduces the potential for misinterpreting the requirements of this SR while maintaining the assumptions of the accident analysis. This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L19 CTS 3.3.1(I) and CTS 3.3.4(D) require that the engineered safety features valves for the reactor building (RB) spray system, RB cooling system and spray additive system be OPERABLE or locked in the Engineered Safeguards (ES) position whenever the associated system or component was required to be OPERABLE. These requirements apply to automatic valves only since manual valves would be considered OPERABLE when they are in the appropriate position. Locking of a valve under these requirements is only appropriate if a valve is not OPERABLE. ITS LCOs 3.6.5 and 3.6.6 will retain these requirements as a condition of system OPERABILITY. However, NUREG-1430 and the ITS allow the ES valves to be verified OPERABLE by actuation to the correct position or by being locked, sealed or otherwise secured in position. The expanded options for administratively controlling valve position will be adopted by the ITS. This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

3.6.5-2

3.6.6-2

L20 CTS 3.6.5 requires that the reactor building (RB) manual isolation valves that are required to be closed, be confirmed closed and locked. ITS SR 3.6.3.2 and SR 3.6.3.3 require that the outside and inside reactor building manual isolation valves be verified closed unless locked, sealed or otherwise secured in position. This relaxes the requirement that all valves that are required to be closed be locked closed. Furthermore, the position verification requirements for valves that are closed and locked are allowed to be administratively controlled outside of Technical Specification requirements. The expanded options for controlling and verifying valve position will be adopted by the ITS. This is a less restrictive condition on unit operation which is adopted in the ITS consistent with NUREG-1430.

L21 NUREG 3.6.2 provides several allowances that the CTS do not explicitly provide. NUREG 3.6.2 Action Note 1 allows entry and exit to perform repairs on affected air lock components. NUREG 3.6.2 Actions Note 2 allows a separate Condition entry for each air lock. NUREG 3.6.2 Condition A Required Action Note 2 allows entry and exit for 7 days under administrative controls if both air locks are inoperable. These allowances have been incorporated in ITS 3.6.2. These less restrictive requirements allowing entry and exit are acceptable due to the relatively short periods of time the operable door is open during entry and exit and the low probability of an accident requiring reactor building isolation during these short periods of time. Separate Condition entry recognizes the fact that both airlocks may become inoperable at different times, thus requiring accelerated actions in responding to the second inoperability. The ITS contains the actions required to ensure reactor building integrity can be maintained, consistent with NUREG-1430.

3.6.2-1

3.6.2-4

3.6.3-8 L22 NUREG 3.6.3 Condition C provides the Required Actions and associated Completion Times for inoperable penetration flow paths with one reactor building isolation valve inoperable for General Design Criteria (GDC) 57 penetrations associated with closed systems. A GDC 57 penetration relies upon the existence of a closed system as a passive isolation barrier in lieu of a second isolation valve. Currently, in the event of reactor building isolation valve inoperability associated with a closed system,

CTS 3.6.6 would not apply since the actions require closing the operable valve and the actions of CTS 3.6.1 would be conservatively entered.

The proposed ITS incorporate the NUREG 3.6.3 Condition C requirements, as modified by generic change TSTF-30, Rev 3. This results in extending the Completion time from the CTS 3.6.1 allowed 1 hour to 72 hours. This less restrictive change is acceptable due to the proposed compensatory actions to ensure an appropriate level of reactor building integrity is maintained and is consistent with NUREG-1430, as modified by TSTF-30, Rev 3.

3.6.5-8 L23 CTS 4.5.2.1.1.a and CTS 4.5.2.1.2.c.1 require that a "test signal" be applied to demonstrate actuation of the reactor building spray system and the reactor building emergency cooling system, respectively. NUREG SRs 3.6.6.5, 3.6.6.6, and 3.6.6.7 allow these tests to be performed by use of "an actual or simulated actuation signal." One interpretation of the CTS requirements would allow the use of a simulated actuation signal, but would not allow crediting an actual actuation signal. The allowance to credit an actual actuation signal has been incorporated in ITS SRs 3.6.5.5, 3.6.5.6, and 3.6.5.7. This change is consistent with NUREG-1430 and is acceptable because an actual actuation signal would result in actuating the entire system, such as is done with a simulated test signal.

L24 CTS 4.26.2 requires that a leakrate of the reactor building purge supply and exhaust isolation valves be verified within acceptable limits prior to exceeding conditions which require establishment of reactor building integrity (RCS temperature ≥200°F, RCS pressure ≥300 psig, and nuclear fuel in the core), unless the test has been successfully completed within the last three months. According to the CTS 4.26 Bases, this surveillance requirement frequency was based on Generic Issue B-20 "Containment Leakage Due to Seal Deterioration", which was concerned with isolation valves that have resilient seals. In 1998, ANO-1 made a design change, which replaced all the Reactor Building Purge valves. The new valves, manufactured by Atwood & Morrill Co., Inc., have multilayer metallic wafer seats. Therefore, the issue related to resilient seated valves described in Generic Issue B-20 is no longer a concern.

3.6.1-4
3.6.3-3
3.6.3-4
3.6.3-13

ANO-1 has determined that NUREG SR 3.6.3.6 should not be adopted since it is specific to resilient seated purge valves, the surveillance testing criteria, including the CTS specified test frequency, have been retained in ITS section 5.5.16, "Reactor Building Leakage Testing Program" and still remain under the control of 10 CFR 50.36. The incorporation of the testing requirements for the reactor building purge valves in ITS 5.5.16 retains the CTS 4.26.2 required frequency of performance.

LESS RESTRICTIVE—ADMINISTRATIVE DELETION OF REQUIREMENTS

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

	CTS Location	New Location
3.6.1-1	1.7 a, b, c, & d	B3.6.1 BKG
	1.7 a. & e.	B3.6.1 LCO
	1.7 a. & b.	B3.6.2 LCO
	1.7 c.	B3.6.3 LCO
	3.3.4 (A)	B3.6.5 BKG
	3.3.4 (D)	B3.6.5 BKG & B3.6.6 BKG
	3.14.1	B3.6.7 BKG
	3.23.1	B3.6.3 BKG
	4.5.2.1.1 (b)	B3.6.5 SR 3.6.5.8
	4.5.2.1.1 (c)	B3.6.5 SR 3.6.5.5 & 3.6.5.6
	4.5.2.1.2(a)(2)	B3.6.5 BKG & SR 3.6.5.3
3.6.5-9	4.5.2.1.2(b)(1)	B3.6.5 SR 3.6.5.2
	4.5.2.1.2(c)(3)	B3.6.5 SR 3.6.5.7
3.6.5-3	4.5.2.2.1	B3.6.5 SR 3.6.5.4
	4.12.1 a.	B3.6.7 SR 3.6.7.1
	4.12.1 b.2.	B3.6.7 SR 3.6.7.2
	4.12.1 b.3.	B3.6.7 SR 3.6.7.3
	4.26.1	B3.6.3 SR 3.6.3.1

LA2 This information has been moved to the Technical Requirements Manual (TRM). This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The TRM will be controlled by 10 CFR 50.59 and 10 CFR 50.71, as applicable. This change is consistent with NUREG-1430.

CTS Location	New Location
4.12.1.b.1	TRM

LA3 This information has been moved to a licensee controlled document such as the Reactor Building Leakage Rate Testing Program (RBLRTP), In-Service Testing (IST),and plant procedures, etc. This information provides details of the method of implementation that are not directly pertinent to the actual requirement. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The details relocated to the RBLRTP, RBTSP, and IST will be controlled by 10 CFR 50.59. The CTS location and ITS location for each of these items is listed below. This change is consistent with NUREG-1430.

	CTS Location	New Location	Existing IST Frequency
3.6.5-3	4.5.2.2.1 & 4.5.2.2.2	IST	3 months

3.6.1
3.6.2
3.6.3

1.7 REACTOR BUILDING

Reactor building integrity exists when the following conditions are satisfied:

A1 A18

3.6.1 LCO
3.6.2 LCO

a. The equipment hatch is closed and sealed and both doors of the personnel lock and emergency lock are closed and sealed, or b. below. (OPERABLE) (OPERABLE) LA1 Bases A18

3.6.2 LCO

b. At least one door on each of the personnel lock and emergency lock is closed and sealed during personnel access or repair. (are OPERABLE) LA1 Bases A18 A19

3.6.3-2
3.6.3-11
3.6.3 LCO
3.6.3 Act. Note 1

c. All non-automatic reactor building isolation valves and blind flanges are closed as required. (manual valves) (OPERABLE) LA1 Bases A22

3.6.1-1
3.6.1 + 3.6.3 LCO

d. All automatic reactor building isolation valves are operable or deactivated in the closed position. A18

SR 3.6.1.1

e. The reactor building leakage determined at the last testing interval satisfies specification 4.4.1. (is in accordance with the RBLRTP) LA1 Bases A20

1.8 FIRE SUPPRESSION WATER SYSTEM

LA1 Bases

<Later> (1.0)

The fire suppression water system consists of: water sources, pumps, and distribution piping with associated sectionalizing isolation valves. Such valves include the hose standpipe shutoff valves and the first valve ahead of the water flow alarm device or each sprinkler system.

1.9 STAGGERED TEST BASIS

Later

A staggered test basis shall consist of:

a. A test schedule for n systems, subsystems, trains or designated components obtained by dividing the specified test interval into n equal subintervals.

b. The testing of one system, subsystem, train or designated component at the beginning of each subinterval.

3.3 EMERGENCY CORE COOLING, REACTOR BUILDING EMERGENCY COOLING AND REACTO BUILDING SPRAY SYSTEMS

~~Applicability~~

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

~~Objectivity~~

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

A1

Specification

in MODES 1,2,3 & 4

3.6.5 LCO
3.6.5 Appl. & (Later) (3.3D, 3.5, 3.7)
3.6.5 LCO Note
3.6.5 LCO Note

3.3.1 The following equipment shall be operable whenever containment integrity is established as required by Specification 3.6.1:

M1 & Later

(A) One reactor building spray pump and its associated spray nozzle header.

(B) One train of reactor building emergency cooling.

(Later) (3.7) ~~(C) Two out of three service water pumps shall be operable, powe from independent essential buses, to provide redundant and independent flow paths.~~ Later

(Later) (3.5) ~~(D) Two engineered safety feature actuated Low Pressure Injectio (LPI) pumps shall be operable.~~ Later

(Later) (3.7) ~~(E) Both low pressure injection coolers and their cooling water supplies shall be operable.~~ Later

(Later) (3.3D) ~~(F) Two Borated Water Storage Tank (BWST) level instrument chanr shall be operable.~~ Later

(Later) (3.5) ~~(G) The borated water storage tank shall contain a level of 48.2 ± 1.8 ft. (387,400 ± 17,300 gallons) of water having a concentration of 2470 ± 200 ppm boron at a temperature not less than 40F. The manual valve on the discharge line from the borated water storage tank shall be locked open.~~ Later

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

3.6.5
3.6.6

3.6.5 LCO + <LATER> (3.5, 3.7)

(I) The engineered safety features valves associated with each of the above systems shall be operable or locked (Sealed, or otherwise secured) in the ES position. +Later L19

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

<LATER> (3.5)

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

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

LATER

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

(A) The two core flooding tanks shall each contain an indicated minimum of 13 ± 0.4 feet (1840 ± 30 ft³) of borated water at 600 ± 25 psig.

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

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

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

3.3.4 The reactor shall not be made critical (in MODE 1 or 2) unless the following equipment in addition to 3.3.1, 3.3.2, and 3.3.3 above is operable. A1

3.6.5 LCO + Appl.

3.6.6 LCO + Appl.

(A) Two reactor building spray pumps and their associated spray nozzle headers and two trains of reactor building emergency cooling. The two reactor building spray pumps shall be powered from operable independent emergency buses and the two reactor building emergency cooling trains shall be powered from operable independent emergency buses. LA1 Bases

SR 3.6.6.2
SR 3.6.6.3

(B) The sodium hydroxide tank shall (Verify once each 184 days that) contain a volume of ≥9,000 gallons of sodium hydroxide at a concentration >5.0 wt% and <16.5 wt%. M7

3.6.6 LCO

(C) All manual valves in the main discharge lines of the sodium hydroxide tanks shall be locked open. L14

<Add SR 3.6.6.1 + SR 3.6.6.4> M10

<Add SR 3.6.5.1>

3.6.5-2

ANO-71

3.6.5
3.6.6

3.6.5 LCO
3.6.6 LCO
<Later>
(3.5 3.7)

(D) Engineered safety feature valves and interlocks associated with 3.3.1, 3.3.2, and 3.3.3 shall be operable or locked, in the ES position. <Sealed, or otherwise secured> L19 & LATER

3.3.5 ~~Maintenance shall be allowed during power operation on any component(s) in the high pressure injection, low pressure injection, service water reactor building spray and reactor building emergency cooling~~ L3 Later

<Later>
(3.5, 3.7)

& <LATER> (3.5, 3.7) systems which will not remove more than one train of each system from service. Maintenance shall not be performed on components which would make the affected system train inoperable for more than 24 consecutive hours. Prior to initiating maintenance on any component of a train in any system, the redundant component of that system shall be demonstrated to be operable within 24 hours prior to the maintenance. — & LATER — L3

<LATER> (3.3D, 3.5, 3.7) — LATER

3.3.6 If the conditions of Specifications 3.3.1, 3.3.2, 3.3.3, 3.3.4 and 3.3.5 cannot be met except as noted in 3.3.7 below, reactor shutdown shall be initiated and the reactor shall be in hot shutdown condition within 30 hours, and, if not corrected, in cold shutdown condition within an additional 72 hours. — 3.6.5 Cond. A — 3.6.5 Cond. D — restore in 72 hrs or — G — M18 — MODE 3 — L6 — A1 — L15

3.3.7 Exceptions to 3.3.6 shall be as follows:

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

<LATER> (3.5) (B) If the conditions of Specification 3.3.3(D) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other CET instrument channel (pressure of level) shall be operable. — LATER

3.6.5 Cond. B (C) If the conditions of Specification 3.3.4(A) cannot be met because one train of the required reactor building emergency cooling is inoperable but both reactor building spray systems are operable, restore the inoperable train of cooling to operable status within 7 days or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours. — A6 — A1 — MODE 3 — L5

3.6.5 Cond. D

3.6.5 Cond. C (D) If the conditions of Specification 3.3.4(A) cannot be met because two trains of the required reactor building emergency cooling are inoperable but both reactor building spray systems are operable, restore at least one train of cooling to operable status within 72 hours or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours. Restore both above required cooling trains to operable status within 7 days of initial loss or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours. — A6 — A1 — MODE 3 — L5

3.6.5 Cond. D

3.6.5 Cond. B

<Add 3.6.5 RA A.1 & B.1 CT of 10 days> — M8

See Pg. 38-1

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

<LATER> (3.3.D, 3.5, 3.7)

LATER

3.3.6 If the conditions of Specifications 3.3.1, ~~3.3.2, 3.3.3,~~ 3.3.4 and 3.3.5 cannot be met ~~except as noted in 3.3.7 below, reactor shutdown shall be initiated and the reactor shall be in hot shutdown condition within 36 hours, and,~~ if not corrected, in ~~cold shutdown~~ condition within an additional ~~72~~ 36 hours. In 36 hours MODE 5 A7 A1 M19

3.6.5-1

3.3.7 Exceptions to 3.3.6 shall be as follows:

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

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

See Pg. 38-1

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

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

See Pg. 38-1

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

<LATER> (3.3 D, 3.5, 3.7)

LATER

3.6.6 Cond. A 3.3.6 If the conditions of Specifications 3.3.1, ~~3.3.2, 3.3.3,~~ 3.3.4 and 3.3.5 cannot be met except as noted in 3.3.7 below, ~~reactor shutdown shall be initiated and~~ the reactor shall be in ~~hot shutdown~~ condition within ~~36~~ 6 hours, ~~and, if not corrected, in cold shutdown condition within an additional 72 hours.~~

M18 MODE 3 L6 4L L15

3.6.6 Cond. B restore in 72 hours or

3.3.7 Exceptions to 3.3.6 shall be as follows:

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

See Pg. 38-1

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

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

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

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

3.6.5 Cond. A

3.6.5 Cond. D

A14

MODE 3 A1

L5

MODE 3 A1

L5

<Add 3.6.5 Cond. G>

M20

Bases

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

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

A2

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

The borated water storage tank is used for three purposes:

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

(B) As an alternate supply of borated water for reaching cold shutdown.([3])

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

370,100 gallons of borated water are supplied for emergency core cooling and reactor building spray in the event of a loss-of-coolant accident. This amount fulfills requirements for emergency core cooling. Approximately 16,000 gallons of borated water are required to reach cold shutdown. The original nominal borated water storage tank capacity of 380,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature to prevent crystallization and local freezing of the boric acid. The minimum required BWST boron concentration of 2270 ppm assures that the core will be maintained at least 1 percent Δk/k subcritical at 70°F without any control rods in the core. A2

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

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

Specification 3.3.4 assures that prior to going critical the redundant train of reactor building emergency cooling and spray train are operable.

The spray system utilizes common suction lines with the low pressure injection system. If a single train of equipment is removed from either system, the other train must be assured to be operable in each system.

The volume specified by 3.3.4.B is the safety analysis volume and does not contain allowances for instrument uncertainty. 9,000 gallons corresponds to a level of approximately 26 feet at a temperature of 77°F and a NaOH concentration of 5.0 wt%. No maximum volume is specified as the value used as the maximum volume in the safety analysis bounds the physical size of the NaOH tank. Additional allowances for instrument uncertainties, as determined in Reference 6, are incorporated in the operating procedures associated with the level instrumentation used in the control room.

When the reactor is critical, maintenance is allowed per Specification 3.3.5. Operability of the specified components shall be based on the results of testing as required by Technical Specification 4.5. The maintenance period of up to 24 hours is acceptable if the operability of equipment redundant to that removed from service is demonstrated within 24 hours prior to removal. Exceptions to Specification 3.3.6 permit continued operation for seven days if one of two BWST level instrument channels is operable or if either the pressure or level instrument channel in the CFT instrument channel is operable.

In the event that the need for emergency core cooling should occur, functioning of one train (one high pressure injection pump, one low pressure injection pump, and both core flooding tanks) will protect the core and in the event of a main coolant loop severance, limit the peak clad temperature to less than 2200°F and the metal-water reaction to that representing less than 1 percent of the clad.

The service water system consists of two independent but interconnected, full capacity, 100% redundant systems, to ensure continuous heat removal.(4)

One service water pump is required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant accident.

A10-71

~~REFERENCES~~

~~(1) FSAR, Section 14.2.5~~

~~(2) FSAR, Section 3.2~~

~~(3) FSAR, Section 9.5.2~~

~~(4) FSAR, Section 9.3.1~~

~~(5) FSAR, Section 6.3~~

~~(6) ANO Calculation 91-E-0019-01~~

A2

3.6.1
3.6.2
3.6.3
3.6.4

3.6 REACTOR BUILDING

Applicability

Applies to the operability of the reactor building. A1

Objective

To assure reactor building operability.

Specification

3.6.1 LCO
3.6.1 APPL
3.6.2 APPL
3.6.3 APPL

3.6.1 The reactor building shall be operable in MODES 1,2,3,4 whenever all three (3) of the following conditions exist: M1

a. Reactor coolant pressure is 300 psig or greater.

b. Reactor coolant temperature is 200°F or greater.

c. Nuclear fuel is in the core.

3.6.2 Act Note 3
3.6.1 RA A.1/B.1/B.2
3.6.2 RA A.1/D.1/D.2
3.6.3 RA D.1/D.2

With the reactor building inoperable, restore the reactor building to operable status within one hour or be in at least MODE 3 Hot Standby within the next 6 hours and in MODE 5 Cold Shutdown within the following 30 hours. M2 M22 A1

LATER (3.9)

3.6.2 Reactor building integrity shall be maintained when the reactor coolant system is open to the reactor building atmosphere and the requirements for a refueling shutdown are not met. The provisions of Specification 3.0.3 are not applicable. LATER

3.6.3 Positive reactivity insertions which would result in the reactor being subcritical by less than 1% Δk/k shall not be made by control rod motion or boron dilution whenever reactor building integrity is not in force. The provisions of Specification 3.0.3 are not applicable. A3

3.6.4 LCO
3.6.4 APPL
3.6.4 COND A
3.6.4 RA B.1/B.2

3.6.4 The reactor shall not be taken critical or remain critical in MODES 1,2,3,4 if the reactor building internal pressure exceeds 3.0 psig or a vacuum of 5.5 inches H2O. With the reactor critical restore the containment pressure to within its limits within one hour or be in at least Hot Standby MODE 3 within the next 6 hours and in Cold Shutdown MODE 5 within the following 30 hours. M3 M14 -1.0 psig M2 A1

3.6.5 Prior to criticality following a refueling shutdown a check shall be made to confirm that all manual reactor building isolation valves which should be closed are closed and locked and blind flanged if not locked, sealed, secured as required. The provisions of Specification 3.0.3 are not applicable. L20 M16

SR 3.6.3.2: (for outside RB): Every 31 days

SR 3.6.3.3: (for inside RB): Prior to entering MODE 4 from MODE 5 if not performed in previous 92 days M4

3.6.3-6

CTS INSERT 54A

<CTS INSERT 54A>

3.6.2-1	<Add 3.6.2 ACTIONS Notes 1, 2, & 3>	L21
	<Add 3.6.2 Condition A Note 1>	A9
3.6.2-4	<Add 3.6.2 Condition A Note 2>	L21
	<Add 3.6.2 Required Action A.2>	M11
	<Add 3.6.2 Required Action A.3 & Note>	M11
	<Add 3.6.2 Condition B with 2 Notes, B.1, B.2 & B.3>	M12
	<Add 3.6.2 Condition C with RA's C.1, C.2, C.3, & CT's>	M12
	<Add SR 3.6.3.2 Note & SR 3.6.3.3 Note>	L12
	<Add SR 3.6.4.1>	M13
3.6.3-12	<Add 3.6.3 Action Note 4>	A1

<ADD: 3.6.3 ACTIONS NOTES 2 & 3> M5
<ADD: 3.6.3 COND A NOTE> A5
<ADD: 3.6.3 RA A.2, & NOTES & CTs> M5
<ADD: 3.6.3 COND B & NOTE> A5
<ADD: 3.6.3 ACTION C & NOTES> L22

3.6.3

3.6.3-8

3.6.3 APPL 3.6.6 If, ~~while the reactor is critical,~~ (MODES 1,2,3,4) a ~~reactor building~~ isolation valve is determined to be inoperable in a position other than the closed position, the other reactor building isolation valve (except for check valves) in the line shall be tested to insure operability. ~~If the inoperable valve is not restored within 48 hours, the reactor shall be brought to the cold shutdown condition within an additional 24 hours or the operable valve will be closed.~~ (MODE 5) (MODE 3: 6 hrs, MODE 5: 36 hrs)

M3 L16 A1 A1 L17

3.6.3 RA A.1
3.6.3 RA D.2
3.6.3 RA D.1

Bases

Included in reactor building operability are both the reactor building integrity as defined in Specification 1.7 and the reactor building structural integrity. Structural integrity limitations as described in the ANO Containment Inspection Program ensure the reactor building will be maintained comparable to the original design standards throughout the facility life span. Visual and other required examinations of tendons, anchorages and surfaces are performed periodically in accordance with station procedures. These procedures embody applicable requirements of the 1992 Edition with the 1992 Addenda of Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code as set forth in 10 CFR 50.55a(g)(6)(ii)(B). Any degradations exceeding the Containment Inspection Program acceptance criteria during inspection surveillances will be reviewed under an engineering evaluation within 60 days of the completion of the inspection to determine what impact the degradation has on overall containment operability, if any.

The reactor coolant system conditions of cold shutdown assure that no steam will be formed and hence there will be no pressure buildup in the reactor building if the reactor coolant system ruptures.

The selected shutdown conditions are based on the type of activities that are being carried out and will preclude criticality in any occurrence.

The reactor building is designed for an internal pressure of 59 psig and an external pressure 3.0 psi greater than the internal pressure. The design external pressure of 3.0 psi corresponds to a margin of 0.5 psi above the differential pressure that could be developed if the building is sealed with an internal temperature of 110°F and the building is subsequently cooled to an internal temperature of less than 50°F.

When reactor building integrity is established, the limits of 10 CFR 100 will not be exceeded should the maximum hypothetical accident occur.

A2

REFERENCE

FSAR, Section 5.

3.14 HYDROGEN RECOMBINERS

~~Applicability~~

~~Applies to the operating status of the hydrogen recombiner systems.~~ A1

~~Objective~~

~~To ensure that the hydrogen recombiner systems will perform within acceptable levels of efficiency and reliability.~~

Specification LA1 Bases

3.6.7 LCO Appl. 3.14.1 Two ~~independent~~ hydrogen recombiner systems shall be operable ~~whenever reactor building integrity is required.~~ in MODES 1&2 L7

RA.A.1 3.14.2 With one hydrogen recombiner system inoperable, restore the inoperable system to operable status within 30 days or the reactor shall be placed in the ~~hot shutdown~~ MODE 3 condition within the next 6 hours. A1 A1

RA.B.1

<Later> (3.3D) ~~3.14.3 Hydrogen concentration instruments shall be operable.~~ Later

~~3.14.4 With one of two hydrogen concentration instruments inoperable, restore the inoperable analyzer to OPERABLE status within 30 days or be in at least hot shutdown within the next 6 hours.~~

~~Bases~~

~~The hydrogen recombiner systems are designed to operate as necessary to limit the hydrogen concentration in the reactor building following a Loss of Coolant Accident.~~ A2

~~The system is composed of two redundant 100% capacity Internal Electrical Hydrogen Recombiners, manufactured by Westinghouse.~~

<Add 3.6.7 RA A.1 Note> L13

3.23 REACTOR BUILDING PURGE VALVES

APPLICABILITY

This specification applies to the reactor building purge supply and exhaust isolation valves.

OBJECTIVE

To specify that reactor building isolation purge valves be closed whenever containment integrity is required by TS 3.6.1.

A1

LA1 Bases

SPECIFICATION

3.6.3 LCO Appl. ACT. Note 1

3.23.1 The reactor building purge supply and exhaust isolation valves shall be closed and handswitch keys removed whenever containment integrity is required by TS 3.6.1. in MODES 1,2,3,4

M1

BASES

The reactor building supply and exhaust isolation valves are required to be closed during normal plant operation in order to ensure reactor building integrity. Purging is allowed only when containment integrity is not required by TS 3.6.1.

A2

Table 4.1-2
Minimum Equipment Test Frequency

	Item	Test	Frequency	
(Later) (3.1)	1. Control Rods	Rod Drop Times of all Full Length Rods 1/	Each Refueling Shutdown	LATER
	2. Control Rod Movement	Movement of Each Rod	Every Two Weeks Above Cold Shutdown Conditions	
(Later) (3.4B)	3. Pressurizer Code Safety Valves	Setpoint	One Valve Every 18 Month	LATER
(Later) (3.7)	4. Main Steam Safety Valves	Setpoint	Four Valves Every 18 Mon	LATER
	5. Refueling System Interlocks	Functioning	Start of Each Refueling Shutdown	R TRM
(Later) (3.4B)	6a. Reactor Coolant System Leakage	Evaluate	Daily	LATER
	b. Reactor Coolant System Pressure Isolation Valves	Leakage Test Per Table 3.1.6.9	See Notes 1 & 2	
	7. Emergency-powered Pressurizer Heaters	Power availability	Daily	
		Heater capacity functional test	Every 18 Months	A16
SR 3.6.3.5	8. Reactor Building Isolation Trip	Functioning Verify auto. RB iso. valves actuate on act. or Sim. Signal	Every 18 Months	
(Later) (3.7)	9. Service Water Systems	Functioning	Every 18 Months	LATER
	10. Spent Fuel Cooling System	Functioning	Every 18 Months when irradiated fuel is in the pool	R TRM

(Later) (3.1) 1/ Same as tests listed in Section 4.7 — Later

Notes:

(Later) (3.4B)

(1) Leak testing for each valve shall be individually accomplished to demonstrate operability following each refueling, following each time the plant is placed in a cold shutdown condition if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement. — LATER

(2) Whenever integrity of a pressure isolation valve listed in Table 3.1.6.9 cannot be demonstrated the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in the high pressure piping shall be recorded daily.

Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

Item	Test	Frequency	
1. Reactor Coolant Samples	a. Gamma Isotopic Analysis	a. Bi-weekly (7)	LATER (3.4B)
	b. Gross Activity Determination	b. 3 times/week and at least every third day (1)(6)(	
	c. Gross Radioiodine Determination	c. Weekly (3)(6)(7)	
	d. Dissolved Gases	d. Weekly (7)	LATER (3.1)
	e. Chemistry (Cl, F, and O_2)	e. 3 times/week (8)	LATER (3.4A)
	f. Boron Concentration	f. 3 times/week	LATER (3.9) & (R) TRM
	g. Radiochemical Analysis for $\bar{E}$ Determination (2) (4)	g. Monthly (7)	LATER (3.4B)
2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly and after each makeup	LATER (3.5)
3. Core Flooding Tank Sample	Boron Concentration	Monthly and after each makeup	
4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly and after each makeup (9)	LATER (3.7)
5. Secondary Coolant Samples	a. Gross Radioiodine Concentration	a. Weekly (5)(7)(10	(R) TRM
	b. Isotopic Radioiodine Concentration (4)	b. Monthly (7)(10)	(M7)
SR 3.6.6.3 6. Sodium Hydroxide Tank Sample	Sodium Hydroxide Concentration	Quarterly and after each makeup	(L9) every 184 days

Notes:

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units μCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled. Whenever t gross radioactivity concentration exceeds 10% of the limit specified i the Specification 3.1.4.1 or increases by 10 μCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established. (Later) (3.4B)

3.6.1
3.6.2
3.6.3

3.6 **4.4 REACTOR BUILDING**

4.4.1 Reactor Building Leakage Tests

~~**Applicability**~~

~~Applies to the reactor building.~~

~~**Objective**~~

~~To verify that leakage from the reactor building is maintained within allowable limits.~~ A1

Specification

SR 3.6.1.1 4.4.1.1 Integrated leakage rate tests shall be conducted and visual inspections performed in accordance with the Reactor Building Leakage Rate Testing Program.

~~4.4.1.1.1 Deleted~~
~~4.4.1.1.2 Deleted~~
~~4.4.1.1.3 Deleted~~ A1

SR 3.6.1.1 4.4.1.1.4 Integrated leakage rate testing frequencies shall be in accordance with the Reactor Building Leakage Rate Testing Program.

~~4.4.1.1.5 Deleted~~
~~4.4.1.1.6 Deleted~~
~~4.4.1.1.7 Deleted~~ A1

SR 3.6.1.1
SR 3.6.2.1 4.4.1.2 Local leakage rate tests shall be conducted in accordance with the Reactor Building Leakage Rate Testing Program.

~~4.4.1.2.1 Deleted~~
~~4.4.1.2.2 Deleted~~
~~4.4.1.2.3 Deleted~~
~~4.4.1.2.4 Deleted~~ A1

SR 3.6.1.1
SR 3.6.2.1 4.4.1.2.5 Local leakage rate testing frequencies shall be in accordance with the Reactor Building Leakage Rate Testing Program. A1

~~4.4.1.3 Deleted~~ LA3 IST

SR 3.6.3.4 4.4.1.4 Isolation Valve Functional Tests L18

~~Every three months,~~ ~~remotely operated~~ Automatic reactor building isolation valves shall be stroked, to the position required to fulfill their safety function and timed ~~unless such operation is not practical during plant operation. The latter valves shall be tested once every 18 months.~~ A17 M23

~~4.4.1.5 Deleted~~ A1

ANO-74

⟨Add SR 3.6.2.1 Notes⟩ M11

⟨Add SR 3.6.2.2⟩ M12

A2

Bases (1)

The reactor building is designed for an internal pressure of 59 psig and a steam-air mixture temperature of 285°F.

The peak calculated reactor building pressure for the design basis loss of coolant accident, P_a, is 54 psig. The maximum allowable reactor building leakage rate, L_a, shall be 0.20% of containment air weight per day at P_a.

The reactor building will be periodically leakage tested in accordance with the Reactor Building Leakage Rate Testing Program. These periodic testing requirements verify the reactor building leakage rate does not exceed the assumptions used in the safety analysis. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 L_a for the combined Type B and Type C leakage, and ≤ 0.75 L_a for overall Type A leakage. At all other times between required leakage tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a.

REFERENCE

(1) FSAR, Sections 5 and 13.

A23

3.6.5-4

4.5.2 Reactor Building Cooling Systems

~~Applicability~~

~~Applies to testing of the reactor building emergency cooling systems.~~ A1

~~Objective~~

~~To verify that the reactor building emergency cooling systems are operable.~~

Specification

3.6.5-8

4.5.2.1 System Tests

4.5.2.1.1 Reactor Building Spray System

SR 3.6.5.5
SR 3.6.5.6

(a) Once every 18 months, a system test shall be conducted to demonstrate proper operation of the system. ~~A test~~ An actual or simulated actuation signal will be applied to demonstrate actuation of the reactor building spray system (except for reactor building inlet valves to prevent water entering nozzles). L23

SR 3.6.5.8

~~(b) Station compressed air or smoke will be introduced into the spray headers to verify the availability of the headers and spray nozzles at least every five years.~~ 10 LA1 Bases L10

AND-73

SR 3.6.5.5
SR 3.6.5.6

~~(c) The test will be considered satisfactory if visual observation and control board indication verifies that all components have responded to the actuation signal properly.~~ LA1 Bases

4.5.2.1.2 Reactor Building Cooling System 31 L11

SR 3.6.5.3

(a) At least once per ~~14~~ days, each reactor building emergency cooling train shall be tested to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

SR 3.6.5.3

(1) Verifying a service water flow rate of ≥ 1200 gpm to each train of the reactor building emergency cooling.[1] A12

~~(2) Addition of a biocide to the service water during the surveillance in 4.5.2.1.2.a.1 above, whenever service water temperature is between 60F and 80F.~~ LA1 Bases

SR 3.6.5.2

(b) At least once per 31 days, each reactor building emergency cooling train shall be tested to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

3.6.5-9

~~(1) Starting (unless already operating) each operational cooling fan from the control room.~~ LA1 Bases

A12

~~[1] Surveillance Requirement 4.5.2.1.2(a)(1) will not be performed on the green train of the reactor building emergency cooling system until cooling fan VSF-1D is repaired and the green train is returned to normal configuration. This note will remain in effect until July 14, 1995.~~

A23

3.6.5-4

(2) Verifying that each operational cooling fan operates for at least 15 minutes.

SR 3.6.5.7 & LATER (3.7)

3.6.5-8

(c) Once every 18 months, a system test shall be conducted to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

L23

SR 3.6.5.7

(1) ~~A test~~ An actual or simulated actuation signal will be applied to actuate the reactor building emergency cooling operation.

<Later> (3.7)

~~(2) Verification of the engineered safety features function of the service water system which supplies the reactor building emergency coolers shall be made to demonstrate operability of the coolers.~~ Later

SR 3.6.5.7 & <Later> (3.7)

(3) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly.

LA1 Bases & Later

4.5.2.2 Component Tests

4.5.2.2.1 Pumps

SR 3.6.5.4

3.6.5-3

~~At intervals not to exceed 3 months~~ the reactor building spray pumps shall be started and operated to verify proper operation. Acceptable performance will be indicated if the pump starts, operates for fifteen minutes, and the discharge pressure and flow are within ±10% of a point on the pump head curve.

LA3 IST

LA1 Bases

4.5.2.2.2 Valves

~~At intervals not to exceed three months each engineered safety features valve in the reactor building spray and reactor building emergency cooling system and each engineered safety features valve associated with reactor building emergency cooling in the service water system shall be tested to verify that it is operable.~~

LA3 IST

<LATER> (3.7) & <LATER> (3.7)

LATER LA3 IST & LATER

<Add SR 3.6.5.1>

M15

Bases

~~The reactor building emergency cooling system and reactor building spray system are redundant to each other in providing post-accident cooling of the reactor building atmosphere to prevent the building pressure from exceeding the design pressure. As a result of this redundancy in cooling capability, the allowable out of service time requirements for the reactor building emergency cooling system have been appropriately adjusted. However, the allowable out of service time requirements for the reactor building spray system have been maintained consistent with that assigned other inoperable engineered safeguard equipment since the reactor building spray system also provides a mechanism for removing iodine from the reactor building atmosphere.~~

A2

3,6,5

Addition of a biocide to service water is performed during reactor building emergency cooler surveillance to prevent buildup of Asian clams in the coolers when service water is pumped through the cooling coils. This is performed when service water temperature is between 60F and 80F since in this water temperature range Asian clams can spawn and produce larva which could pass through service water system strainers.

A2

The delivery capability of one reactor building spray pump at a time can be tested by opening the valve in the line from the borated water storage tank, opening the corresponding valve in the test line, and starting the corresponding pump. Pump discharge pressure and flow indication demonstrate performance.

With the pumps shut down and the borated water storage tank outlet closed, the reactor building spray injection valves can each be opened and closed by operator action. With the reactor building spray inlet valves closed, low pressure air or smoke can be blown through the test connections of the reactor building spray nozzles to demonstrate that the flow paths are open.

The equipment, piping, valves, and instrumentation of the reactor building emergency cooling system are arranged so that they can be visually inspected. The cooling fans and coils and associated piping are located outside the secondary concrete shield. Personnel can enter the reactor building during power operations to inspect and maintain this equipment. The service water piping and valves outside the reactor building are inspectable at all times. Operational tests and inspections will be performed prior to initial startup.

Two service water pumps are normally operating. At least once per month operation of one pump is shifted to the third pump, so testing will be unnecessary.

As the reactor building fans are normally operating, starting for testing is unnecessary for those verified to be operating.

Reference

FSAR, Section 6

4.12 HYDROGEN RECOMBINERS SURVEILLANCE

~~Applicability~~

~~Applies to the surveillance of the hydrogen recombiner systems.~~ A1

~~Objective~~

~~To verify an acceptable level of efficiency and operability of the hydrogen recombiner systems.~~

SR 3.6.7.1 Specification

4.12.1 Each hydrogen recombiner system shall be demonstrated OPERABLE: L8

a. At least once per ~~6 months~~ 18 Months by verifying during a recombiner system functional test that the minimum heater sheath temperature increases to greater than or equal to 700°F within 90 minutes. Upon reaching 700°F, increase the power setting maximum power for 2 minutes and verify that the power meter reads greater than or equal to 60 KW. LA1 Bases

SR 3.6.7.2
SR 3.6.7.3

b. At least once per 18 months by:

1. ~~Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits,~~ LA2 TRM

2. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.), and AL3 LA1 Bases

3. Verifying the integrity of the heater electrical circuits by performing a resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms. LA1 Bases

<Later> (23D) 4.12.2 Hydrogen concentration instruments shall be calibrated once every 18 months with proper consideration to moisture effect. LATER

~~Bases~~

~~The OPERABILITY of the recombiners for the control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the~~ A2

~~expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water, and 3) corrosion of metals within containment. The hydrogen recombiner systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following LOCA", Rev. 2, November, 1978..~~ A2

4.26 REACTOR BUILDING PURGE VALVES

Applicability

This specification applies to the reactor building purge supply and exhaust isolation valves. A1

Objective

To assure reactor building integrity.

LA1 Bases

Specification

SR 3.6.3.1 4.26.1 The reactor building purge supply and exhaust isolation valves shall be determined closed at least once per 31 days when containment integrity is required by TS 3.6.1. M1 during MODES 1, 2, 3, and 4

<Later> (5.0) 3.6.1-4 3.6.3-3 3.6.3-4 3.6.3-13

4.26.2 Prior to exceeding conditions which require establishment of reactor building integrity per TS 3.6.1, the leak rate of the purge supply and exhaust isolation valves shall be verified to be within acceptable limits per TS 4.4.1, unless the test has been successfully completed within the last three months. LATER L24

Bases

Determination of reactor building purge valve closure will ensure that reactor building integrity is not unintentionally breached. A2

As a result of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration," it was concluded that excess leakage past valve resilient seals is typically caused by severe environmental conditions and/or wear due to use. Recommended leak test frequencies of three months are deemed to be adequate to detect seal degradation of resilient seals.

The three month test need not be conducted with the precision of the Type C 10CFR50, Appendix J criteria, however the test must be sufficient to detect degradation.

"R" - Relocation of requirements:

Relocating requirements which do not meet the Technical Specification selection criteria to documents with an established control program allows the Technical Specifications to be reserved only for those conditions or limitations upon reactor operation which are necessary to adequately limit the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety, thereby focusing the scope of Technical Specifications.

Therefore, requirements which do not meet the Technical Specification selection criteria in 10 CFR 50.36 have been relocated to other controlled license basis documents. This regulation addresses the scope and purpose of Technical Specifications. In doing so, it establishes a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in Technical Specifications. These criteria are as follows:

Criterion 1: Installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable that is an initial condition of a design basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier.

Criterion 4: A structure, system or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

The application of these criteria is provided in the "Application of Selection Criteria to the ANO-1 Technical Specifications." Requirements which met the criteria have been included in the proposed improved Technical Specifications. Entergy Operations proposes to remove the requirements which do not meet the criteria from the Technical Specifications and relocate the requirements to a suitable owner controlled document. The requirements in the relocated Specifications will not be affected by this Technical Specification change. Entergy Operations will initially continue to perform the required operation and maintenance to assure that the requirements are satisfied. Relocating specific requirements for systems or variables will have no impact on the system's operability or the variable's maintenance, as applicable.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

License basis document control mechanisms, such as 10 CFR 50.59, 10 CFR 50.54(a)(3), and ITS Section 5, "Administrative Controls," will be utilized for the relocated Specifications as they will be placed in other controlled license basis documents. This would allow Entergy Operations to make changes to these requirements, without NRC approval, as allowed by the applicable regulatory requirements. These controls are considered adequate for assuring structures, systems and components in the relocated Specifications are maintained operable and variables in the relocated Specifications are maintained within limits.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

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

   The proposed change relocates requirements and surveillances for structures, systems, components or variables which did not meet the criteria for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the ANO-1 Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled license basis document and maintained pursuant to the applicable regulatory requirements. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or change in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the affected requirement will be relocated to an owner controlled license basis document for which future changes will be evaluated pursuant to the requirements of the applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"A" - Administrative changes to requirements:

Reformatting and rewording the remaining requirements in accordance with the style of the improved Babcock & Wilcox Standard Technical Specifications in NUREG-1430 will make the Technical Specifications more readily understandable to plant operators and other users. Application of the format and style will also assure consistency is achieved between specifications. As a result, the reformatting and rewording of the Technical Specifications has been performed to make them more readily understandable by plant operators and other users. During this reformatting and rewording process, no technical changes (either actual or interpretational) to the Technical Specifications were made unless they were identified and justified.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

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

   The proposed change involves reformatting and rewording of the existing Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, there is no technical change to the requirements and therefore, there is no significant reduction in the margin of safety.

"LA" - Less restrictive, Administrative deletion of requirements:

Portions of some Specifications provide information that is descriptive in nature regarding the equipment, system(s), actions or surveillances. This information is proposed to be deleted from the specification and relocated to other license basis documents which are under licensee control. These documents include the TS Bases, Safety Analysis Report (SAR), Technical Requirements Manual, and Programs and Manuals identified in ITS Section 5, "Administrative Controls." The removal of descriptive information is permissible, because the documents containing the relocated information will be controlled through the applicable process provided by the regulatory requirements, e.g., 10 CFR 50.59, 10 CFR 50.54(a)(3), and ITS Section 5, "Administrative Controls." This will not impact the actual requirements but may provide some flexibility in how the requirement is conducted. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

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

   The proposed change relocates requirements from the Technical Specifications to other license basis documents which are under licensee control. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the Technical Specifications to other license basis documents, which are under licensee control, are the same as the existing Technical Specifications. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

"M" - More restrictive changes to requirements:

The ANO-1 Technical Specifications are proposed to be modified in some areas to impose more stringent requirements than previously identified. These more restrictive modifications are being imposed to be consistent with the improved Babcock & Wilcox Standard Technical Specifications. Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

The modification of the ANO-1 Technical Specifications and the changes made to achieve consistency within the specifications have been performed in a manner such that the most stringent requirements are imposed, except in cases which are individually evaluated.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

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

   The proposed change provides more stringent requirements for the ANO-1 Technical Specifications. These more stringent requirements are not assumed to be initiators of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes do not impact the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated for ANO-1.

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

The imposition of more stringent requirements prevents a reduction in the margin of plant safety by:

a) Increasing the analytical or safety limit,
b) Increasing the scope of the specification to include additional plant equipment,
c) Increasing the applicability of the specification,
d) Providing additional actions,
e) Decreasing restoration times,
f) Imposing new surveillances, or
g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

ANO-1 ITS SECTION 3.6 : Reactor Building Systems

"L" - Less Restrictive Changes to Requirements:

Entergy Operations has evaluated these proposed Technical Specification changes and has determined that they involve no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92(c) as indicated below:

3.6 L1 Not used.

3.6 L2 Not used.

3.6 L3

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

Availability of the reactor building cooling system and the reactor building spray system (with the sodium hydroxide system) is needed to mitigate several previously analyzed accidents, however, the systems are not accident initiators and therefore the change doesn't affect the probability of an accident. Elimination of the need to demonstrate the OPERABILITY of one redundant system train in order to perform maintenance on the other system train doesn't affect the consequences of an accident as the redundant trains are 100% capacity and performing maintenance on one doesn't impact the availability of the other.

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

The proposed change doesn't necessitate a physical alteration of the unit (no new or different kinds of equipment will be installed) or changes in parameters governing normal unit operation. The proposed change will still ensure the required reactor building cooling and spray system capacity is available when required or prompt and appropriate compensatory actions will be taken. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated

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

This change does not involve a significant reduction in a margin of safety because removing the need to demonstrate the OPERABILITY of an already OPERABLE, redundant system train prior to maintenance has no impact on the ability of the system to function in the maintenance configuration. Therefore, this change does not affect the margin of safety, especially for system trains which are also 100% capacity.

3.6 L4 Not used.

3.6 L5

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

The reactor building (RB) cooling system and the RB spray system (with the sodium hydroxide system) are needed to mitigate previously analyzed accidents which are assumed to start with the reactor critical at full power in MODE 1. Since the start conditions of an accident are not affected by whether the unit is taken to MODE 3 or MODE 5 in a shutdown, this change doesn't affect the probability of an accident. Furthermore, the Required Action to proceed to MODE 3 rather than MODE 5 does not significantly affect the consequences of an accident. Therefore this change of the Required Action does not significantly increase the consequences of any previously analyzed accident.

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

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed), a change to a previous analysis or changes in parameters governing normal unit operation. The proposed change will still ensure prompt and appropriate compensatory actions are taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

This change does not involve a significant reduction in a margin of safety since placing the unit in MODE 3 places the reactor in a sub-critical state. Additionally, during MODES 3 and 4, the reactor coolant is reduced to a temperature and pressure significantly below operating conditions at power and, therefore, no core damage is anticipated from a LOCA initiated during shutdown. Since a MODE 3 or 4 event is not analyzed in the ANO safety analysis, the reactor building OPERABILITY is maintained during MODES 3 and 4 to ensure that the offsite radiation exposure of 10CFR100 are not exceeded.

3.6 L6

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

The reactor building (RB) spray system and sodium hydroxide system are needed to mitigate previously analyzed accidents which are assumed to start with the reactor critical at full power in MODE 1. Since the start conditions of an accident are not affected by whether the unit is taken to MODE 3 or MODE 5 in a shutdown, this change doesn't affect the probability of an accident. Furthermore, the Required Action to proceed to MODE 3 rather than MODE 5 does not significantly affect the consequences of an accident. Therefore this change of the Required Action does not significantly increase the consequences of any previously analyzed accident.

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

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed), a change to a previous analysis or changes in parameters governing normal unit operation. The proposed change will still ensure prompt and appropriate compensatory actions are taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

This change does not involve a significant reduction in a margin of safety since placing the unit in MODE 3 places the reactor in a sub-critical state. Additionally, each of the trains of RB spray and sodium hydroxide (two independent trains) is 100% capacity and whether all or some portion of more than one train of their combined capacity is available, is non-significant to the margin-of-safety.

3.6 L7

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

The hydrogen recombiners are available in the reactor building to mitigate hydrogen buildup in the reactor building atmosphere after an accident. This equipment is not an accident initiator and no hardware changes are proposed, therefore, the change doesn't affect the probability of an accident. Changing the Applicability of this equipment to MODES 1 and 2 has little or no effect on the consequences of an accident previously evaluated since the Loss of Coolant Accident (LOCA) Design Basis Accident, which this equipment mitigates, assumes accident initiation during MODE 1.

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

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. The proposed change will still ensure prompt and appropriate compensatory actions are taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

This change does not involve a significant reduction in a margin of safety since the applicable MODES for the hydrogen recombiners include MODE 1 which is the starting mode for the LOCA analyses as well as the MODE from which the maximum hydrogen concentration could be produced.

3.6 L8

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

The hydrogen recombiners are not an accident initiator and no hardware changes are proposed, therefore, the extension of the system function test surveillance Frequency from 6 months to 18 months doesn't affect the probability of a previously analyzed accident. The surveillance Frequency has no impact on the availability expectations for this equipment, especially since each train is 100% capacity, and indicates little if any impact on the consequences of an accident previously evaluated.

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

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. The proposed change will still ensure the availability of the hydrogen recombiners for prompt and appropriate compensatory actions in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Extending the surveillance Frequency doesn't affect the actual availability of the equipment. Also, each train of the hydrogen recombiners has 100% of the required hydrogen concentration reduction capacity which provides significant redundancy and indicates that the margin of safety is not significantly reduced.

3.6 L9

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

The sodium hydroxide (NaOH) concentration is not an accident initiator and there are no hardware changes proposed, therefore the extension of the sampling Frequency and the elimination of a requirement for testing after each makeup doesn't affect the probability of an accident. There is a low probability of uncontrolled changes to the sodium hydroxide concentration since the tank is normally isolated from a makeup source and there is tank volume indication in the control room. These controls are adequate to maintain NaOH concentration at a level which will not cause an adverse impact and will not affect the consequences of an accident previously evaluated.

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

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. The proposed change will still provide adequate assurance of the availability of sodium hydroxide in sufficient concentration for appropriate compensatory actions in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

Extending the surveillance Frequency doesn't affect the actual concentration of the sodium hydroxide but rather the potential that the concentration might be altered without being noticed. However, the controls on the sodium hydroxide concentration are such that changes in concentration would be noticed and corrective action taken. Therefore, extension of the surveillance Frequency does not involve a significant reduction in a margin of safety.

3.6 L10

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

The reactor building spray headers and nozzles are not an accident initiator and there are no hardware changes proposed, therefore the extension of flow path testing from 5 years to 10 years doesn't affect the probability of an accident. Testing of the reactor building spray headers and nozzles for obstruction and function has indicated good performance and supports the conclusion that the availability of this passive component is not affected by changing the surveillance Frequency. The lack of impact on the ability of the spray headers and nozzles to function as intended indicates that there is no significant impact on the consequences of a previously evaluated accident.

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

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. The proposed change will still provide adequate assurance of the availability of the reactor building spray headers and nozzles for appropriate compensatory actions in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

This change does not involve a significant reduction in a margin of safety since extending the surveillance Frequency will introduce only a small increase in the potential that a nozzle or header could be obstructed or degraded between surveillances. Previous testing indicates good availability performance of the reactor building spray headers and nozzles and supports the conclusion that there is no significant reduction in a margin of safety. Furthermore, it should be recognized that each train of the RB spray system is independent and redundant. This level of system redundancy also supports the conclusion that there is no significant reduction in a margin of safety from the extension of surveillance Frequency.

3.6 L11

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

The reactor building (RB) cooling train service water flow rate test frequency is not an accident initiator and there are no hardware changes proposed, therefore the extension of the flow rate test Frequency doesn't affect the probability of an accident. Whether the consequences of a previously analyzed accident are affected is dependent on the availability of adequate service water flow to the RB coolers. The surveillance Frequency of 31 days is consistent with the test Frequency of other components in the RB cooling system and is appropriate for a system which is considered reliable. The reliability of adequate service water flow to the RB coolers indicates that extending the service water flow test Frequency from 14 days to 31 days will not affect the consequences of an accident previously evaluated.

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

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. The proposed change will still provide adequate assurance of the availability of adequate reactor building (RB) cooling system service water flow for appropriate compensatory actions in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

This change does not involve a significant reduction in a margin of safety since extending the surveillance Frequency will introduce only a small increase in the potential that the reactor building (RB) cooling system service water flow could be degraded between surveillances. Previous testing indicates good availability performance of the RB cooling system service water flow. Furthermore, it should be recognized that each train of the RB cooling system, including the service water supply, is independent and 100% capacity. This level of system redundancy also supports the conclusion that there is no significant reduction in a margin of safety from the extension of surveillance Frequency.

3.6 L12

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

The adoption of an alternate method for position verification of valves in high radiation areas is not an accident initiator and there are no hardware changes proposed. Therefore, the introduction of an option to validate valve position by administrative means does not affect the probability of an accident. The consequences of an accident previously evaluated could only be affected if the valves of interest are out of proper position. The administrative control of valves in high radiation areas is considered adequate to maintain the proper position of this valve population. This conclusion is based on the limited accessibility of valves in this population, the strength of the ALARA program and the effectiveness of other administrative control programs. The adoption of an alternate method for position verification of valves in high radiation areas is not considered to cause a significant increase in the consequences of an accident previously evaluated.

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

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. The proposed change will still provide adequate assurance of the proper positioning of valves to provide compensatory actions in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

This change does not involve a significant reduction in a margin of safety since just altering the method of verifying correct position for the limited set of valves located in a high radiation area has a non-significant impact on the OPERABILITY of the valves in question.

3.6 L13

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

The hydrogen recombiners are available in the reactor building to mitigate hydrogen buildup in the reactor building atmosphere after an accident. This equipment is not an accident initiator and no hardware changes are proposed, therefore, the change doesn't affect the probability of an accident. Adopting a Note which makes LCO 3.0.4 not applicable while one of the hydrogen recombiners is inoperable, has no impact on the consequences of a previously evaluated accident. This conclusion is based on the availability of the other, 100% capacity, hydrogen recombiner which would be available to mitigate an accident.

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

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

This change does not involve a significant reduction in a margin of safety since each hydrogen recombiner is 100% capacity and the probability of a LOCA occurring while one of the hydrogen recombiners is inoperable is very small.

3.6 L14

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

Sodium hydroxide (NaOH) is added to the reactor building (RB) spray solution for pH control. There are two independent flow paths, including manual and control valves, which provide gravity flow of the NaOH solution to the RB spray headers. This equipment is not an accident initiator and no hardware changes are proposed, therefore, the change doesn't affect the probability of an accident. Deleting a requirement that the manual valves in the sodium hydroxide system be locked open has no impact on the consequences of a previously evaluated accident. Locking these valves open has small impact on the availability of the NaOH system which is virtually eliminated by the Frequency of valve position surveillance which is adopted.

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

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

This change does not involve a significant reduction in a margin of safety since deleting the requirement to lock open the sodium hydroxide manual valves has minimal impact on the availability of the sodium hydroxide system. Furthermore, valve position surveillance, which is maintained, for the sodium hydroxide system is considered sufficient to provide system availability in the event of an accident.

3.6 L15

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

The restoration time for an inoperable reactor building spray train or sodium hydroxide system is not an accident initiator and, since no hardware changes are proposed, the change doesn't affect the probability of an accident. The extension of the restoration time is a non-significant change to the consequences to previously evaluated Design Basis Accidents. This conclusion is based on the small probability that an accident will occur during the extended restoration time.

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

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

This change does not involve a significant reduction in a margin of safety since the extension of the restoration time has minimal impact on the availability of individual trains of the reactor building (RB) spray system or the sodium hydroxide system. Furthermore, the change to the margin of safety is not impacted since the other trains, which remain OPERABLE, are each 100% capacity.

3.6 L16

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

This change will delete the requirement to test the redundant, OPERABLE, valve when the first valve is declared inoperable in a two valve reactor building (RB) penetration. Furthermore, the change will relax the limitation the isolation must occur using only the Operable valve; allowing the isolation to be made with a closed and deactivated automatic valve, a closed manual valve, a blind flange, or a check valve with the flow secured. The testing of RB penetrations is not an accident initiator and there are no penetration hardware changes proposed, therefore, the change doesn't affect the probability of an accident. Deleting a requirement that the second valve be tested has no impact on the consequences of a previously evaluated accident. This is because testing the "other" valve has no impact on whether the reactor building can be isolated by the closing of RB isolation valves. The allowed methods of isolating the penetration are only those already included in the plant design and therefore, there will be no impact on the consequences of a previously evaluated accident.

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

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions (the isolation valves will be closed) will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

This change does not involve a significant reduction in a margin of safety since deleting the requirement to test the redundant, OPERABLE, valve when one valve in a two valve reactor building (RB) penetration is declared inoperable, has minimal impact on the availability of the penetration to provide reactor building isolation. Isolating the penetration provides assurance that penetration function is maintained in compliance with design and hence margin of safety is not significantly reduced.

3.6 L17

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

This change will extend the Completion Time for taking the unit to MODE 5 upon failure to restore an inoperable reactor building (RB) isolation valve or isolate the RB penetration flow path. This change and equipment is not an accident initiator and since no hardware changes are proposed, the change doesn't affect the probability of an previously evaluated accident. The extension of the Completion Time is consistent with the time allowed in NUREG-1430 and is furthermore a small extension of the transition period to MODE 5. This change is a non-significant impact on the consequences of an accident since the accident parameters are not affected, only a limited extension of the time the unit would be in MODE 1, 2, 3 or 4.

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

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

This change does not involve a significant reduction in a margin of safety since no accident parameters are changed.

3.6 L18

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

This change will delete the set of "power operated" valves, which do not receive a reactor building isolation signal, from the set of "automatic" valves, which do receive a reactor building isolation signal, for the purpose of valve closure time testing. This equipment is not an accident initiator and no hardware changes are proposed, therefore, the change doesn't affect the probability of an accident. The consequences of a previously evaluated accident are not impacted since the valves included in the accident analysis are part of the set of automatic valves and there is , therefore, no change to the analysis assumptions and conclusions.

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

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

This change does not involve a significant reduction in a margin of safety since there is no change to the accident scenario; only a change to the surveillance requirements of "power operated" valves which do not receive a reactor building isolation signal.

3.6 L19

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

This change will introduce the option to lock, seal, or otherwise secure the engineered safeguards (ES) valves for the reactor building (RB) spray system and the RB cooling system when RB OPERABILITY is required. Before this change, the only option was to lock the valves in position. The method of verifying ES valve position is not an accident initiator and no hardware changes are proposed, therefore, the change doesn't affect the probability of an accident. Expanding the methods available for verifying ES valve position has no impact on the consequences of a previously evaluated accident since the valves of interest are still placed in proper position for their safety function.

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

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

This change does not involve a significant reduction in a margin of safety since expanding the methods of verifying the RB spray system and RB cooling system ES valves has minimal impact on the availability of the systems and the valves. Furthermore, valve position surveillance, regardless of method of verification, is considered sufficient to provide system availability in the event of an accident.

3.6 L20

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

This change will introduce the option to close but leave unlocked, seal, or otherwise secure the inside and outside reactor building manual isolation valves when reactor building OPERABILITY is required. Before this change, the only option was to close and lock the valves. Furthermore, if locked, sealed or otherwise secured, the required position verification is allowed to be administratively controlled outside of Technical Specifications. The method and frequency of verifying the inside and outside reactor building manual isolation valve position is not an accident initiator and no hardware changes are proposed; therefore, the change doesn't affect the probability of an accident. Removing the requirement for the administrative control of locking, sealing, or otherwise securing the inside and outside reactor building manual isolation valves in position has no impact on the consequences of a previously evaluated accident since the valves of interest are still placed in proper position for their safety function.

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

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

This change does not involve a significant reduction in a margin of safety since removing the requirement for the administrative control of locking, sealing, or otherwise securing the inside and outside reactor building manual isolation valves in position has minimal impact on the availability of the systems. Furthermore, the removing the re-verification of the position of valves that have been locked, sealed, or otherwise secured (they were verified to be in position when locked, sealed, or otherwise secured) is does not involve a significant reduction in a margin of safety. The original verification and concurrent administrative controls is considered sufficient to provide assurance of reactor building OPERABILITY in the event of an accident.

3.6.2-1

3.6.2-4

3.6 L21

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

The proposed changes will allow entry and exit to perform repairs on an inoperable airlock door, separate Condition entry for each airlock, and allow entry and exit for 7 days under administrative controls if both air locks are inoperable. The personnel and escape air locks and their components are not considered to be accident initiators and no hardware changes are proposed; therefore, the change does not affect the probability of an accident. Entry and exit through an airlock is an evolution that requires a very short period of time as the door may be open for a few minutes, at most. Due to the low probability of an accident requiring reactor building isolation during this short period of time, this change does not involve a significant increase in the consequences of an accident. Allowing a separate Condition entry for each airlock does not involve a significant increase in the consequences of an accident since appropriate compensatory measures are contained in the proposed ITS requirements.

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

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

This change does not involve a significant reduction in a margin of safety since the proposed ITS also include prompt and appropriate compensatory measures to ensure an acceptable level of safety with regard to reactor building integrity.

3.6.3-8

3.6 L22

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

The proposed changes will allow 72 hours to isolate the affected penetration flow path in the event one reactor building isolation valve is inoperable in a penetration servicing a closed system. General Design Criteria (GDC) 57 requires that each line of a closed system that penetrates containment shall have at least one containment isolation valve which shall either be automatic, or locked closed, or capable of remote manual operation. One isolation valve is sufficient, since the closed system is an isolation barrier. Since reactor building isolation valves are not considered to be accident initiators, this change does not involve a significant increase in the probability of an accident. Since the proposed ITS also include prompt and appropriate compensatory measures to ensure an acceptable level of reactor building integrity is maintained, this change does not involve a significant increase in the consequences of an accident.

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

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Prompt and appropriate compensatory actions will still be taken in the event of an accident. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

This change does not involve a significant reduction in a margin of safety since the proposed ITS also include prompt and appropriate compensatory measures to ensure an acceptable level of safety with regard to reactor building integrity.

3.6.5-8

3.6 L23

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

The proposed changes will allow an actual actuation signal to be used in lieu of a simulated test signal to demonstrate actuation of the reactor building spray system and the reactor building emergency cooling system. Since the proposed change does not require the use of an actual actuation of the system to satisfy the testing requirement, but only allows crediting an actual actuation, this change does not involve a significant increase in the probability of an accident. Since no new testing configuration is proposed, this change does not involve a significant increase in the consequences of an accident.

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

The proposed change does not necessitate a physical alteration of the unit (no new or different type of equipment will be installed) or changes in parameters governing normal unit operation. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

This change does not involve a significant reduction in a margin of safety since the proposed ITS only allows the unit to take credit for an actual actuation signal in lieu of a simulated signal. This could result in reducing the number of system actuations that would occur during a test interval if an actual actuation occurs. Therefore, this change does not result in a significant reduction in a margin of safety.

3.6.1-4

3.6.3-3

3.6.3-4

3.6.3-13

3.6 L24

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

The proposed change relocates the frequency for the performance of the reactor building purge isolation valve leak rate test to the Reactor Building Leakage Rate Testing Program. The existing frequency is retained in the proposed program. No new accidents or increased dose consequences have been introduced by relocating this surveillance requirement.

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

The proposed change relocates the frequency for the performance of the reactor building purge isolation valve leak rate test to the Reactor Building Leakage Rate Testing Program. No new testing requirements or methods of testing have been added. Therefore, no new changes have been created that would introduce a possibility of a new or different kind of accident than any accident previously evaluated.

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

The proposed change relocates the frequency for the performance of the reactor building purge isolation valve leak rate test to the Reactor Building Leakage Rate Testing Program.. The same surveillance frequency has been retained in the proposed program. NRC approval would still be required before a change in the test frequency could be made. Therefore, relocating these requirements does not reduce the margin of safety.

1. NUREG Section 3.6 - The title of this section is changed from "Containment" to "Reactor Building" Systems to comply with ANO-1 terminology in the license basis documents. References to "containment" in the NUREG-1430 text are changed to "reactor building," "the reactor building," or the abbreviation "RB" as appropriate for the ITS context. However, marking up the NUREG pages to show these changes introduces significant clutter to the page with little value for the purpose of the markup. Therefore, only one reference to this DOD item will be placed on each page of the NUREG/ITS markup for this section at the first occurrence (usually in the section title area) with subsequent changes on that page not marked or annotated with this DOD number to conserve margin space.

   In the NUREG-1430 3.6 Bases, use of the term "containment" is very frequent and showing the change would affect readability and reduce margin space. In order not to clutter the markup with material of negligible value, only the first occurrence of the change (usually in the section title area) will be shown in the ITS Bases markup and annotated once on each page; therefore, the word "containment" should be read as "reactor building" (or "the reactor building") wherever it occurs in the Bases text. However, since a generic rule usually doesn't apply to every case, some changes will still be marked in the Bases text as appropriate for sentence clarification, unit specific changes, or to provide emphasis. These changes are consistent with current license basis.

3.6.1-3 2. NUREG SR 3.6.1.1, SR 3.6.2.1 – The ITS is revised to reflect ANO-1 CTS 4.4.1 requirements and TSTF-52, Rev. 3. These surveillance requirements indicate that the reactor building leakage rate testing is to be performed in accordance with the Reactor Building Leakage Rate Testing Program described in CTS 6.8.4. These CTS requirements were established by Amendment 185 which implemented 10 CFR 50, Appendix J, Option B. In the ITS, the Reactor Building Leakage Rate Testing Program will be covered by ITS Specification 5.5.6. The Bases were revised as appropriate. These changes are consistent with current license basis.

   The NUREG 3.6.1, 3.6.2, and 3.6.3 Bases have also been revised to incorporate TSTF-52, Rev.3.

3. NUREG 3.6.2 Bases, Background, - 3rd paragraph is deleted since it does not reflect the ANO-1 design.

4. NUREG SR 3.6.3.3, SR 3.6.3.4 - Incorporated TSTF-45, Rev. 2.

5. NUREG SR 3.6.3.5 (ITS SR 3.6.3.4) – Incorporated TSTF-46, Rev. 1.

6. NUREG 3.6.3 Required Action C.1 Completion Time – Incorporates TSTF-30, Rev. 3.

   3.6.3-9 NUREG 3.6.3 Bases - The TSTF-30, Rev. 3, reference to Standard Review Plan 6.2.4 in the NUREG 3.6.3 Condition C Bases and the subsequent reference number changes are not adopted as the Standard Review Plan is not consistent with the ANO-1 license basis. In

place of this reference to the SRP 6.2.4 requirements, ANO-1 has provided a discussion of the design requirements associated with the only closed system within the reactor building to which the actions of 3.6.3 Condition C would apply.

7. The ANO-1 current license basis (CTS 3.23) requires that the reactor building (RB) purge valves be closed and not operated whenever RB integrity is required (that is, above cold shutdown). The current license basis for the RB purge valves will be maintained in the ITS.

   NUREG 3.6.3 ACTIONS Note 1 was modified and NUREG SR 3.6.3.2 was deleted since the ANO-1 RB purge system design features one suction and one discharge train with 24 inch valves in series (one inside and one outside the reactor building) and has no smaller-diameter bypass purge valves or alternate purge flow paths.

3.6.3-12

   NUREG 3.6.3 ACTIONS A, B, and D - In accordance with ANO-1 current license basis, the ANO-1 RB purge isolation valves are closed and not operated whenever RB integrity is required. As a result of this requirement, the RB purge valves in the CTS are considered to be "otherwise secured" for the duration of the operating cycle and therefore do not receive special leakage monitoring above cold shutdown. The CTS requirements for the ANO-1 RB purge isolation valves require closing these valves and removing the handswitch key when RB integrity is required and furthermore provides that operation of the purge valves when RB integrity is required is specifically prohibited by CTS 3.23. NUREG 3.6.3, in Conditions A, B, and D, addresses purge valves uniquely with respect to other RB isolation valves assuming that these valves may be operated during periods when the RB must be OPERABLE. Since these valves at ANO-1 are not operated when the RB must be OPERABLE, then their OPERABLE condition for the ITS will be the same as in CTS 3.23.1 which is "closed with the handswitch key removed" and therefore, no different than other RB isolation valves. These CTS requirements will be adopted in the ITS by deleting the purge valve references from NUREG 3.6.3 Condition A and Condition B, by deleting NUREG 3.6.3 Condition D in its entirety, and by deleting the "sealed" term which has no definition for these valves in the CTS.

3.6.3-14

   NUREG 3.6.3 Bases Applicable Safety Analyses discussions are revised to reflect ANO-1 specifics. During operation in MODES 1, 2, 3, and 4 the purge valves are maintained closed with the handswitch keys removed, as discussed above. Removing the handswitch keys prevents inadvertent operation of the purge valves, but does not prevent spurious operation since the control circuit remains energized in this configuration. The Applicable Safety Analysis discussion has been revised to delete reference to spurious actuation and discuss inadvertent actuation instead. This is consistent with the current license basis.

   Since NUREG 3.6.3 Condition D was deleted, Condition E was renumbered to Condition D for adoption by the ITS.

   NUREG SR 3.6.3.1 is revised for adoption in the ITS to reflect the ANO-1 purge isolation valve configuration and operation in a manner consistent with CTS 4.26.1. NUREG-1430

ASA Bases, SR 3.6.3.2, and SR 3.6.3.8 are not adopted by the ITS since these surveillances address a purge system configuration which is not applicable to ANO-1. NUREG SR 3.6.3.6 is not adopted by the ITS since leakage rate testing for the purge valves at ANO-1 is no different than that done for other reactor building isolation valves that are closed during MODES 1, 2, 3, and 4. The leakage rate testing for the purge valves is done as part of the Reactor Building Leakage Rate Testing Program (RBLRTP) and is addressed in ITS Section 5.0. Since these NUREG SRs are not adopted, the other NUREG SRs are renumbered as follows: SR 3.6.3.3 to ITS SR 3.6.3.2, SR 3.6.3.4 to ITS SR 3.6.3.3, SR 3.6.3.5 to
ITS SR 3.6.3.4, and SR 3.6.3.7 to ITS SR 3.6.3.5.

The NUREG-1430 3.6.3 Bases were modified as appropriate for ANO-1 specific purge valve changes.

8. NUREG SR 3.6.2.2 – Incorporates TSTF-17, Rev. 2, as modified for the ANO-1 fuel cycle length. The present outage cycle for ANO-1 is approximately 18 months, therefore the Frequency is changed from 184 days to 18 months instead of 24 months as shown in TSTF-17, Rev. 2.

9. The NUREG-1430 title for NUREG LCO 3.6.7, the "Spray Additive System", is adopted for ITS LCO 3.6.6. Adopting this name for the system is considered to emphasize its post-LOCA radioactive iodine-removal function and is consistent with the design intent. However, the CTS terminology for the components of the system is retained in the ITS. Therefore, a specific component reference, for instance in a Surveillance Requirement, to the "sodium hydroxide tank" or "NaOH tank" will not become the "spray additive tank". This change is consistent with NUREG-1430 as well as with the CTS. The ITS 3.6.6 Bases are revised as appropriate for component names.

10. NUREG 3.6.8 Condition B - ANO-1 has no alternate hydrogen control system for use when two hydrogen recombiners are inoperable; therefore, NUREG 3.6.8 Condition B is not applicable and is not adopted in the ITS. The subsequent NUREG 3.6.8 Condition C is renumbered to Condition B for adoption by the ITS. The discussion of when this deleted NUREG Condition could be adopted is retained in the ITS Bases. This change is consistent with current license basis.

11. NUREG 3.6.2 Bases Background - ANO-1 terminology differentiates between the two air locks. The two ANO-1 air locks are similar in function but different in size and mechanism. The NUREG 3.6.2 Bases are revised where appropriate for the different air locks. Only the initial change is annotated for this item on a page in order to save margin space. The remaining changes on a page are marked up but not annotated. This change is consistent with current license basis.

12. NUREG 3.6.5 is not adopted in the ITS since there are no comparable requirements in the ANO-1 CTS for reactor building (RB) air temperature. Administrative monitoring of the ANO-1 RB temperature indicates that the actual RB bulk air temperature is maintained at a

conservative value with respect to the design basis bulk air temperature. ANO-1 has completed modifications which address RB bulk average temperature concerns that were recognized in the 1987 timeframe (1CAN088707). The current condition and operating experience for ANO-1 is that there is a significant margin between the design basis accident analysis initial temperature (140°F) and the maximum observed RB bulk air temperature (less than 120°F). This margin, furthermore, is not sustained by additional active components but rather by passive design changes such as improving the insulation on the RCS components and improvements to the existing RB air distribution systems. Operating experience for ANO-1 indicates that maintenance of a conservative RB temperature is a robust process with an adequate administrative monitoring system in place. This change is consistent with current license basis.

13. NUREG SR 3.6.7.5 is not adopted in the ITS. The ANO-1 Spray Additive System design features gravity flow of the NaOH solution from the NaOH tank to the Borated Water Storage Tank (BWST) headers upstream of the suction of the reactor building (RB) spray pumps. There are no means of directly measuring flow-before-run-out or rate-of-flow of the NaOH solution for the purposes of surveillance. Assurance of introduction of an adequate amount of NaOH during the design basis accident (DBA) was established analytically based on the passive features of the system including BWST and NaOH tank levels and time to run-out, flow characteristics of the associated pumps and valves, and the associated piping configurations. The availability of the active components of the Spray Additive System is verified by other ITS Surveillance Requirements. This change is consistent with current license basis.

14. NUREG 3.6.4 Bases - ITS 3.6.4 Bases discussion for reactor building (RB) pressure is revised to address inclusion of the Emergency Core Cooling System analysis as a contributor to the limits on RB pressure. This Bases discussion is also revised to address the fact that the LOCA pressure margin to design pressure is a feature of the analysis results rather than the analysis input and that the tornado loads have no effect on the reactor building pressure limit values. This change is consistent with current license basis.

15. NUREG 3.6.6 Bases - The ITS 3.6.5 Applicable Safety Analysis Bases discussion of cooling system total delay time was revised to reflect ANO-1 unit specific design and analysis results. The ANO-1 LOCA analysis conservatively assumes that the RB spray and cooling systems do not affect the early stages of the event and that they are capable of functioning in a degraded condition when they are used. The ITS 3.6.5 LCO Bases discussion of the Reactor Building (RB) Spray System and the RB Cooling System was revised to include unit specific details of the ANO-1 configuration. The ITS 3.6.5 Bases, in general, are revised to describe the significant differences of the ANO-1 RB cooling system configuration from that depicted in NUREG-1430. These changes are consistent with current license basis.

16. NUREG 3.6.7 Bases – ITS 3.6.6 Bases for the Spray Additive System were revised to provide specific design details of the Spray Additive System and components. These details include: gravity flow of sodium hydroxide and mixing, absence of capability to

directly measure sodium hydroxide solution flow and details of the sodium hydroxide tank level indication. These changes are consistent with the current license basis.

17. NUREG 3.6.8 Bases - The ITS 3.6.7 Bases for the Hydrogen Recombiners were revised to describe the ANO-1 installed configuration. The ANO-1 design features two, independent, 100% capacity hydrogen recombiners installed inside the reactor building. The design also is based on a 4 v/o (volume percent) minimum flammability concentration for hydrogen which is more conservative than 4.1 v/o. These changes are consistent with current license basis.

18. NUREG 3.6.3 - Incorporated TSTF-269, Rev 2.

3.6.3-7 19. Not used.

3.6.4-1 20. Not used.

21. NUREG 3.6.6 is renumbered to ITS 3.6.5 because NUREG 3.6.5, Containment Temperature, was not adopted [Ref. DOD 12]. ITS 3.6.5 is modified by the addition of a Note and revision of the Conditions and Required Actions. The NUREG is modified to retain the CTS 3.3.1 and 3.3.4 requirements for two trains of reactor building (RB) spray and two trains of RB cooling OPERABLE during MODES 1 and 2, and one train of RB spray and one train of RB cooling OPERABLE during MODES 3 and 4. During MODES 3 and 4, the potential energy release of the high energy systems inside the reactor building is significantly less than during MODES 1 and 2. Requiring the availability of one train of RB spray and one train of RB cooling during MODES 3 and 4 is considered adequate given the reasons above and the limited time spent in these MODES. These changes are consistent with current license basis.

    ITS SR 3.6.5.1, SR 3.6.5.4, SR 3.6.5.5, SR 3.6.5.6, and SR 3.6.5.8 were modified to incorporate the requirements for required trains of reactor building spray during MODES 3 and 4. These changes are consistent with current license basis.

    The NUREG 3.6.7 Applicability is modified to MODES 1 and 2 for ITS 3.6.6 in order to retain the CTS 3.3.4 requirements for availability of sodium hydroxide. NUREG 3.6.7 Condition B is modified to retain the CTS requirements and to make ITS 3.6.6 Condition B compatible with ITS 3.6.5 Condition D. These changes are consistent with current license basis.

    The NUREG 3.6.7 Bases and LCO 3.6.6 Bases are both marked up to reflect these changes. These changes are consistent with current license basis.

22. NUREG 3.6.1 Bases – The Applicable Safety Analyses (ASA) discussion includes a statement that satisfactory leakage rate test results are a requirement of containment OPERABILITY. This statement is also paraphrased and included, more appropriately, in the discussion for the NUREG 3.6.1 LCO. The ITS will delete the redundant sentence

from the ASA and retain the appropriate statement in the LCO discussion. This change constitutes an editorial preference.

23. A maximum volume for the sodium hydroxide tank was deleted from NUREG SR 3.6.7.2 (renumbered in the ITS as SR 3.6.6.2). In CTS 3.3.4(B) only the minimum NaOH tank volume is required. The ANO-1 calculation assumption, upon which a maximum tank volume would be based, exceeds the physical volume of the NaOH tank and is therefore moot. This change complies with the CTS.

3.6.2-4 24. Not used.

25. NUREG 3.6.6 Bases - The ITS 3.6.5 Bases for ACTIONS A.1 and B.1 are revised to emphasize that the remaining OPERABLE reactor building (RB) spray train is redundant for reactor building cooling and for support of the Spray Additive System function of iodine removal. In the Conditions associated with these ACTIONS, the OPERABLE RB cooling trains are also redundant for cooling but not iodine removal. This change is consistent with current license basis.

26. NUREG 3.6.6 Bases - The ITS SR 3.6.5.2 Bases are revised to reflect the current license
3.6.5-9 basis of the unit. The installed configuration of the reactor building (RB) cooling trains is not able to indicate the status of individual train components for vibration or blockage, etc. A discussion of how the SR is performed has been added. This change is consistent with current license basis.

27. NUREG 3.6.6 Bases - The ITS 3.6.5 Bases Background discussion was revised to identify the version of the General Design Criteria used during the design ANO Unit 1. The bulk of ANO-1 design, including the general design criteria, occurred prior to the 1971 inception of 10 CFR 50, Appendix A, and the titles and criteria statements in SAR Section 1.4 are slightly different. This change is consistent with current license basis.

28. 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).

29. Not used.

30. NUREG 3.6.3 Bases - The NUREG 3.6.3 Bases Background, Applicable Safety Analysis, and SR 3.6.3.5 Bases discussions are revised for the ITS to address that the ANO-1 analyses for reactor building isolation set no specific values for reactor building isolation valve response times. Consideration of the response of the reactor building isolation valves is satisfactorily addressed by assuring valve response times are maintained in accordance with the industry standards for sizing valve operators. This is consistent with the current ANO-1 licensing basis and supports the objective to minimize the potential leakage paths to the environment.

31. NUREG 3.6.6 Bases - The ITS SR 3.6.5.3 Bases discussion is revised to emphasize the relationship of the 1200 gpm service water flow to the original design intent for the reactor building coolers. This change is consistent with current license basis.

32. NUREG 3.6.2 Bases - The ITS 3.6.2 Background discussion deletes this sentence since it discusses airlock operation in MODES 5 and 6. Guidance is provided in ITS 3.9, "Refueling Operations."

33. NUREG 3.6.6 Bases - The reference to "relatively cold" borated water is deleted from the ITS 3.6.5 Bases Background discussion. A subjective value for the fluid temperature is not pertinent to this Bases discussion.

34. NUREG 3.6 Bases - Bases discussions are revised to detail attributes of the ANO-1 design bases. The ANO-1 Design Basis Accidents consider that Power Operation (ITS MODE 1) conditions are the initial conditions.

35. NUREG 3.6.3 ACTIONS Note 1 has an inclusive intent (penetration flow paths) and an exclusive intent (purge valve flow paths). In the CTS, the option to open penetration flow paths under administrative control is provided by the "as required" allowance in definition 1.7 item c. However, the exclusion of the purge valve path is provided by CTS 3.23.1 which states: "shall be closed ... whenever containment integrity is required ...".

36. NUREG SR 3.6.1.2 and its associated Bases are not incorporated in the proposed ITS. Amendment 199, September 9, 1999, that revised the current reactor building structural integrity requirements did not include a similar SR since reactor building structural integrity is verified in accordance with Subsection IWL of Section XI of the ASME Boiler and Pressure Vessel Code, incorporated by reference in 10CFR50.55a. Additional statements have been added to the LCO discussion in the Bases of ITS 3.6.1 to provide assurance that structural integrity is also a measure of reactor building operability. This change is consistent with the current license basis.

37. NUREG 3.6.4 Background and NUREG 3.6.6 ASA are modified to remove discussions of inadvertent containment spray issues. The functional and design requirements related to structural integrity during an inadvertent containment spray are not related to the

10 CFR 50.36 criteria for Technical Specification process parameters. The values retained (and discussed in the Bases) relate the DBA and ECCS analyses.

38. NUREG 3.6.3 Required Action A.1 Completion Time (and its Bases) is revised. The time allowed to isolate a penetration with one inoperable isolation valve and one operable valve is revised to 48 hours to reflect the current licensing basis in the CTS.

39. NUREG 3.6.4 statement in the Bases has been provided to explicitly establish that the safety analysis parameters presented in the ITS do not contain allowances for instrumentation error. This change is considered to be administrative in nature.

3.6.3-17 40. NUREG 3.6.3 LCO Bases discussing normally closed isolation valves is revised to correct erroneous descriptions. The LCO Bases are intended to provide a few details regarding OPERABILITY. These details are not all-encompassing, but only serve to highlight the important features of OPERABILITY. The paragraph discussing normally closed valves attempts to define OPERABILITY, but makes statements that are not true in all cases. The current wording could lead one to believe that manual valves are only operable when in the closed position. This would present a conflict when a manual isolation valve is open under administrative controls, a condition clearly allowed by the specification. In addition, the discussion results in two OPERABILITY requirements for check valves. Check valves are classified as automatic components, however, this paragraph states that a normally closed check valve must have flow through the valve to be OPERABLE. Similar wording has been approved for the FERMI ITS conversion. One difference between the FERMI wording and that proposed for ANO-1 is FERMI also incorporated a sentence on normally closed automatic isolation valve OPERABILITY. This statement is not required because the first sentence in the second paragraph of the LCO discussion requires the same OPERABILITY requirements for all automatic power operated isolation valves. This does not modify the requirements or the interpretation of the NUREG 3.6.3 LCO requirements.

41. NUREG 3.6.3 Actions Bases have been revised to replace "dedicated operator" with "dedicated individual" and to include a statement that the administrative controls allowing
3.6.3-20 penetration flow paths to be unisolated may allow the dedicated individual to be stationed in a nearby area vice "at the valve controls" due to ALARA considerations. Use of the term "operator" could be interpreted to limit the individual assigned to isolate the penetration to someone within the Operations Department. In some cases, this function could be performed by someone from another department who is qualified to operate the valve, such as a chemist. This change is acceptable since the administrative controls will still require a dedicated individual be present to quickly isolate the penetration when required.

A literal reading of the NUREG Bases could result in a dedicated individual being stationed literally at the valve controls in a higher dose area. This change would result in reducing the dedicated individual radiation dose, while providing assurance that the intent of being able to rapidly close the penetration is retained.

42. NUREG 3.6.2 Background Bases contains a description of the air locks which includes a specific diameter for the air lock. This discussion has been revised in the ITS to remove the specific diameter. The ANO-1 SAR, Section 5.2.2.1.3, describes the equipment and personnel hatches and does not contain this level of detail. The CTS also does not contain this level of detail. Since the removal of this information will have no impact on the operator's understanding of the specification, this deletion is acceptable.

3.6.2-5

43. NUREG 3.6.1 Applicable Safety Analyses Bases and NUREG 3.6.2 Applicable Safety Analyses Bases have been revised to delete the rod ejection accident from the list of DBAs that challenge reactor building OPERABILITY. The rate of mass and energy input to the reactor building due to a postulated pressure housing failure associated with a rod ejection is considerably lower than that subsequently reported for the smallest rupture size considered in the loss of coolant accident. The maximum hole size resulting from a rod ejection is approximately 2.76 inches. This lower rate of energy input results in a much lower reactor building pressure than those obtained for the limiting rupture sizes considered in the Loss of Coolant Accident (LOCA). See ANO-1 SAR Section 14.2.2.4.6

ANO-72

44. NUREG 3.6.3 LCO Bases has been revised to show that the automatic power operated valve listing is contained in the ANO-1 SAR, while the associated valve stroke times are contained in the Inservice Testing Program. The ANO-1 SAR does not contain the associated stoke times for these valves.

3.6.3-15

45. NUREG SR 3.6.6.4 Bases (ITS SR 3.6.5.4 Bases)have been revised to incorporate a discussion of the acceptance criteria for the SR. This change incorporates details of the SR from the current license basis.

3.6.5-3

46. NUREG SR 3.6.6.5 and SR 3.6.6.6 (ITS SR 3.6.5.5 and SR 3.6.5.6) Bases have been revised to incorporate a discussion of the testing methodology used to test the reactor building spray isolation valves. Since these valves must be closed and deenergized during the required testing of the spray pump in these SRs to prevent spraying the reactor building, the existing wording could result in confusion. The existing wording states that valves that are locked, sealed, or otherwise secured in position under administrative controls are not required to be tested. This could be interpreted such that testing of the reactor building spray isolation valves is not required since they are closed with breakers open. The inserted discussion describes how the testing is accomplished to ensure that the valves are appropriately tested. This change is consistent with ANO-1 current practices, as allowed by the current license basis.

3.6.5-7

ANO-73

These Bases have also been revised to incorporate a discussion of the acceptable performance of the SRs that was previously contained in CTS 4.5.1.1c. This provides additional details on the performance of the SRs.

CTS

Reactor Building
Containment
3.6.1

1

REACTOR BUILDING

3.6 ~~CONTAINMENT~~ SYSTEMS
Reactor Building
3.6.1 ~~Containment~~
The reactor building
LCO 3.6.1 ~~Containment~~ shall be OPERABLE.

1.7 a & e
3.6.1

3.6.1

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. ~~Containment~~ Reactor building inoperable.	A.1 Restore ~~containment~~ reactor building to OPERABLE status.	1 hour	3.6.1
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours	3.6.1
	AND		
	B.2 Be in MODE 5.	36 hours	3.6.1

CTS

Reactor Building ~~Containment~~ 3.6.1

1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	Perform required visual examinations and leakage rate testing except for ~~containment~~ reactor building air lock testing, in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions. The leakage rate acceptance criterion is ≤ 1.0 L_a. However, during the first unit startup following testing performed in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, the leakage rate acceptance criteria are < 0.6 L_a for the Type B and Type C tests, and < 0.75 L_a for the Type A test.	~~-----NOTE-----~~ ~~SR 3.0.2 is not applicable~~ In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions The Reactor Building Leakage Rate Testing Program.
~~SR 3.6.1.2~~	~~Verify containment structural integrity in accordance with the Containment Tendon Surveillance Program.~~	~~In accordance with the Containment Tendon Surveillance Program~~

4.4.1.1
4.4.1.1A
4.4.1.2
4.4.1.2.5
1.7e

2

36

CTS
1

REACTOR BUILDING
3.6 CONTAINMENT SYSTEMS
Reactor Building
3.6.2 Containment Air Locks

reactor building
LCO 3.6.2 [Two] containment air lock[s] shall be OPERABLE. 1.7a & b

APPLICABILITY: MODES 1, 2, 3, and 4. 3.6.1

ACTIONS

----------NOTES---------- NA

1. Entry and exit is permissible to perform repairs on the affected air lock components.
2. Separate Condition entry is allowed for each air lock. NA
3. Reactor Building Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when air lock leakage results in exceeding the overall containment leakage rate acceptance criteria. NA

reactor building ----------

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more reactor building containment air locks with one containment air lock door inoperable.	----------NOTES---------- 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls [if both air locks are inoperable]. ----------	NA
	A.1 Verify the OPERABLE door is closed in the affected air lock.	1 hour 3.6.1
	AND	(continued)

3.6.2-4

H-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Lock the OPERABLE door closed in the affected air lock.	24 hours
	AND	
	A.3 --------NOTE--------- Air lock doors in high radiation areas may be verified locked closed by administrative means. ----------	
	Verify the OPERABLE door is locked closed in the affected air lock.	Once per 31 days
B. One or more Reactor building ~~containment~~ air locks with ~~containment~~ air lock interlock mechanism inoperable.	----------NOTES---------- 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit of the reactor building ~~containment~~ is permissible under the control of a dedicated individual. ----------	
	B.1 Verify an OPERABLE door is closed in the affected air lock.	1 hour
	AND	

NA

NA

NA

NA

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Lock an OPERABLE door closed in the affected air lock.	24 hours
	AND	
	B.3 --------NOTE--------- Air lock doors in high radiation areas may be verified locked closed by administrative means. ---------- Verify an OPERABLE door is locked closed in the affected air lock.	Once per 31 days
C. One or more Reactor building ~~containment~~ air locks inoperable for reasons other than Condition A or B.	C.1 Initiate action to evaluate overall Reactor building ~~containment~~ leakage rate per LCO 3.6.1.	Immediately
	AND	
	C.2 Verify a door is closed in the affected air lock.	1 hour
	AND	
	C.3 Restore air lock to OPERABLE status.	24 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours
	AND	
	D.2 Be in MODE 5.	36 hours

NA

3.6.1

3.6.1

H(1)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.6.2.1	----------NOTES---------- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1 ~~in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.~~ ----------		NA NA
	Perform required air lock leakage rate testing in accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions.~~ ~~The acceptance criteria for air lock testing are:~~ ~~a. Overall air lock leakage rate is ≤ [0.05 L_a] when tested at ≥ P_a.~~ ~~b. For each door, leakage rate is ≤ [.01 L_a] when tested at ≥ [10.0 psig].~~	~~----NOTE---- SR 3.0.2 is not applicable ----------~~ In accordance ~~with 10 CFR 50, Appendix J, as modified by approved exemptions~~ the Reactor Building Leakage Rate Testing Program	4.4.1.2 4.4.1.2.5 (2)
SR 3.6.2.2	~~----------NOTE---------- Only required to be performed upon entry or exit through the containment air lock. ----------~~ Verify only one door in the air lock can be opened at a time.	18 months ~~184 days~~	(8) NA

1

REACTOR BUILDING
3.6 ~~CONTAINMENT~~ SYSTEMS

Reactor Building
3.6.3 ~~Containment~~ Isolation Valves

1.7c & d
3.23.1

reactor building
LCO 3.6.3 Each ~~containment~~ isolation valve shall be OPERABLE.

3.6.1
3.6.6
3.23.1

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

7

35

ACTIONS

1.7c.
3.23.1

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

1. Penetration flow paths except for ~~48 inch~~ purge valve penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path. NA

3. Enter applicable Conditions and Required Actions for system(s) made inoperable by ~~containment~~ reactor building isolation valves. NA

4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "~~Containment~~ Reactor Building," when isolation valve leakage results in exceeding the overall ~~containment~~ reactor building leakage rate acceptance criteria.

3.6.3-12

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Only applicable to penetration flow paths with two ~~containment~~ reactor building isolation valves. ---------- One or more penetration flow paths with one ~~containment~~ reactor building isolation valve inoperable ~~(except for purge valve leakage not within limit)~~.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. AND	~~8~~ 48 hours 38

NA
3.6.6

7

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 ---------NOTE--------- (5) (1) Isolation devices in high radiation areas may be verified by use of administrative means. ---------- <INSERT 3.6-8A> Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside ~~containment~~ the reactor building AND (18) NA N/A NA Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside ~~containment~~ the reactor building NA
B. ---------NOTE--------- Only applicable to penetration flow paths with two ~~containment~~ reactor building isolation valves. ---------- One or more penetration flow paths with two ~~containment~~ reactor building isolation valves inoperable ~~(except for purge valve leakage not within limit)~~. (7)	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	1 hour NA

(continued)

<INSERT 3.6-8A>

2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. --------NOTE--------- Only applicable to penetration flow paths with only one reactor building ~~containment~~ isolation valve and a closed system. ----------	C.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	~~[72~~ ~~[1]~~ hours NA 6
	AND	
One or more penetration flow paths with one reactor building ~~containment~~ isolation valve inoperable.	C.2 --------NOTE--------- 5 1. Isolation devices in high radiation areas may be verified by use of administrative means. ---------- 18	
<INSERT 3.6-9A>	Verify the affected penetration flow path is isolated.	Once per 31 days
~~D. One or more penetration flow paths with one or more containment purge valves not within purge valve leakage limits.~~	~~D.1 Isolate the affected penetration flow path by use of at least one [closed and de-activated automatic valve, closed manual valve, or blind flange].~~	~~24 hours~~ 7
	~~AND~~	

(continued)

<INSERT 3.6-9A>

2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means.

1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. (continued)	D.2 --------NOTE--------- Isolation devices in high radiation areas may be verified by use of administrative means. --------- Verify the affected penetration flow path is isolated.	Once per 31 days for isolation devices outside containment AND Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days for isolation devices inside containment
	AND D.3 Perform SR 3.6.3.6 for the resilient seal purge valves closed to comply with Required Action D.1.	Once per [ ] days
D. E. Required Action and associated Completion Time not met.	D. E.1 Be in MODE 3.	6 hours
	AND D. E.2 Be in MODE 5.	36 hours

7

3.6.1

7

3.6.1
3.6.6

1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.6.3.1	Verify each reactor building isolation [48] inch purge valve is sealed closed, except for one purge valve in a penetration flow path while in Condition D of the LCO.	31 days	4.26.1 7
~~SR 3.6.3.2~~	~~Verify each [8] inch purge valve is closed except when the [8] inch purge valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.~~	~~31 days~~	7
SR 3.6.3.~~7~~ 2	----------NOTE---------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ---------- Verify each ~~containment~~ the reactor building isolation manual valve and blind flange that is located outside ~~containment~~ the reactor building and is required to be closed during accident conditions is closed, except for ~~containment~~ isolation valves that are open under administrative controls.	31 days	NA 3.6.5 4

and not locked, sealed, or otherwise secured,

(continued)

1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.6.3.~~1~~ 3	----------NOTE---------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ---------- Verify each ~~containment~~ reactor building isolation manual valve and blind flange that is located inside ~~containment~~ the reactor building and required to be closed during accident conditions is closed, except for ~~containment~~ reactor building isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days	NA 3.6.5
SR 3.6.3.~~2~~ 4	Verify the isolation time of ~~each power operated and~~ each automatic ~~containment~~ reactor building isolation valve is within limits. (Power operated)	[In accordance with the Inservice Testing Program ~~or 92 days~~]	5 4.4.1.4
~~SR 3.6.3.6~~	~~Perform leakage rate testing for containment purge valves with resilient seals.~~	~~184 days AND Within 92 days after opening the valve~~	7
SR 3.6.3.~~7~~ 5	Verify each automatic ~~containment~~ reactor building isolation valve that is not locked, sealed, or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	{18} months	Tbl 4.1-2 #8

(continued)

3.6-7

1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.6.3.8~~	~~Verify each [ ] inch containment purge valve is blocked to restrict the valve from opening > [50]%.~~	~~[18] months~~

7

REACTOR BUILDING

3.6 ~~CONTAINMENT~~ SYSTEMS

3.6.4 Reactor Building ~~Containment~~ Pressure

LCO 3.6.4 Reactor building ~~Containment~~ pressure shall be ≥ ~~[-2.0]~~ -1.0 psig and ≤ ~~[+3.0]~~ psig. 3.6.4

APPLICABILITY: MODES 1, 2, 3, and 4. 3.6.4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. Reactor building ~~Containment~~ pressure not within limits.	A.1 Restore reactor building ~~containment~~ pressure to within limits.	1 hour	3.6.4
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours	3.6.4
	AND		
	B.2 Be in MODE 5.	36 hours	3.6.4

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
3.6.4-1	SR 3.6.4.1 Verify reactor building ~~containment~~ pressure is ~~within limits~~ ≥ -1.0 psig and ≤ +3.0 psig.	12 hours	NA ⊢ edit

CTS

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be ≤ [130]°F.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limit.	A.1 Restore containment average air temperature to within limit.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

12

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify containment average air temperature is within limit.	24 hours

CTS

1

REACTOR BUILDING

3.6 ~~CONTAINMENT~~ SYSTEMS

Reactor Building

3.6 ~~6~~ 5 ~~Containment~~ Spray and Cooling Systems

3.3.1(A&B)
3.3.1 (I)
3.3.4
3.3.4(D)

LCO 3.6 ~~6~~ 5 Two ~~containment~~ reactor building spray trains and two ~~containment~~ reactor building cooling trains shall be OPERABLE.

21

<Insert 3.6-16A>

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

3.3.1
3.3.4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One ~~containment~~ reactor building spray train inoperable in MODE 1 or 2.	A.1 Restore ~~containment~~ reactor building spray train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO	3.3.6 3.3.7(E) NA
~~B.~~ D. Required Action and associated Completion Time of Condition A, B, or C not met.	~~B.1~~ D.1 Be in MODE 3. AND ~~B.2 Be in MODE 5.~~	6 hours ~~84 hours~~	3.3.6 3.3.7(C) (D) & (E) 21
~~B.~~ B. One ~~[required]~~ reactor building ~~containment~~ cooling train inoperable in MODE 1 or 2.	~~B.1~~ B.1 Restore ~~[required]~~ reactor building ~~containment~~ cooling train to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO	3.3.7(C) 3.3.7(D) 3.3.7(E) NA

(continued)

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

Only one train of reactor building spray and one train of reactor building cooling are required to be OPERABLE during MODES 3 and 4. 3.3.1(A) 3.3.1(B)

ITS 3.6.5 ACTIONS NUREG Revisions Reviewer clarification sheet

A.	One reactor building spray train inoperable in MODE 1 or 2.	A.1	Restore reactor building spray train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO
B.	One reactor building cooling train inoperable in MODE 1 or 2.	B.1	Restore reactor building cooling train to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
C.	Two reactor building cooling trains inoperable in MODE 1 or 2.	C.1	Restore one reactor building cooling train to OPERABLE status.	72 hours
D.	Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Be in MODE 3.	6 hours
E.	One required reactor building spray train inoperable in MODE 3 or 4. OR One required reactor building cooling train inoperable in MODE 3 or 4.	E.1	Restore required inoperable train to OPERABLE status.	36 hours
F.	Required Action and associated Completion Time of Condition E not met.	F.1	Be in MODE 5.	36 hours
G.	Two reactor building spray trains inoperable in MODE 1 or 2. OR Any combination of three or more trains inoperable in MODE 1 or 2. OR One required reactor building spray train and one required reactor building cooling train inoperable in MODE 3 or 4.	G.1	Enter LCO 3.0.3.	Immediately

Insert after NUREG pg. 3.6-16

CTS

H 1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
C. ~~B.~~ Two ~~required~~ reactor building ~~containment~~ cooling trains inoperable in MODE 1 or 2 <Insert 3.6-17A>	C. ~~B.~~ 1 Restore one ~~required~~ reactor building ~~containment~~ cooling train to OPERABLE status.	72 hours	3.3.7 (D)
F. ~~D.~~ Required Action and associated Completion Time of Condition ~~B~~ ~~or D~~ not met. E	E.1 Be in MODE 3. AND	6 hours	21
	~~E.2~~ F.1 Be in MODE 5.	36 hours	3.3.6
G. ~~E.~~ Two reactor building ~~containment~~ spray trains inoperable in MODE 1 or 2 OR Any combination of three or more trains inoperable in MODE 1 or 2 <INSERT 3.6-17B>	G. ~~E.~~ 1 Enter LCO 3.0.3.	Immediately	NA

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.6.~~6~~ 5.1	Verify each reactor building ~~containment~~ spray manual, power operated, and automatic valve in ~~the~~ each required flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days	NA H 21

(continued)

<INSERT 3.6-17A>

			CTS
E. One required reactor building spray train inoperable in MODE 3 or 4. OR One required reactor building cooling train inoperable in MODE 3 or 4.	E.1 Restore required inoperable train to OPERABLE status.	36 hours	3.3.6

<INSERT 3.6-17B>

OR One required reactor building spray train and one required reactor building cooling train inoperable in MODE 3 or 4.		

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.6.5.2	Operate each required reactor building ~~containment~~ cooling train fan unit for ≥ 15 minutes.	31 days	4.5.2.1.2 (b)
SR 3.6.5.3	Verify each required reactor building ~~containment~~ cooling train cooling water flow rate is ≥ ~~[1780]~~ 1200 gpm.	31 days	4.5.2.1.2 (a) 4.5.2.1.2(a)(1)
SR 3.6.5.4	Verify each required reactor building ~~containment~~ spray pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program	4.5.2.2.1 L21
SR 3.6.5.5	Verify each automatic each required reactor building ~~containment~~ spray 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 ~~[18]~~ months	4.5.2.1.1(a) L21
SR 3.6.5.6	Verify each required reactor building ~~containment~~ spray pump starts automatically on an actual or simulated actuation signal.	18 ~~[18]~~ months	4.5.2.1.1 (a) L21
SR 3.6.5.7	Verify each required reactor building ~~containment~~ cooling train starts automatically on an actual or simulated actuation signal.	18 ~~[18]~~ months	4.5.2.1.2 (c) & (c)(1) & (c)(3)

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.6.5.8 Verify each required train spray nozzle is unobstructed.	~~At first refueling~~ ~~AND~~ 10 years

1

2L

4.5.2.1.1 (b)

REACTOR BUILDING

3.6 ~~CONTAINMENT~~ SYSTEMS

3.6.7 6 Spray Additive System

LCO 3.6.7 6 The Spray Additive System shall be OPERABLE. 3.3.4 & (C) & (D)

APPLICABILITY: MODES 1, 2, ~~3, and 4~~ and 3.3.4 21

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. Spray Additive System inoperable.	A.1 Restore Spray Additive System to OPERABLE status.	72 hours	3.3.6
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours	3.3.6
	~~AND~~		21
	~~B.2 Be in MODE 5.~~	~~84 hours~~	

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.6.7 6.1	Verify each spray additive System manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days	9 NA

(continued)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.6.7.2	Verify Sodium hydroxide spray additive tank solution volume is ≥ [12,970] gal and ≤ [13,920] gal. 9000	184 days	9 3.3.4 (B) 23
SR 3.6.7.3	Verify Sodium hydroxide spray additive tank [NaOH] solution concentration is ≥ [50,000 ppm] and ≤ [65,000 ppm]. >5.0 wt% and <16.5 wt% NaOH	184 days	3.3.4(B) T 4.1-3 #6.
SR 3.6.7.4	Verify each spray additive System automatic valve in the flow path actuates to the correct position on an actual or simulated actuation signal.	18 [18] months	9 NA
~~SR 3.6.7.5~~	~~Verify Spray Additive System flow [rate] from each solution's flow path.~~	~~5 years~~	13

CTS

1

REACTOR BUILDING

3.6 ~~CONTAINMENT~~ SYSTEMS

edit

3.6.8 Hydrogen Recombiners ~~(if permanently installed)~~
7

LCO 3.6.8 Two hydrogen recombiners shall be OPERABLE. 3.14.1
7

APPLICABILITY: MODES 1 and 2. 3.14.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One hydrogen recombiner inoperable.	A.1 --------NOTE--------- LCO 3.0.4 is not applicable. ---------- Restore hydrogen recombiner to OPERABLE status.	30 days	NA 3.14.2
~~B. Two hydrogen recombiners inoperable.~~	~~B.1 Verify by administrative means that the hydrogen control function is maintained.~~ ~~AND~~ ~~B.2 Restore one hydrogen recombiner to OPERABLE status.~~	~~1 hour~~ ~~AND~~ ~~Every 12 hours thereafter~~ ~~7 days~~	10
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours	3.14.2

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.6.7.1	Perform a system functional test for each hydrogen recombiner.	18 ~~[18]~~ months	4.12.1 a.
SR 3.6.7.2	Visually examine each hydrogen recombiner enclosure and verify there is no evidence of abnormal conditions.	18 ~~[18]~~ months	4.12.1 b.
SR 3.6.7.3	Perform a resistance to ground test for each heater phase.	18 ~~[18]~~ months	4.12.1 b.

REACTOR BUILDING

B 3.6 CONTAINMENT SYSTEMS
Reactor Building
B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of the concrete reactor building (RB), its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. For containments with ungrouted tendons, the cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions, and the dome roof is prestressed using a three way post tensioning system. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

The concrete RB is required for structural integrity of the containment under DBA conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions.

The isolation devices for the penetrations in the containment boundary are a part of the containment leak tight barrier. To maintain this leak tight barrier:

a. All penetrations required to be closed during accident conditions are either:

1. capable of being closed by an OPERABLE automatic containment isolation system, or

<Insert B3.6-1A>

(continued)

<INSERT B3.6-1A>

except as provided in LCO 3.3.5, "Engineered Safeguards Actuation System (ESAS) Instrumentation," LCO 3.3.6, "ESAS Manual Initiation," and LCO 3.3.7, "ESAS Actuation Logic"

BASES

BACKGROUND (continued)

2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks"; and

c. the ~~All~~ equipment hatches ~~are~~ is closed ~~/~~ and sealed

~~d. The pressurized sealing mechanism associated with each penetration, except as provided in LCO 3.6.[ ], is OPERABLE.~~

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a ~~loss of coolant accident (LOCA), a steam line break, and a rod ejection accident (REA)~~ (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA ~~or REA~~. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of ~~[0.25]~~ 0.2 % of containment air weight per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable leakage rate at the calculated maximum peak containment pressure (P_a) resulting from the limiting ~~DBA~~ design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be ~~[0.25]~~ 0.2 % per day in the safety analysis at P_a = ~~[53.8]~~ 54.0 psig (Ref. 3).

~~Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.~~

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

In MODES 3 and 4, the reactor building satisfies Criterion 4 of 10 CFR 50.36

(continued)

BASES (continued)

Reactor Building Leakage Rate Testing Program

LCO

the applicable leakage limits must be met

<INSERT B3.6-3A>

in conjunction with LCO 3.6.2

3.6.1-3

Containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required (10 CFR 50, Appendix J) leakage test. At this time, the combined Type B and C leakage must be < 0.6 L_a, and the overall Type A leakage must be < 0.75 L_a. Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates specified for the containment air lock (LCO 3.6.2) [and purge valves with resilient seals ([LCO 3.6.3])] are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of Appendix J 1.0 L_a.

APPLICABILITY

<Insert B3.6-3B>

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.3, "~~Containment~~ Reactor Building Penetrations."

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining ~~containment~~ reactor building OPERABILITY during MODES 1, 2, 3, and 4. This time period also ensures the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

(continued)

<INSERT B3.6-3A>

3.6.1-3

Reactor building OPERABILITY for leakage is attained by ensuring that the equipment hatch and both doors of the personnel and emergency air locks are closed and sealed, except as appropriate for maintenance activities, and that the other isolation devices are closed, deactivated in the closed position, or OPERABLE as required. Reactor building OPERABILITY is also maintained by monitoring the deviation of key design parameters of the RB structure from the original design configuration and ensuring that structural limits are not exceeded. Visual and other required examinations of tendons, anchorages and surfaces are performed periodically in accordance with station procedures. These procedures embody applicable requirements of the 1992 Edition with the 1992 Addenda of Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code, as set forth by 10CFR50.55a(g)(6)(ii)(B) (Ref. 5). Any degradations exceeding the Containment Inspection Program acceptance criteria during inspection surveillances will be reviewed under an engineering evaluation within 60 days of the completion of the inspection to determine what impact the degradation has on overall reactor building OPERABILITY, is if any.

<INSERT B3.6-3B>

In MODES 1, 2, 3 and 4, the reactor building OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES.

BASES

ACTIONS (continued)

B.1 and B.2

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

If the Required Actions and associated Completion Times are not met

Unit

edit

edit

edit

edit

unit

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of 10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions. Failure to meet air lock and purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test is required to be $\leq 0.6\ L_a$ for combined Type B and C leakage, and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by Appendix J, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

the Reactor Building Leakage Rate Testing Program

2

7

its

edit

Reactor Building Leakage Rate Testing Program

2

the Reactor Building Leakage Rate Testing Program

2

3.6.1-3

~~SR 3.6.1.2~~

~~For ungrouted, post tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and~~

36

~~(continued)~~

H1

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.6.1.2 (continued)~~

~~Frequency are consistent with the recommendations of Regulatory Guide 1.35 (Ref. 4).~~

36

REFERENCES

1. 10 CFR 50, Appendix J, Option B
2. FSAR, Chapter 14, ~~Sections 14.1 and 14.2~~. edit
3. FSAR, Section ~~15.6~~ 5.2. edit
4. ~~Regulatory Guide 1.35, Revision [1]~~ 36

28

10CFR 50.36.

5. 10CFR50.55a (g)(6)(ii)(B). 36

H2

3.6.1-3

1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

11

BACKGROUND

Containment air locks (also known as the personnel air lock and the emergency (or escape) air lock) form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

42

Each air lock is nominally a right circular cylinder, ~~10 ft~~ in diameter, with a door at each end. The doors are interlocked to prevent simultaneous opening. ~~During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary.~~ Each air lock door has been designed and is tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. As such, closure of a single door supports containment OPERABILITY. Each of the doors contains double gasketed seals and local leakage rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door).

32

3.6.2-5

~~Each personnel air lock door is provided with limit switches that provide control room indication of door position. Additionally, control room indication is provided to alert the operator whenever an air lock door interlock mechanism is defeated.~~

3

The containment air locks form part of the containment pressure boundary. As such, air lock integrity and leak tightness ~~is~~ are essential for maintaining the containment leakage rate within limit in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.

edit

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA), a steam line break, ~~and a rod ejection accident~~ (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of ~~[0.25]%~~ ≤0.2% of containment air weight per day (Ref. 3). This leakage rate is defined in 10 CFR 50, Appendix J (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated maximum peak containment pressure, (P_a), following a ~~DBA~~ design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock. ~~L_a is [0.25]% per day and P_a is [53.9] psig, resulting from the limiting design basis LOCA.~~

In MODES 1 and 2, the containment air locks satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10CFR50.36 (Ref. 4).

In MODES 3 and 4, the reactor building air locks satisfy Criterion 4 of 10CFR50.36

LCO

Each containment air lock forms the part of the containment pressure boundary. As a part of containment, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into and exit from containment.

(ie., Closed and Sealed)

and Sealing

normally

or

and Sealed

BASES (continued)

APPLICABILITY

<Insert B3.6-8A>

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Reactor Building Penetrations."

but capable of being swung

and the airlock barrel

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this not is practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

an inoperable inner door

Opening

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.

In the event the air lock leakage results in exceeding the overall containment leakage rate, Note 3 directs entry into the applicable Conditions and Required Actions of LCO 3.6.1, "Containment."

(continued)

<INSERT B3.6-8A>

In MODES 1, 2, 3 and 4, the reactor building air lock OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES.

H-1

BASES

ACTIONS (continued)

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

With one air lock door inoperable in one or more containment air locks, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock.

This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires containment be restored to OPERABLE status within 1 hour.

In addition, the affected air lock penetration must be isolated by locking closed the remaining OPERABLE air lock door within the 24 hour Completion Time. The 24 hour Completion Time is considered reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls. Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of the door, once it has been verified to be in the proper position, is small. edit

The Required Actions have been modified by two Notes. Note 1 clarifies that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. The exception of Note 1 does not affect tracking the Completion Time from the

3.6.2-4

(continued)

1

BASES

ACTIONS

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

initial entry into Condition A; only the requirement to comply with the Required Actions. Note 2 allows use of the air lock for entry and exit for 7 days under administrative controls if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment was entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

3.6.2-4

B.1, B.2, and B.3

With an air lock interlock mechanism inoperable in one or more air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A.

The Required Actions have been modified by two Notes. Note 1 clarifies that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1 and C.2 are the appropriate remedial actions. Note 2 allows entry into and exit from the containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

Required Action B.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the

(continued)

1

BASES

ACTIONS

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

probability of misalignment of the door, once it has been verified to be in the proper position, is small.

C.1, C.2, and C.3

With one or more air locks inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be immediately initiated to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or if the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour (per LCO 3.6.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a ~~plant~~ unit shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits. edit

Required Action C.2 requires that one door in the affected containment air lock must be verified to be closed. This action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which requires that containment be restored to OPERABLE status within 1 hour.

Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. The specified time period is considered reasonable for restoring an inoperable air lock to OPERABLE status assuming that at least one door is maintained closed in each affected air lock.

3.6.2-6

If the Required Actions and associated Completion Times are not met

D.1 and D.2

~~If the inoperable containment air lock cannot be restored to OPERABLE status within the Required Completion Time, the~~ ~~plant~~ unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the ~~plant~~ unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, edit edit edit

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

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

edit

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of ~~10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions~~ the Reactor Building Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by ~~Appendix J, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply.~~

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable, since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria ~~of~~ which is applicable to SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the ~~overall~~ Combined Type B and C containment leakage rate.

3.6.2-2

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.2 (continued)

<Insert B3.6-13 A>

<Insert B3.6-13 B>

<Insert B3.6-13 C>

will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is ~~only~~ not normally challenged when the containment air lock door is ~~opened~~, this test is only required to be performed upon entering or exiting a containment air lock but is not required more frequently than every ~~184 days~~ 18 month. The ~~184 day~~ frequency is based on engineering judgment and is considered adequate in view of other indications of door and interlock mechanism status available to operations personnel.

REFERENCES

1. 10 CFR 50, Appendix J, Option B
2. ~~FSAR, Sections [14.1 and 14.2]~~ Chapter 14 edit
3. ~~FSAR, Section [5.6]~~ Chapter 5 edit
4. 10 CFR 50.36.

<Insert B3.6-13A>

used for entry and exit (procedures require strict adherence to single door opening),

<Insert B3.6-13B>

every 18 months. 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 loss of reactor building OPERABILITY if the Surveillance were performed with the reactor at power. The 18 month Frequency for the interlock is justified based on generic operating experience.

<Insert B3.6-13C>

given that the interlock is not expected to be challenged during use of the airlock.

1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on an automatic isolation signal. These isolation devices consist of either passive devices or active (automatic) devices. Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close following an accident without operator action, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers (typically containment isolation valves) make up the Containment Isolation System.

3.6.3-2

Containment isolation occurs upon receipt of a high containment pressure ~~or diverse containment isolation~~ signal. The containment isolation signal closes automatic containment isolation valves in fluid penetrations not required for operation of engineered safeguard systems to prevent leakage of radioactive material. ~~Upon actuation of high pressure injection, automatic containment valves also isolate systems not required for containment or Reactor Coolant System (RCS) heat removal.~~ Other penetrations are isolated by the use of valves in the closed position or blind flanges. As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated in the event of a release of radioactive material to containment atmosphere from the RCS following a Design Basis Accident (DBA).

Also, upon receipt of a low RCS pressure signal certain

edit

edit

edit

OPERABILITY of the containment isolation valves (and blind flanges) supports containment OPERABILITY during accident conditions.

(continued)

1

BASES

BACKGROUND (continued)

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the time limits assumed in the safety analysis. Therefore, the OPERABILITY requirements provide assurance that the containment function assumed in the safety analysis will be maintained. 30

The Reactor Building Purge System is part of the Reactor Building Ventilation System. The Purge System was designed for intermittent operation, providing a means of removing airborne radioactivity caused by minor leakage from the RCS prior to personnel entry into containment. The Containment Purge System consists of one [48] inch line for exhaust and one [48] inch line for supply, with supply and exhaust fans capable of purging the containment atmosphere at a rate of approximately [50,000] ft^3/min. This flow rate is sufficient to reduce the airborne radioactivity level within containment to levels defined in 10 CFR 20 (Ref. 1) for a 40 hour workweek within 2 hours of purge initiation during reactor operation. The containment purge supply and exhaust lines each contain two isolation valves that receive an isolation signal on a unit vent high radiation condition. 7

24

reactor building isolation Signal

Failure of the purge valves to close following a design basis event would cause a significant increase in the radioactive release because of the large containment leakage path introduced by these [48] inch purge lines. Failure of the purge valves to close would result in leakage considerably in excess of the containment design leakage rate of [0.25]% of containment air weight per day (L_a) (Ref. 2). Because of their large size, the [48] inch purge valves in some units are not qualified for automatic closure from their open position under DBA conditions. Therefore, the [48] inch purge valves are maintained sealed closed (SR 3.6.3.1) in MODES 1, 2, 3, and 4 to ensure the containment boundary is maintained. 7

24

≤0.2

24

1

24

tested

with the hand switch keys removed

Supply and exhaust

The [8 inch] containment minipurge valves operate to:

a. Reduce the concentration of noble gases within containment prior to and during personnel access; and

b. Equalize internal and external pressures.

Since the minipurge valves are designed to meet the requirements for automatic containment isolation valves, 7

(continued)

1

BASES

BACKGROUND (continued)

these valves may be opened as needed in MODES 1, 2, 3, and 4.

7

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analysis of any event requiring isolation of containment is applicable to this LCO.

3.6.3-18

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA), a main steam line break, and a rod ejection accident (Ref. 8). In the analysis for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analysis assumes that the [48] inch purge valves are closed at event initiation.

and 2 edit 30 24

<Insert B3.6-16A>

The DBA analysis assumes that, within 60 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, L_a. The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times.

30

The single-failure criterion required to be imposed in the conduct of unit safety analyses was considered in the original design of the containment purge valves. Two valves in a series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. The inboard and outboard isolation valves on each line are provided with diverse power sources, motor operated and pneumatically operated spring closed, respectively. This arrangement was designed to preclude common mode failures from disabling both valves on a purge line.

7

(continued)

<Insert B3.6-16A>

The LOCA analysis assumes a fixed amount of core inventory escapes. No mechanistic scenario is evaluated to determine what portion of the inventory is released prior to closure of the reactor building isolation valves. Industry standards for sizing valve operators govern the closure times of the reactor building isolation valves.

This prevents inadvertent operation of the reactor building purge valves while in MODES 1, 2, 3 and 4.

AND-1 does not currently allow reactor building purging in MODES 1, 2, 3, and 4.

with its handswitch Key removed

BASES

reactor building

APPLICABLE SAFETY ANALYSES (continued)

3.6.3-14

In MODES 3 and 4, the reactor building isolation valves satisfy Criterion 4 of 10 CFR 50.36.

The purge valves may be unable to close in the environment following a LOCA. Therefore, each of the purge valves is required to remain sealed closed during MODES 1, 2, 3, and 4. In this case, the single-failure criterion remains applicable to the containment purge valves because of failure in the control circuit associated with each valve. Again, the purge system valve design prevents a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO.

In MODES 1 and 2, The containment isolation valves satisfy Criterion 3 of the NRC Policy Statement.

10 CFR 50.36 (Ref. 3)

3.6.3-16

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valve safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA.

reactor building

edit

24

The automatic power operated isolation valves are required to have isolation times within limits and to actuate on an automatic isolation signal. The [48] inch purge valves must be maintained sealed closed [or have blocks installed to prevent full opening]. [Blocked purge valves also actuate on an automatic signal.] The valves covered by this LCO are listed along with their associated stroke times in the FSAR (Ref. 4).

3.6.3-15

Their associated stroke times are contained in the Inservice Testing Program.

reactor building

the

The normally closed isolation valves are considered OPERABLE when manual valves are closed, check valves have flow through the valve secured, and blind flanges are in place, and closed systems are intact. These passive isolation valves/devices are those listed in Reference 4.

3.6.3-17

or open under administrative Controls

Purge valves with resilient seals must meet additional leakage rate requirements. The other containment isolation valve leakage rates are addressed by LCO 3.6.1, "Containment," as Type C testing.

This LCO provides assurance that the containment isolation valves and purge valves will perform their designated safety functions to minimize the loss of reactor coolant inventory and establish the containment boundary during accidents.

(continued)

BASES (continued)

APPLICABILITY

(Insert B3.6-18A)

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment isolation valves are not required to be OPERABLE in MODE 5. The requirements for containment isolation valves during MODE 5 and 6 are addressed in LCO 3.9.3, "Containment Penetrations." (, primarily related to movement of irradiated fuel in the reactor building,)

edit

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow paths, except for [48] inch purge valve penetration flow paths, to be unisolated intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator (individual) at the valve controls, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for containment isolation is indicated. Due to the size of the containment purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, the penetration flow paths containing these valves may not be opened under administrative controls. A single purge valve in a penetration flow path may be opened to effect repairs to an inoperable valve, as allowed by SR 3.6.3.1.

Due to ALARA concerns, it is permissible for the dedicated individual to be stationed in a nearby lower dose area, provided the intent of rapidly isolating the penetration is retained.

3.6.3-19

3.6.3-20

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable containment isolation valve. Complying with the Required Actions may allow for continued operation, and subsequent inoperable containment isolation valves are governed by subsequent Condition entry and application of associated Required Actions.

edit

The ACTIONS are further modified by a third Note, which ensures appropriate remedial actions are taken, if necessary, if the affected systems are rendered inoperable by an inoperable containment isolation valve.

In the event isolation valve leakage results in exceeding the overall containment leakage rate, Note 4 directs entry

(continued)

<INSERT B3.6-18A>

In MODES 1, 2, 3 and 4, the reactor building isolation valves OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES.

BASES

ACTIONS (continued)

~~into the applicable Conditions and Required Actions of LCO 3.6.1.~~

A.1 and A.2

In the event one containment isolation valve in one or more penetration flow paths is inoperable ~~(except for purge valve leakage not within limit)~~, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic containment isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. For a penetration isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to containment. Required Action A.1 must be completed within the 4 hour Completion Time. The specified time period is reasonable, considering the time required to isolate the penetration and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4.

For affected penetration flow paths that cannot be restored to OPERABLE status within the 4 hour Completion Time and that have been isolated in accordance with Required Action A.1, the affected penetration flow paths must be verified to be isolated on a periodic basis. This periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident and no longer capable of being automatically isolated will be in the isolation position should an event occur. This Required Action does not require any testing or device manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment and capable of being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside containment" is appropriate considering the fact that the devices are operated under administrative controls and the probability of their misalignment is low. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if not performed within the previous

(continued)

H1

BASES

ACTIONS

A.1 and A.2 (continued)

92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.

edit

Condition A has been modified by a Note indicating this Condition is only applicable to those penetration flow paths with two containment isolation valves. For penetration flow paths, with only one containment isolation valve and a closed system, Condition C provides appropriate actions.

In closed systems

edit

Required Action A.2 is modified by a Note that applies to isolation devices located in high radiation areas and allows the devices to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable since access to these areas is typically restricted. Therefore, the probability of misalignment of these devices, once they have been verified to be in the proper position, is small.

<INSERT B3.6-20A>

18

B.1

With two containment isolation valves in one or more penetration flow paths inoperable (except for purge valve leakage not within limit), the affected penetration flow path must be isolated within 1 hour. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1. In the event the affected penetration is isolated in accordance with Required Action B.1, the affected penetration must be verified to be isolated on a periodic basis per Required Action A.2, which remains in effect. This periodic verification is necessary to assure leak tightness of containment and that penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying each affected penetration flow path is isolated is appropriate considering the fact that the valves are

H7

(continued)

<INSERT B3.6-20A>

Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned.

1

BASES

ACTIONS

B.1 (continued)

operated under administrative controls and the probability of their misalignment is low.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two containment isolation valves. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

C.1 and C.2

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration. Required Action C.1 must be completed within the ~~[4]~~ 72 hour Completion Time. The specified time period is reasonable, considering the relative ~~stability~~ structural integrity of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of supporting containment OPERABILITY during MODES 1, 2, 3, and 4. In the event the affected penetration is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure ~~leak tightness of containment and~~ that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate considering the fact that the valves are operated under administrative controls and the probability of their misalignment is low.

6 edit

edit

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. This Note is necessary since this Condition is

<INSERT B 3.6-21A>

6

3.6.3-9

(continued)

<INSERT B3.6-21A>

3.6.3-9

The service water system is the only closed system within the reactor building to which Specification 3.6.3 Condition C applies. The service water system within the reactor building is designed to seismic category 1 standards. Because the system is located outside of the secondary shield walls, it is protected from missiles and pipe whip from reactor coolant system components. The service water system is capable of withstanding reactor building design pressure and temperature and is designed to withstand the LOCA accident transient and environment.

1

BASES

ACTIONS

C.1 and C.2 (continued)

written to specifically address those penetration flow paths in a closed system.

Required Action C.2 is modified by two Notes. Note 1 applies to valves and blind flanges located in high radiation areas and allows these devices to be verified by use of administrative means. Allowing verification by administrative means is considered acceptable since access to these areas is typically restricted. Therefore, the probability of misalignment of these devices, once verified to be in the proper position, is small.

18

<INSERT B3.6-22A>

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

~~In the event one or more containment purge valves in one or more penetration flow paths are not within the purge valve leakage limits, purge valve leakage must be restored to within limits or the affected penetration flow path must be isolated. The method of isolation must be by the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a [closed and de-activated automatic valve, closed manual valve, and blind flange]. A purge valve with resilient seals utilized to satisfy Required Action D.1 must have been demonstrated to meet the leakage requirements of SR 3.6.3.6. The specified Completion Time is reasonable, considering that one containment purge valve remains closed so that a gross breach of containment does not exist.~~

~~In accordance with Required Action D.2, this penetration flow path must be verified to be isolated on a periodic basis. The periodic verification is necessary to ensure that containment penetrations required to be isolated following an accident, which are no longer capable of being automatically isolated, will be in the isolation position should an event occur. This Required Action does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those isolation devices outside containment and potentially capable of being mispositioned are in the correct position. For the isolation devices inside containment, the time period specified as "prior to entering MODE 4 from MODE 5 if~~

7

(continued)

<INSERT B3.6-22A>

Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned.

1

BASES

ACTIONS

~~D.1, D.2, and D.3 (continued)~~

~~not performed within the previous 92 days" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the isolation devices and other administrative controls that will ensure that isolation device misalignment is an unlikely possibility.~~

~~For the containment purge valve with resilient seal that is isolated in accordance with Required Action D.1, SR 3.6.3.6 must be performed at least once every [ ] days. This provides assurance that degradation of the resilient seal is detected and confirms that the leakage rate of the containment purge valve does not increase during the time the penetration is isolated. The normal Frequency for SR 3.6.3.6, 184 days, is based on an NRC initiative, Generic Issue B-20 (Ref. 7). Since more reliance is placed on a single valve while in this Condition, it is prudent to perform the SR more often. Therefore, a Frequency of once per [ ] days was chosen and has been shown acceptable based on operating experience.~~

7

~~D.1 and D.2~~

edit

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

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1

Each [48] 24 inch containment purge isolation valve in the Purge System Supply and exhaust is required to be verified ~~sealed~~ closed at 31 day intervals. This Surveillance is designed to ensure that a gross breach of containment is not caused by an inadvertent ~~or spurious~~ opening of a containment purge valve. Detailed analysis of the purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to limit offsite doses. Therefore, these valves are required to be in the

7

(continued)

1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1 (continued)

~~sealed~~ closed position during MODES 1, 2, 3, and 4. A containment purge valve that is ~~sealed~~ closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or by removing the air supply to the valve operator. In this application, the term "sealed" has no connotation of leak tightness. The Frequency is a result of an NRC initiative, Generic Issue B-24 (Ref. 6), related to containment purge valve use during unit operations. In the event purge valve leakage requires entry into Condition D, the Surveillance permits opening one purge valve in a penetration flow path to perform repairs.

edit

7

removing the Valve handswitch key

consistent with other reactor building isolation valves discussed in SR 3.6.3.2.

SR 3.6.3.2

This SR ensures that the minipurge valves are closed as required or, if open, open for an allowable reason. If a purge valve is open in violation of this SR, the valve is considered inoperable. If the inoperable valve is not otherwise known to have excessive leakage when closed, it is not considered to have leakage outside of limits. The SR is not required to be met when the minipurge valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The minipurge valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.3.

7

SR 3.6.3.~~3~~ 2

edit

4

and not locked, sealed, or otherwise secured,

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those containment isolation valves outside containment and capable

(continued)

1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.~~6~~ 2 (continued)

edit

of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside ~~containment is relatively easy,~~ the 31 day Frequency ~~is based on engineering judgment and~~ was chosen to provide added assurance of the correct positions. The SR specifies that containment isolation valves open under administrative controls are not required to meet the SR during the time the valves are open.

edit

4

Insert B3.6-25A

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is low.

SR 3.6.3.~~4~~ 3

edit

This SR requires verification that each containment isolation manual valve and blind flange that is located inside containment and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate, since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. The SR specifies that containment isolation valves open under administrative controls are not required to meet the SR during the time they are open.

4

and not locked, sealed, or otherwise secured

3.6.3-7

4

Insert B3.6-25A

The Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since the access to these areas is typically restricted during MODES 1, 2, 3, and 4 for ALARA reasons. Therefore, the

(continued)

<Insert B3.6-25A>

This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.~~4~~ 3 (continued)

edit

probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.~~5~~ 4

edit

Verifying that the isolation time of each ~~power operated and~~ automatic containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period ~~less than or equal to that assumed in the safety analyses~~. The isolation time and frequency of this SR are in accordance with the Inservice Testing Program ~~or 92 days~~.

5

30

edit

Power operated

Consistent with industry standards for sizing valve operators

SR 3.6.3.6

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of once per 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 7).

Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (greater than that occurring to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.

7

SR 3.6.3.~~7~~ 5

edit

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.~~7~~ 5 (continued)

radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This SR is not required for valves that are locked, sealed, or otherwise secured in position under administrative controls. The ~~[18]~~ month Frequency is based on the need to perform this Surveillance under the conditions that apply during a ~~plant~~ unit outage and on the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the ~~[18]~~ month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

~~SR 3.6.3.8~~

~~[Reviewer's Note: This SR is only required for those units with resilient seal purge valves allowed to be open during [MODE 1, 2, 3, or 4] and having blocking devices on the valves that are not permanently installed.]~~

~~Verifying that each [48] inch containment purge valve is blocked to restrict opening to ≤ [50%] is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 3 and 4. If a LOCA occurs, the purge valves must close to maintain containment leakage within the values assumed in the accident analysis. At other times when purge valves are required to be capable of closing (e.g., during movement of irradiated fuel assemblies), pressurization concerns are not present, thus the purge valves can be fully open. The [18] month Frequency is appropriate because the blocking devices are typically removed only during a refueling outage.~~

REFERENCES

1. ~~10 CFR 20.~~ Chapter 5
2. ~~FSAR, Section [8.8].~~ Chapter 14
3. ~~FSAR, Sections [14.1 and 14.2].~~
4. 10CFR 50.36.

(continued)

H 1

BASES

REFERENCES (continued)

4. FSAR, Section [6.3] Table 5-1 edit
5. ~~FSAR, Section [5.3]~~ edit
6. ~~Generic Issue B-24.~~
7. ~~Generic Issue B-20.~~

H 7

1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere in the event of inadvertent actuation of the Containment Spray System. 37 14

Insert B3.6-29A

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside these limits coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values. 37 edit

DBA and ECCS analyses

and ECCS performance

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB, which are analyzed using computer pressure transients. The worst-case LOCA generates larger mass and energy release than the worst-case SLB. Thus, the LOCA event bounds the SLB event from the containment peak pressure standpoint (Ref. 1). edit

The initial pressure condition used in the containment analysis was [17.7] psia ([3.0] psig). This resulted in a maximum peak pressure from a LOCA of [53.8] psig. The LCO limit of [3.0] psig ensures that, in the event of an accident, the design pressure of [55] psig for containment is not exceeded. In addition, the Building was designed for an internal pressure equal to [3] psig above external pressure during a tornado. The containment was also designed for an internal pressure equal to [2.5] psig below external pressure, to withstand the resultant pressure drop from an accidental actuation of the Containment Spray System. The LCO limit of [-2.0] psig ensures that operation 14 14.7 3.0 59 14 37 -1.0

(continued)

<Insert B3.6-29A>

Additionally, keeping the reactor building pressure within the limits maintains the initial conditions assumed for the reactor building design basis accident (DBA) and Emergency Core Cooling System (ECCS) analyses.

BASES <INSERT B3.6-30C>

APPLICABLE SAFETY ANALYSES (continued)

within the design ~~limit of [-2.5] psig~~ assumptions for ECCS is maintained (Ref. 2).

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling Systems during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase the containment backpressure is ~~calculated~~ assumed in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. ~~2~~ 3).

In MODES 1 and 2, Containment pressure satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10CFR 50.36 (Ref. 4).

In MODES 3 and 4, reactor building pressure satisfies Criterion 4 of 10 CFR 50.36.

LCO

Maintaining containment pressure less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. ~~Maintaining containment pressure greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Containment Spray System.~~

<Insert B3.6-30A>

APPLICABILITY <Insert B3.6-30B>

~~In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within design basis limits is essential to ensure initial conditions assumed in the accident analysis are maintained, the LCO is applicable in MODES 1, 2, 3, and 4.~~

In MODES 5 and 6, the probability and consequences of ~~these~~ events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODES 5 and 6.

(continued)

<Insert B3.6-30A>

Additionally, keeping the reactor building pressure within the limits maintains the initial conditions assumed for the ECCS analyses.

<Insert B3.6-30B>

In MODES 1, 2, 3 and 4, the reactor building OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES. Since maintaining reactor building pressure within design basis limits is essential to ensure that the peak reactor building pressure from an accident does not exceed the reactor building design pressure, the LCO is applicable in MODES 1, 2, 3, and 4.

<Insert B3.6-30C>

3.6.4-4

The LCO limit of 3.0 psig does not consider instrument uncertainty. The LCO limit of -1.0 psig is considered to be an as-indicated value.

1

BASES (continued)

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, containment pressure must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the bounds of the containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

If the Required Actions and associated Completion Times are not met

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that operation remains within the limits assumed in the ECCS and containment analyses. The 12 hour Frequency of this SR was developed after taking into consideration operating experience related to trending of containment pressure variations during the applicable MODES. 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 an abnormal containment pressure condition. 14

3.6.4-1

REFERENCES

1. ~~UFSAR, Section 14.2~~ Chapter 14. edit

2. 10 CFR 50, Appendix K. edit

3. SAR, Chapter 5. edit

4. 10 CFR 50.36. 28

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND

The containment structure serves to contain radioactive material, which may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. This LCO ensures that initial conditions assumed in the analysis of a DBA are not violated during unit operations. The total amount of energy to be removed from the Containment Cooling System during post accident conditions is dependent upon the energy released to the containment due to the event as well as the initial containment temperature and pressure. The higher the initial temperature, the higher the resultant peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

12

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses. Average air temperature is also used to establish the containment environmental qualification operating envelope. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analysis for containment.

Several accidents (primarily LOCA and SLB) result in a marked increase in containment temperature and pressure due to energy release within the containment. Of these, the LOCA results in the greatest sustained increase in containment temperature. By maintaining containment air temperature at less than the initial temperature assumed in

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

the LOCA analysis, the reactor building design condition will not be exceeded.

The LOCA that was identified as presenting the greatest challenge to containment OPERABILITY was a cold leg Reactor Coolant System break, of specified size, at a reactor coolant pump suction.

Containment average air temperature satisfies Criterion 2 of the NRC Policy Statement.

12

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the containment design temperature. As a result, the ability of containment to perform its design function is ensured.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment average air temperature within the limit is not required in MODE 5 or 6.

ACTIONS

A.1

When containment average air temperature is not within the limit of the LCO, it must be restored within 8 hours. This Required Action is necessary to return operation to within the bounds of the containment analysis. The 8 hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter and provides sufficient time to correct minor problems.

B.1 and B.2

If the containment average air temperature cannot be restored to within its limit within the required Completion

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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

12

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1

Verifying that containment average air temperature is within the LCO limit ensures that containment operation remains within the limit assumed for the containment analyses. In order to determine the containment average air temperature, an arithmetic average is calculated, using measurements taken at locations within the containment selected to provide a representative sample of the overall containment atmosphere. The 24 hour Frequency of this SR is considered acceptable based on observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment temperature condition.

REFERENCES

None.

1

B 3.6 CONTAINMENT SYSTEMS

B 3.6 5 Containment Spray and Cooling Systems

BASES

BACKGROUND

The Containment Spray and Containment Cooling systems provide containment atmosphere cooling to limit post accident pressure and temperature in containment to less than the design values. Reduction of containment pressure and the iodine removal capability of the spray reduces the release of fission product radioactivity from containment to the environment, in the event of a Design Basis Accident (DBA), to within limits. The Containment Spray and Containment Cooling systems are designed to meet the requirements of 10 CFR 50, Appendix A, GDC 38, "Containment Heat Removal," GDC 39, "Inspection of Containment Heat Removal Systems," GDC 40, "Testing of Containment Heat Removal Systems," GDC 41, "Containment Atmosphere Cleanup," GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems" (Ref. 1), or other documents that were appropriate at the time of licensing (identified on a unit specific basis).

21 edit

27 edit edit edit

as discussed in the Safety Analysis Report (SAR), specifically

27

The Containment Cooling System and Containment Spray System are Engineered Safety Feature (ESF) systems. They are designed to ensure that the heat removal capability required during the post accident period can be attained. The Containment Spray System and Containment Cooling System provide redundant containment heat removal operation. The Containment Spray System and Containment Cooling System provide redundant methods to limit and maintain post accident conditions to less than the containment design values.

Safeguards (ES)

edit

Containment Spray System

The Containment Spray System consists of two separate trains of equal capacity, each capable of meeting the design basis. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate ESF bus. The borated water storage tank (BWST) supplies borated water to the Containment Spray System during the injection phase of operation. In the recirculation mode of operation, Containment Spray System

ES

edit

(continued)

1

BASES

BACKGROUND

Containment Spray System (continued)

pump suction is manually transferred ~~from the BWST~~ to the containment sump.

21

33

The Containment Spray System provides a spray of ~~relatively cold~~ borated water ~~mixed with sodium hydroxide from the spray additive tank~~ into the upper regions of containment to ~~reduce the containment pressure and temperature and to reduce the concentration of fission products in the containment atmosphere during a DBA.~~ In the recirculation mode of operation, heat is removed from the containment sump water by the decay heat removal coolers. Each train of the Containment Spray System provides adequate spray coverage to meet the system design requirements for containment heat removal.

edit

21

Insert B3.6-36A

The Containment Spray System is actuated ~~automatically~~ by a ~~containment High-High pressure signal coincident with a containment high pressure signal and a low pressure injection signal.~~ An automatic actuation opens the Containment Spray System pump discharge valves and starts the ~~two~~ Containment Spray System pumps. ~~[A manual actuation of the Containment Spray System requires the operator to actuate two separate switches on the main control board to begin the same sequence.]~~

edit

edit

Insert B 3.6-36B

~~Containment Cooling System~~

~~The Containment Cooling System consists of three containment cooling trains connected to a common duct suction header with four vertical return air ducts. Each cooling train is equipped with demisters, cooling coils, and an axial flow fan driven by a two speed water cooled electric motor. Each unit connection (two per unit) to the common header is provided with a backpressure damper for isolation purposes.~~

~~During normal operation, two containment cooling trains are required to operate. The third unit is on standby and isolated from the operating units by means of the backpressure dampers. The swing unit is equipped with a transfer switch. It can be manually placed to either the "A" or "B" power train to operate in case one of the operating units fails. Upon receipt of an emergency signal, the two operating cooling fans running at high speed will~~

15

21

(continued)

<Insert B3.6-36A>

During MODE 1 or 2 the Reactor Building Spray System supports the Spray Additive System function of iodine removal by providing the distribution mechanism. In MODES 3 and 4, sodium hydroxide is not mixed with the spray flow.

<Insert B3.6-36B>

Reactor Building Cooling System

The RB Cooling System during normal operations consists of five (5) chilled water supplied cooling coils each in-line with a fan. Four (4) of these fan and chiller coil circuits have in-line service water cooling coils. During normal operations the service water to these coils is isolated. The post accident configuration of the Reactor Building Cooling System consists of the four service water cooling coils and their respective axial flow fans and dampers arranged as two independent trains.

Upon receipt of an Engineered Safeguards Actuation System (ESAS) RB high pressure signal, the four (4) fans associated with the service water coils receive a start signal, the chilled water is isolated, the service water supply and discharge valves open, the RB cooler bypass dampers open (which causes the return air to bypass the chilled water coils) and the RB cooler backdraft dampers receive an open signal. This equipment is powered from class 1E electrical power.

Each of the four (4) service water coil and fan air paths receives return air separately and directly from the RB atmosphere and discharges through ducting to a common plenum for distribution to the various reactor building spaces. The four (4) fans are mounted vertically on the ventilation units and are axial-flow type. The fan motors are single speed and operate in post-accident conditions at the same speed as normal conditions. Reducing fan motor speed during accident conditions is not required due to the reduced suction pressure drop (and hence fan load relative to normal conditions) created by bypassing the chilled water coils. An RB cooling train consists of two coolers and their associated fans which have sufficient capacity to meet post accident heat removal requirements. Conservatively each reactor building emergency cooling train consists of two fans powered from the same emergency bus and their associated coils, but other combinations may be justified by an engineering evaluation (Ref. 2). The continuous availability of appropriate service water flow to the RB Cooling System is assured by the periodic addition of a biocide to the Service Water System.

BASES

BACKGROUND

Containment Cooling System (continued)

~~automatically stop. The two cooling unit fans connected to the ESF buses will automatically restart and run at low speed, provided normal or emergency power is available.~~

~~In post accident operation following an actuation signal, the Containment Cooling System fans are designed to start automatically in slow speed if they are not already running. If they are running at high (normal) speed, the fans automatically stop and restart in slow speed. The fans are operated at the lower speed during accident conditions to prevent motor overload from the higher density atmosphere.~~

APPLICABLE SAFETY ANALYSES

The Containment Spray System and Containment Cooling System limit the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break. The postulated DBAs are analyzed, with regard to containment ESF systems, assuming the loss of one ESF bus. This is the worst-case single active failure, resulting in one train of the Containment Spray System and one train of the Containment Cooling System being inoperable.

The analysis and evaluation show that, under the worst-case scenario, the highest peak containment pressure is [53.8] psig (experienced during a LOCA). The analysis shows that the peak containment temperature is [276] °F (experienced during a LOCA). Both results are less than the design values. (See the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5, "Containment Air Temperature," for a detailed discussion.) The analyses and evaluations assume a power level of [2568] MWt, one containment spray train and one containment cooling train operating, and initial (pre-accident) conditions of [120] °F and [17.7] psia. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

~~The effect of an inadvertent containment spray actuation has been analyzed. An inadvertent spray actuation results in a [2.5] psig containment pressure drop and is associated with the sudden cooling effect in the interior of the leak tight~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~containment. Additional discussion is provided in the Bases for LCO 3.6.4.~~

~~The modeled Containment Spray System actuation from the containment analyses is based on a response time associated with exceeding the containment pressure High-High setpoint coincident with a high pressure injection signal to achieve full flow through the containment spray nozzles. The~~ Containment Spray System total response time of [56] seconds includes diesel generator (DG) startup (for loss of offsite power), block loading of equipment, containment spray pump startup, and spray line filling (Ref. 2).

Containment cooling train performance for post accident conditions is given in Reference 3. The result of the analysis is that each train can provide 33% of the required peak cooling capacity during the post accident condition. The train post accident cooling capacity under varying containment ambient conditions, required to perform the accident analyses, is also shown in Reference 3.

<Insert B3.6-38A>

~~The modeled Containment Cooling System actuation from the containment analysis is based on a response time associated with exceeding the containment pressure high setpoint to achieve full Containment Cooling System air and safety grade cooling water flow. The Containment Cooling System total response time of [28] seconds includes signal delay, DG startup (for loss of offsite power), and service water pump startup times (Ref. 3).~~

~~The Containment Spray System and the Containment Cooling System satisfy Criterion 3 of the NRC Policy Statement.~~

<INSERT B3.3-38C>

LCO

~~During a DBA, a minimum of one containment cooling train and one containment spray train are required to maintain the containment peak pressure and temperature below the design limits. Additionally, one containment spray train is required to remove iodine from the containment atmosphere and maintain concentrations below those assumed in the safety analysis. To ensure that these requirements are met, two containment spray trains and two containment cooling units must be OPERABLE. Therefore, in the event of an accident, the minimum requirements are met, assuming the worst-case single active failure occurs.~~

<Insert B3.6-38B>

(continued)

<Insert B3.6-38A>

... conservatively envelopes signal delay, DG startup, block loading of equipment, fan startup, and service water pump startup times

<Insert B3.6-38B>

In MODE 3 or 4, one reactor building spray train and one reactor building cooling train are required to be OPERABLE. The LCO is provided with a Note which clarifies this requirement.

<Insert B3.6-38C>

In MODES 3 and 4, the Reactor Building Spray System and Reactor Building Cooling System satisfy Criterion 4 of 10 CFR 50.36.

BASES

LCO (continued)

The Each Containment Spray System typically includes a spray pump, spray headers, nozzles, valves, piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the BWST upon an Engineered Safety Features Actuation System signal and manually transferring suction to the containment sump. (15) (5) Safeguards edit

The Each Containment Cooling System typically includes demisters, cooling coils, dampers, an axial flow fan driven by a two speed water cooled electrical motor, instruments, and controls to ensure an OPERABLE flow path. (15) (5) Single Speed fan motors, edit

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment and an increase in containment pressure and temperature requiring the operation of the containment spray trains and containment cooling trains. (34) (Insert B3.6-39A)

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Thus, the Containment Spray System and the Containment Cooling System are not required to be OPERABLE in MODES 5 and 6.

ACTIONS

A.1

With one containment spray train inoperable, in MODE 1 or 2 the inoperable containment spray train must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE spray and cooling trains are adequate to support perform the iodine removal and containment cooling functions. The 72 hour Completion Time takes into account the redundant heat removal capability afforded by the the Containment Spray System train OPERABLE, reasonable time for repairs, and the low probability of a DBA occurring during this period. (21) (25) and iodine

The 10 day portion of the Completion Time for Required Action A.1 is based upon engineering judgment. It takes into account the low probability of coincident entry into two Conditions in this LCO coupled with the low probability of an accident occurring during this time. Refer to Section 1.3, Completion Times, for a more detailed edit

(continued)

36.5-10

<Insert B3.6-39A>

In MODES 1, 2, 3 and 4, the reactor building OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in the lower MODES would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in the lower MODES. Since an event could cause a release of radioactive material in the reactor building as well as a temperature and pressure rise, the Reactor Building Spray System and the Reactor Building Cooling System are required to be OPERABLE in MODES 1, 2, 3, and 4.

BASES

ACTIONS

A.1 (continued)

discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

If the Required Actions and associated Completion Times are not met

D

~~B.1 and B.2~~

~~If the inoperable containment spray train cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach MODE 5 allows additional time to attempt restoration of the containment spray train and is reasonable when considering the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3.~~

unit unit unit is unit 2L edit

B

~~C.1~~

remaining OPERABLE

With one of the ~~required~~ containment cooling trains inoperable, the inoperable containment cooling train must be restored to OPERABLE status within 7 days. The components ~~in this degraded condition provide iodine removal capabilities and~~ are capable of providing at least 100% of the heat removal needs after an accident. The 7 day Completion Time was developed taking into account the redundant heat removal capabilities afforded by combinations of the Containment Spray System and Containment Cooling System and the low probability of a DBA occurring during this period.

in MODE 1 or 2

edit 2L 25

B

The 10 day portion of the Completion Time for Required Action ~~C~~.1 is based ~~upon engineering judgment. It takes into account~~ the low probability of coincident entry into two Conditions in this LCO coupled with the low probability of an accident occurring during this time. Refer to Section 1.3 for a more detailed discussion of the purpose of the "from discovery of failure to meet the LCO" portion of the Completion Time.

edit

3.6.5-10

3.6.5-5

3.6.5-10

(continued)

BASES

ACTIONS (continued)

C ~~B~~.1

in MODE 1 or 2

With two of the ~~required~~ containment cooling trains inoperable, one of the ~~required~~ containment cooling trains must be restored to OPERABLE status within 72 hours. The components ~~in this degraded condition~~ (both spray trains are OPERABLE or else Condition ~~F~~ is entered) ~~provide~~ iodine removal capabilities and are capable of providing at least 100% of the heat removal needs after an accident. The 72 hour Completion Time ~~was developed~~ taking into account the redundant heat removal capabilities afforded by ~~combinations of~~ the Containment Spray System ~~and Containment Cooling System~~ and the low probability of a DBA occurring during this period.

Remaining Spray System

G

edit
edit
support
edit

Insert B3.6-41A

F ~~B~~.1 ~~and E.2~~

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

E

edit
edit
edit

G ~~B~~.1

inoperable in MODE 1 or 2

With two containment spray trains or any combination of three or more containment spray and containment cooling trains inoperable, the unit is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be entered immediately.

Insert B3.6-41B

SURVEILLANCE REQUIREMENTS

SR 3.6.~~B~~.1 5

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in

(continued)

<Insert B3.6-41A>

E.1

With either one required reactor building (RB) spray train or one required reactor building cooling train inoperable in MODE 3 or 4, the inoperable train must be restored to OPERABLE status in 36 hours. The 36 hour Completion Time is reasonable based on consideration of the cooling capacity of the remaining required train of RB cooling or RB spray, the reduced reactor coolant energy in these MODES and the short time spent in these MODES.

<Insert B3.6-41B>

... in MODE 1 or 2, or one required reactor building spray train and one required reactor building cooling train inoperable in MODE 3 or 4, then LCO 3.0.3 must be entered immediately.

The first part of this Condition addresses the loss of Spray Additive System support which would result from two inoperable reactor building spray trains in MODE 1 or 2. The second part of this Condition considers the loss of adequate reactor building cooling capacity in MODE 1 or 2 which would result from the loss of three or more of the four RB spray and RB cooling trains. Finally, the third part of this Condition addresses loss of reactor building cooling capability in MODES 3 and 4 when only one train of RB spray and one train of RB cooling are required.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.~~6~~5.1 (continued)

position, since these were 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 SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment and capable of potentially being mispositioned are in the correct position.

or control room indication

SR 3.6.~~6~~5.2

Operating each [required] containment cooling train fan unit for ≥ 15 minutes ensures that all trains are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. The 31 day Frequency was developed considering the known reliability of the fan units and controls, the two train redundancy available, and the low probability of a significant degradation of the containment cooling trains occurring between surveillances and has been shown to be acceptable through operating experience.

This SR is performed by starting (unless operating) each operational cooling fan from the control room.

3.6.5-9

SR 3.6.~~6~~5.3

Verifying that each [required] containment cooling train provides an essential raw water cooling flow rate of ≥ [1780] gpm to each cooling unit provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 1). The Frequency was developed considering the known reliability of the Cooling Water System, the two train redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

Insert B3.6-42A

SR 3.6.~~6~~5.4

Verifying that each required containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential

Acceptable performance will be indicated if the pump starts, operates for fifteen minutes, and the discharge pressure and flow rate are within ±10% of a point on the pump head curve.

3.6.5-3

(continued)

<Insert B3.6-42A>

Verifying that a service water flow rate of 1200 gpm is provided to each required reactor building cooling train provides assurance that the original design flow rate is being achieved and that the service water flow rate is not degrading (Ref. 3). Assurance that the flow doesn't degrade by biological fouling between surveillances is provided by the addition of a biocide to the Service Water System whenever the service water temperature is between 60°F and 80°F.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.4 (continued)

3.6.5-3

The discharge pressure and flow are within ± 10% of a point on the pump head curve

pressure are measured during normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 3). Since the Containment Spray System pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. 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 may detect incipient failures by indicating abnormal performance. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.5.5 and SR 3.6.5.6

3.6.5-7 AN0-73

<INSERT B36-43A>

<INSERT B36-43B>

These SRs require verification that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated 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 these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.5.7

by control board indication

This SR requires verification that each required containment cooling train actuates upon receipt of an actual or simulated actuation signal. The 18 month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.5.5 and SR 3.6.5.6, above, for further discussion of the basis for the 18 month Frequency.

(continued)

<Insert B3.6-43A>

ANO-73

The SRs are considered satisfactory if visual observation and control board indication verifies that all components have responded to the actuation signal properly.

<Insert B3.6-43B>

3.6.4-4

During testing of the spray pump, the reactor building isolation valve in the spray line is closed with its breaker open to prevent spraying the reactor building. After spray pump performance is verified, the pump is stopped. Its breaker is racked down to prevent restart. Power is then restored to the reactor building isolation valve for valve testing.

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.~~8~~ 5.8

With the containment spray header isolated and drained of any solution, low pressure air or smoke can be blown through test connections. Performance of this Surveillance demonstrates that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive nature of the design of the nozzles, a test at ~~the first refueling and at~~ 10 year intervals is considered adequate to detect obstruction of the spray nozzles.

edit

REFERENCES

SAR, Section 1.4

1. ~~10 CFR 50, Appendix A, GDC 38, GDC 39, GDC 40, GDC 41, GDC 42, and GDC 43.~~ 27

Chapter 14

2. 3. ~~FSAR, Section [14.1].~~ edit

Chapter 6

3. 2. ~~FSAR, Section [6.3].~~ 15

4. ~~FSAR, Section [14.2].~~ edit

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

4. 10CFR50.36. 28

REACTOR BUILDING

B 3.6 CONTAINMENT SYSTEMS

B 3.6.7 6 Spray Additive System

BASES

BACKGROUND

The Spray Additive System is a subsystem of the Containment Spray System that assists in reducing the iodine fission product inventory in the containment atmosphere resulting from a Design Basis Accident (DBA).

Reduces

edit 16

Insert B3.6-45A

The Containment Spray System and Spray Additive System perform no function during normal operations. In the event of an accident such as a loss of coolant accident (LOCA), however, the Spray Additive System will be automatically actuated upon a high containment pressure signal by the Engineered Safety Features Actuation System.

high-

Safeguards

edit edit 14

Radioiodine in its various forms is the fission product of primary concern in the evaluation of a DBA. It is absorbed by the spray from the containment atmosphere. To enhance the iodine absorption capacity of the spray, the spray solution is adjusted to an alkaline pH that promotes iodine hydrolysis, in which iodine is converted to nonvolatile forms. Sodium hydroxide (NaOH), because of its stability when exposed to radiation and elevated temperature, is the preferred spray additive.

the dose consequences of an accident

edit edit

Insert B3.6-45B

a sprayed solution

Insert B3.6-45C

utilized

edit 23

NaOH

The spray additive tank is designed and located to permit gravity draining into the Containment Spray System. Both Containment Spray System pumps initially take suction from the borated water storage tank (BWST) via two independent flow paths. The spray additive tank has a common header that splits and feeds each of the Containment Spray System suction lines. The system is designed to inject at a rate commensurate with the draining rate of the BWST so that all borated water injected is mixed with NaOH.

16

outlet

discharge 16

NaOH

The flow rate is proportioned to provide a spray solution with a pH between 7.2 and 11.0 (Ref. 1). This range of alkalinity was established not only to aid in removal of airborne iodine, but also to minimize the corrosion of mechanical system components that would occur if the acidic borated water were not buffered. The pH range also considers the environmental qualification of equipment in containment that may be subjected to the spray.

The

edit

which is alkaline

(continued)

<Insert B3.6-45A>

The Reactor Building Spray System supports the Spray Additive System iodine removal function by providing a distribution mechanism for the solution.

<Insert B3.6-45B>

3.6.6-1

Actuation of the Spray Additive System opens the sodium hydroxide isolation valves which are powered from independent buses. When the valves are open, the sodium hydroxide solution is ready to be introduced into the RB Spray System headers.

<Insert B3.6-45C>

The sodium hydroxide volume requirement is given in gallons for compatibility with the design analyses. The minimum NaOH tank volume of 9000 gallons preserves the required NaOH solution contribution from the tank to the post-LOCA minimum sump level.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The containment Spray Additive System is essential to the effective removal of airborne iodine within containment following a DBA.

reactor building
provides for
the reactor building
L6
1
edit
edit

Following the assumed release of radioactive materials into containment, the containment is assumed to leak at its design value following the accident. The analysis assumes that most of the containment volume is covered by the spray.

The DBA response time assumed for the Spray Additive System is the same as for the Containment Spray System and is discussed in the Bases for LCO 3.6.5, "Containment Spray and Cooling Systems."

delay
5
edit
edit

The DBA analyses assume that one train of the Containment Spray System/Spray Additive System is inoperable and that the entire spray additive tank volume is added to the remaining Containment Spray System flow path.

LOCA
Sufficient NaOH
L6
9
edit

In the evaluation of the worst-case LOCA, the safety analysis assumed that an alkaline containment spray effectively reduced the airborne iodine.

Each Containment Spray System suction line is equipped with its own gravity feed from the spray additive tank. Therefore, in the event of a single failure within the Spray Additive System (i.e., suction valve failure), NaOH will still be mixed with the borated water, establishing the alkalinity essential to effective iodine removal.

NaOH
NaOH isolation
9
L6

The Spray Additive System satisfies Criterion 3 of the NRC Policy Statement.

Provide
10 CFR 50.36 (Ref. 2)
28

LCO

The Spray Additive System is necessary to reduce the release of radioactive material to the environment in the event of a DBA. To be considered OPERABLE, the volume and concentration of the spray additive solution must be sufficient to provide NaOH injection into the spray flow until the Containment Spray System suction path is switched from the BWST to the containment sump and to raise the average spray solution pH to a level conducive to iodine removal. The average spray solution pH is between 7.2 and 11.0. This pH range maximizes the effectiveness of the iodine removal mechanism without introducing

NaOH
9
edit
16
long term Sump
in the alkaline range
edit
edit
edit
retention

(continued)

BASES

LCO (continued)

conditions that may induce caustic stress corrosion cracking of mechanical system components. In addition, it is essential that valves in the Spray Additive System flow paths are properly positioned and that automatic valves are capable of activating to their correct positions.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment requiring the operation of the Spray Additive System. The Spray Additive System assists in reducing the iodine fission product inventory prior to release to the environment. 34 21 1

Insert B3.6-47A

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Thus, the Spray Additive System is not required to be OPERABLE in MODES 5 and 6. 21

ACTIONS

A.1

Capability

or non-existant

spray solution

With the containment Spray Additive System inoperable, the system must be restored to OPERABLE status within 72 hours. The pH adjustment of the Containment Spray System for corrosion protection and iodine removal enhancement is reduced in this Condition. The Containment Spray System would still be available and would remove some iodine from the containment atmosphere in the event of a DBA. The 72 hour Completion Time takes into account the redundant flow path capabilities and the low probability of the worst-case DBA occurring during this period. 16 edit LOCA edit

If the Required Actions and associated Completion Times are not met

B.1 and B.2 21

If the Spray Additive System cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. The extended interval to reach unit edit 21

is

(continued)

<Insert B3.6-47A>

In MODES 1 and 2, the reactor building OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in a lower MODE would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in a lower MODE. Although the core is designed to retain structural integrity during an accident, fuel failure with resultant radioactive material release is postulated and the Spray Additive System is required OPERABLE in MODES 1 and 2.

In MODES 3 and 4, there is no postulated fuel failure contribution to radioactive material release and significantly less need for iodine removal capacity. Also, because of the limited time spent in these MODES, the probability of an event requiring use of the Spray Additive System is low. Therefore, the Spray Additive System is not required to be OPERABLE in MODE 3 or 4.

BASES

ACTIONS

B.1 and B.2 (continued)

MODE 5 allows additional time for restoration of the Spray Additive System and is reasonable when considering that the driving force for a release of radioactive material from the Reactor Coolant System is reduced in MODE 3. 21

SURVEILLANCE REQUIREMENTS

SR 3.6.7.1 6

Verifying the correct alignment of spray additive manual, power operated, and automatic valves in the spray additive System 9 flow path provides assurance that the system is able to provide additive to the Containment Spray System in the event of a DBA. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment capable of potentially being mispositioned are in the correct position.

Reactor Building

NaOH

or control room indication

edit 9 1 9

SR 3.6.7.2 6

To provide effective iodine removal, the containment spray must be an alkaline solution. Since the BWST contents are normally acidic, the volume of the spray additive tank must provide a sufficient volume of spray additive to adjust pH for all water injected. This SR is performed to verify the availability of sufficient NaOH solution in the Spray Additive System. The 184 day Frequency is based on the low probability of an undetected change in tank volume occurring during the SR interval (the tank is isolated during normal unit operations). Tank level is also indicated and alarmed in the control room, such that there is a high confidence that a substantial change in level would be detected.

NaOH

should

the most

edit

edit 9

16 23

Insert B3.6-48A

(continued)

<Insert B3.6-48A>

The NaOH tank solution minimum volume of 9000 gallons corresponds to a tank level of approximately 26 feet at a temperature of 77°F and a NaOH concentration of 5.0 wt%. This parameter does not contain an allowance for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures. The minimum NaOH tank volume preserves the required NaOH solution contribution from the tank to the post-LOCA minimum sump level.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.~~7~~ 6.3

This SR provides verification of the NaOH concentration in the ~~spray additive~~ tank and is sufficient to ensure that the spray solution being injected into ~~containment~~ the reactor building is at the correct pH level. The concentration of NaOH in the ~~spray additive~~ tank must be determined by chemical analysis. The 184 day Frequency is sufficient to ensure that the concentration ~~level~~ of NaOH in the ~~spray additive~~ tank remains within the established limits. This is based on the low likelihood of an uncontrolled change in concentration (the tank is normally isolated) and the probability that any substantial variance in tank volume will be detected.

NaOH

There is no instrument uncertainty included in the Surveillance limit values. Additional allowances for instrument uncertainty are contained in the implementing procedures.

SR 3.6.~~7~~ 6.4

This SR provides verification that each automatic valve in the Spray Additive System flow path actuates to its correct position. The [~~18~~] 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a ~~plant~~ unit outage and the potential for an unplanned on 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~~] 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

~~SR 3.6.7.5~~

~~To ensure that the correct pH level is established in the borated water solution provided by the Containment Spray System, the flow [rate] in the Spray Additive System is verified once per 5 years. This SR provides assurance that the correct amount of NaOH will be metered into the flow path upon Containment Spray System initiation. Due to the passive nature of the spray additive flow controls, the 5 year Frequency is sufficient to identify component degradation that may affect flow [rate].~~

REFERENCES

1. ~~FSAR, Section 6.2.1~~ Chapter 6

2. 10CFR 50.36.

REACTOR BUILDING

B 3.6 CONTAINMENT SYSTEMS

B 3.6 8 7 Hydrogen Recombiners

BASES

BACKGROUND

Permanently installed hydrogen recombiners are required to reduce the hydrogen concentration in the containment following a loss of coolant accident (LOCA) or steam line break (SLB). The recombiners accomplish this by recombining hydrogen and oxygen to form water vapor. The vapor 18 returned to the containment, thus eliminating any discharge to the environment. The hydrogen recombiners are manually initiated since flammability limits would not be reached until several days after a Design Basis Accident (DBA).

remains in

LOCA

Two 100% capacity independent hydrogen recombiners are provided. Each consists of controls located in the control room, a power supply, and a recombiner located external to containment. The recombiners have no moving parts. Recombination is accomplished by heating a hydrogen air mixture above 1150°F. The resulting water vapor and discharge gases are cooled prior to discharge from the recombiner. Air flows through the unit at approximately 3000 cfm at a maximum supply temperature of 120°F. A single recombiner is capable of maintaining the hydrogen concentration in containment below the 4.0 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safety Features bus and is provided with a separate power panel and control panel.

located in the auxillary building

in the reactor building

100 scfm

Safeguards (ES)

APPLICABLE SAFETY ANALYSES

The hydrogen recombiners provide for the capability of controlling the bulk hydrogen concentration in containment to less than a concentration of 4.0 v/o following a DBA. This Control would prevent a hydrogen burn inside containment, thus ensuring the pressure and temperature assumed in the accident analysis are not exceeded. The limiting DBA relative to hydrogen generation is a LOCA.

reactor building

Hydrogen may accumulate within containment following a LOCA as a result of:

(continued)

edit

BASES

APPLICABLE SAFETY ANALYSES (continued)

a. A metal steam reaction between the zirconium fuel rod cladding and the reactor coolant;

b. Radiolytic decomposition of water in the Reactor Coolant System (RCS) and the ~~containment~~ reactor building sump; 1

c. Hydrogen in the RCS at the time of the LOCA (i.e., hydrogen dissolved in the reactor coolant and hydrogen gas in the pressurizer vapor space); or

d. Corrosion of metals exposed to Containment Spray System and Emergency Core Cooling Systems solutions.

edit

To evaluate the potential for hydrogen accumulation in containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident ~~is~~ has been evaluated. Conservative assumptions ~~recommended~~ Presented edit by Reference 1 are used to maximize the amount of hydrogen calculated. These evaluations demonstrate approximately ~~10~~ 8.9 days are needed for hydrogen concentration to increase to ~~4.0~~ 4.1 v/o post LOCA without recombiner operation 17

and 2

The hydrogen recombiners satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10CFR 50.36 (Ref. 3). 28

LCO

Two hydrogen recombiners must be OPERABLE. This ensures operation of at least one hydrogen recombiner in the event of a worst-case single active failure.

Operation with at least one hydrogen recombiner ensures that the post LOCA hydrogen concentration can be prevented from exceeding the flammability limit.

APPLICABILITY

<Insert B3.6-51A>

~~In MODES 1 and 2, two hydrogen recombiners are required to control the hydrogen concentration within containment below its flammability limit of 4.1 v/o following a LOCA, assuming a worst-case single failure.~~ 17 34

In MODES 3 and 4, both the hydrogen production rate and the total hydrogen produced ~~after a LOCA~~ would be less than that calculated for the DBA LOCA. Also, because of the limited time in these MODES, the probability of an ~~accident~~ event edit edit

(continued)

<Insert B3.6-51A>

In MODES 1 and 2 the hydrogen recombiner OPERABILITY for the limiting Design Basis Accidents is based on full power operation. Although reduced power in a lower MODE would not require the same level of accident mitigation performance, there are no accident analyses for reduced performance in a lower MODE. Two hydrogen recombiners are required OPERABLE in MODES 1 and 2 to assure control of hydrogen concentration within the reactor building to less than the flammability limit of 4 v/o.

edit

BASES

APPLICABILITY (continued)

requiring the hydrogen recombiners is low. Therefore, the hydrogen recombiners are not required in MODE 3 or 4.

In MODES 5 and 6, the probability and consequences of a LOCA are low, due to the pressure and temperature limitations. Therefore, hydrogen recombiners are not required in these MODES.

ACTIONS

A.1

With one hydrogen recombiner inoperable, the inoperable recombiner must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE recombiner is adequate to perform the hydrogen control function. However, the overall reliability is reduced because a single failure in the OPERABLE recombiner could result in a reduced hydrogen control capability. The 30 day Completion Time is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit. 17

Required Action A.1 has been modified by a Note stating that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one hydrogen recombiner is inoperable. This allowance is based on the availability of the other hydrogen recombiner, the small probability of a LOCA or SLB occurring (that would generate an amount of hydrogen that exceeds the flammability limit), and the amount of time available after a LOCA or SLB (should one occur) for operator action to prevent hydrogen accumulation from exceeding the flammability limit. 17

~~B.1 and B.2~~

~~Reviewer's Note: This Condition is only allowed for units with an alternate hydrogen control system acceptable to the technical staff.~~ 10

(continued)

edit

BASES

ACTIONS

B.1 and B.2 (continued)

With two hydrogen recombiners inoperable, the ability to perform the hydrogen control function via alternate capabilities must be verified by administrative means within 1 hour. The alternate hydrogen control capabilities are provided by [the containment Hydrogen Purge System/hydrogen recombiner/Hydrogen Ignitor System/Hydrogen Mixing System/ Containment Air Dilution System/Containment Inerting System]. The 1 hour Completion Time allows a reasonable period of time to verify that a loss of hydrogen control function does not exist. [Reviewer's Note: The following is to be used if a non-Technical Specification alternate hydrogen control function is used to justify this Condition: In addition, the alternate hydrogen control system capability must be verified every 12 hours thereafter to ensure its continued availability.] [Both] the [initial] verification [and all subsequent verifications] may be performed as an administrative check, by examining logs or other information to determine the availability of the alternate hydrogen control system. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of the alternate hydrogen control system. If the ability to perform the hydrogen control function is maintained, continued operation is permitted with two hydrogen recombiners inoperable for up to 7 days. Seven days is a reasonable time to allow two hydrogen recombiners to be inoperable because the hydrogen control function is maintained and because of the low probability of the occurrence of a LOCA that would generate hydrogen in the amounts capable of exceeding the flammability limit.

10

If the Required Actions and associated Completion Times are not met

B.1

If the inoperable hydrogen recombiner(s) cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

Unit

edit

edit

(continued)

edit

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.~~8~~ 7.1

edit

Performance of a system functional test for each hydrogen recombiner ensures that the recombiners are operational and can obtain and sustain the temperature necessary for hydrogen recombination. In particular, this SR requires verification that the minimum heater sheath temperature increases to ≥ 700°F in ≤ 90 minutes. After reaching 700°F, the power is increased to maximum for approximately 2 minutes and power verified to be ≥ 60 kW. Operating experience has shown that these components usually pass the Surveillance when performed at the ~~[18]~~ 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.~~8~~ 7.2

edit

This SR ensures that there are no physical problems that could affect recombiner operation. Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failures involve loss of power, blockage of the internal flow path, missile impact, etc. A visual inspection is sufficient to determine abnormal conditions that could cause such failures. The ~~[18]~~ 18 month Frequency for this SR was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

edit

SR 3.6.~~8~~ 7.3

edit

3.6.7-1

This SR requires performance of a resistance to ground test, following the performance of SR 3.6.7.2, for each heater phase to ensure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance to ground for any heater phase is ≥ 10,000 ohms. The ~~[18]~~ 18 month Frequency for this SR was developed considering the incidence of hydrogen recombiners failing the SR in the past is low.

REFERENCES

1. SAR, Section 6.6. 17

2. ~~1.~~ Regulatory Guide 1.7, Revision ~~1~~ 2. edit

3. 10CFR 50.36. 28