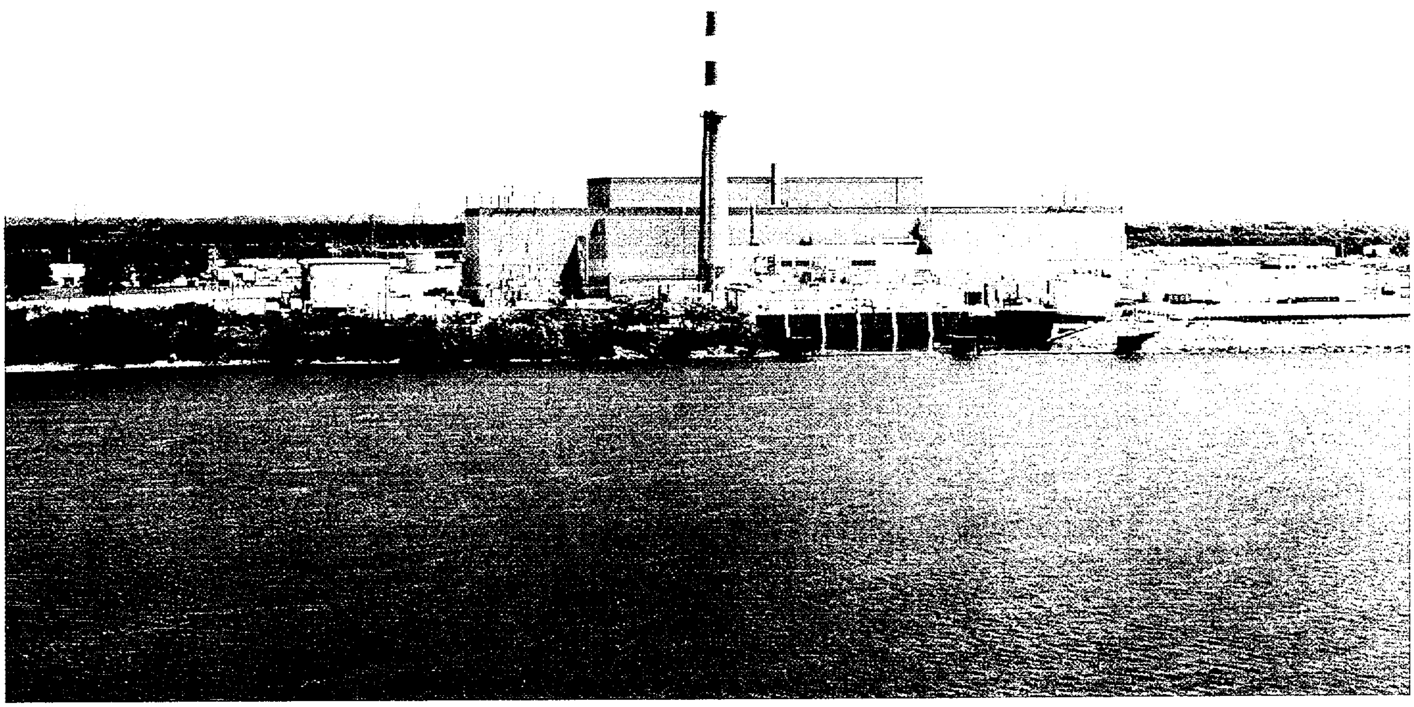
Improved Technical Specifications



Quad Cities Station

Volume 7:
Section 3.6; ISTS/JFDs, ISTS Bases/JFDs, and NSHC & EA

ComEd

ε<CTS>

3.6 CONTAINMENT SYSTEMS

3.6.1.1 Primary Containment

<3.7.A>
<3.7.K.3>

LCO 3.6.1.1 Primary containment shall be OPERABLE.

<Appl 3.7.A>

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

<3.7.A Act>
<Doc L.1>
<3.7.A Act>

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Primary containment inoperable.	A.1 Restore primary containment 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 4.	12 hours 36 hours

<CTS>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<4.7.A.1> SR 3.6.1.1.1	Perform required visual examinations and leakage rate testing except for primary containment air lock testing, in accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~ the Primary Containment Leakage Rate Testing Program. ~~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 Primary Containment Leakage Rate Testing Program [1]
<4.7.K.5> SR 3.6.1.1.2	~~Verify drywell to suppression chamber differential pressure does not decrease at a rate > [0.25] inch water gauge per minute tested over a [10] minute period at an initial differential pressure of [1] psid.~~ Verify drywell-to-suppression chamber bypass leakage is less than or equal to the bypass leakage limit. However, during the first unit startup following bypass leakage testing performed in accordance with this SR, the acceptance criterion ≤ 2% of the drywell-to-suppression chamber leakage limit. [2]	~~[18]~~ 24 months [3] ~~AND -----NOTE----- Only required after two consecutive tests fail and continues until two consecutive tests pass ----- [9 months]~~ [2]

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.6.1.1 - PRIMARY CONTAINMENT

1. A 10 CFR 50 Appendix J Testing Program Plan has been added to Section 5.5. The program references the requirements of 10 CFR 50 Appendix J and approved exemptions, therefore, the surveillances have been modified to reference the program. This is consistent with Current Licensing Basis and with TSTF-52.

2. The words of ITS SR 3.6.1.1.2 are essentially consistent with the BWR/6 ISTS (NUREG-1434) SR 3.6.5.1.1. The changes to the current licensing basis requirements are justified in the Discussion of Changes for ITS 3.6.1.1. This deviation from NUREG-1433, Revision 1 will help ensure consistency between the Technical Specifications of the ComEd Boiling Water Reactors.

3. The brackets have been removed and the proper plant specific values have been included.

<CTS>

3.6 CONTAINMENT SYSTEMS

3.6.1.2 Primary Containment Air Lock

<3.7.C> LCO 3.6.1.2 The primary containment air lock shall be OPERABLE.

<App 3.7.C> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

NOTES

<DOC L.1> <3.7.C footnote b> 1. Entry and exit is permissible to perform repairs of the air lock components.

<DOC A.3> 2. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when air lock leakage results in exceeding overall containment leakage rate acceptance criteria.

<3.7.C Act 1.a> <DOC A.4> <DOC L.2> <3.7.C footnote b>

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One primary containment air lock door inoperable.	NOTES 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls.	
	A.1 Verify the OPERABLE door is closed.	1 hour
	AND	

(continued)

L<CTS>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Lock the OPERABLE door closed.	24 hours
	AND	
	A.3 --------NOTE-------- Air lock doors in high radiation areas or areas with limited access due to inerting may be verified locked closed by administrative means. ------------------	
	Verify the OPERABLE door is locked closed.	Once per 31 days
B. Primary 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 air lock are inoperable and Condition C is entered. 2. Entry into and exit from primary containment is permissible under the control of a dedicated individual. ------------------------	1
	B.1 Verify an OPERABLE door is closed.	1 hour
	AND	
		(continued)

<3.7.C Act 1.a> <DOC L.3> <3.7.C Act 1.b> <DOC A.4> <3.7.C Act 2> <DOC M.4>

L <CTS>

ACTIONS

<3.7.C Act 2> <DOC M.1> <DOC L.3>

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Lock an OPERABLE door closed.	24 hours
	<u>AND</u>		
	B.3	---------NOTE--------- Air lock doors in high radiation areas or areas with limited access due to inerting may be verified locked closed by administrative means. ---------------------	
		Verify an OPERABLE door is locked closed.	Once per 31 days
C. Primary containment air lock inoperable for reasons other than Condition A or B.	C.1	Initiate action to evaluate primary containment overall leakage rate per LCO 3.6.1.1, using current air lock test results.	Immediately
	<u>AND</u>		
	C.2	Verify a door is closed.	1 hour
	<u>AND</u>		
	C.3	Restore air lock to OPERABLE status.	24 hours

<DOC A.3> <3.7.C Act 3>

(continued)

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
⟨Doc A.7⟩ ⟨3.7.C Act 2⟩ ⟨3.7.C Act 3⟩	D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 36 hours

<CTS>

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE (5)	FREQUENCY
<4.7.C.1> <3.7.C footnote c+d> SR 3.6.1.2.1	------NOTE------ [2] (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 3.6.1.1.1. [2] ----------------- Perform required primary containment air lock leakage rate testing in accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~ [3]. ~~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 ≤ [0.01 L_a] when the gap between the door seals is pressurized to [≥ 10 psig for at least 15 minutes].~~ [3]	~~------NOTE------ SR 3.0.2 is not applicable~~ [3] ------------ In accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~ the Primary Containment Leakage Rate Testing Program
<4.7.C.2> SR 3.6.1.2.2	~~------NOTE------ Only required to be performed upon entry into primary containment air lock when the primary containment is de-inerted.~~ TSTF-17 ----------------- Verify only one door in the primary containment air lock can be opened at a time.	~~184 days~~ 24 months

1. The word "primary" has been added for clarity and consistency.

2. An additional Note has been added to ITS SR 3.6.1.2.1 for clarity. This Note is consistent with the BWR/6 ISTS, NUREG-1434, Rev. 1.

3. The Primary Containment Leakage Rate Testing Program Plan is included in CTS 6.8.D.5 and in proposed ITS 5.5.12. The Program references the requirements of 10 CFR 50 Appendix J and approved exemptions, therefore, the Surveillances have been modified to reference the program. In addition, this is also consistent with the Current Licensing Basis and with TSTF-52.

<CTS>

3.6 CONTAINMENT SYSTEMS

3.6.1.3 Primary Containment Isolation Valves (PCIVs)

<3.7.D> <3.6.M> LCO 3.6.1.3 Each PCIV, except reactor building-to-suppression chamber vacuum breakers, shall be OPERABLE.

<Appl 3.7.D> <Appl 3.6.M> <DOC M.1> APPLICABILITY: MODES 1, 2, and 3,
When associated instrumentation is required to be OPERABLE per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation."

ACTIONS

NOTES [1]

<3.7.D footnote (a)> <DOC L.4> 1. Penetration flow paths (except for purge valve penetration flow paths) may be unisolated intermittently under administrative controls.

<DOC A.3> 2. Separate Condition entry is allowed for each penetration flow path.

<DOC A.4> <3.7.D Act 2.b> 3. Enter applicable Conditions and Required Actions for systems made inoperable by PCIVs.

<DOC A.4> 4. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when PCIV leakage results in exceeding overall containment leakage rate acceptance criteria in MODES 1, 2, and 3. [2]

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.7.D Act> <3.6.M Act> A. NOTE: Only applicable to penetration flow paths with two or more [3] PCIVs. One or more penetration flow paths with one PCIV inoperable {except for purge valve leakage not within limit}. [4] due to main steam line isolation (MSIV) rate	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	4 hours except for main steam line AND 8 hours for main steam line (continued)

<CTS>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<4.7.A.2> <4.7.A.2 footnote b> <Doc L.10> A. (continued) TSTF-269	A.2 ----NOTE---- 5 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 primary containment **AND** Prior to entering MODE 2 or 3 from MODE 4, if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days, for isolation devices inside primary containment

(continued)

<CTS>

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<Doc L.3>	B. ---------NOTE--------- Only applicable to penetration flow paths with two PCIVs. or more 3 ------------------------- One or more penetration flow paths with two or more PCIVs inoperable ~~except for~~ due to MSIV ~~purge valve~~ leakage rate not within limit~~s~~. 4	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
<3.7.D Act 1> <3.7.D Act 2> <4.7.A.2> <4.7.A.2 footnote 6> <DOC L.10>	C. ---------NOTE--------- Only applicable to penetration flow paths with only one PCIV. ------------------------- One or more penetration flow paths with one PCIV inoperable. TSTF-269	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. AND C.2 ---------NOTE--------- 5 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.	4 ~~[4]~~ hours except for excess flow check valves (EFCVs) and penetrations with a closed system 5 AND 72 ~~12~~ hours for EFCVs and penetrations with a closed system 5 4 Once per 31 days
<3.6.M Act>	[D. One or more penetration flow paths with MSIV ~~Secondary containment bypass~~ leakage rate not within limit. 4	D.1 Restore leakage rate to within limit.	~~4~~ 8 hours] 4

(continued)

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
[E. One or more penetration flow paths with one or more containment purge valves not within purge valve leakage limits.	E.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
	AND		
TSTF-269 changes not shown	E.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 2 or 3 from MODE 4 if not performed within the previous 92 days for isolation devices inside containment]
	AND		

1

(continued)

<CTS>

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
[E. (continued)]	E.3	Perform SR 3.6.1.3.7 for the resilient seal purge valves closed to comply with Required Action E.1.	Once per [92] days] 1
<3.7.D Acts 1&2> <3.6.M Act> E 1 F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met in MODE 1, 2, or 3.	F.1 E.1	Be in MODE 3.	12 hours
	AND E 1		
	F.2 E.2	Be in MODE 4.	36 hours
[G. Required Action and associated Completion Time of Condition A, B, C, D, or E not met for PCIV(s) required to be OPERABLE during movement of irradiated fuel assemblies in [secondary] containment.	G.1	---------NOTE--------- LCO 3.0.3 is not applicable. ------------------ Suspend movement of irradiated fuel assemblies in [secondary] containment.	Immediately] 1
[H. Required Action and associated Completion Time of Condition A, B, C, D, or E not met for PCIV(s) required to be OPERABLE during CORE ALTERATIONS.	H.1	Suspend CORE ALTERATIONS.	Immediately]

(continued)

<CTS>

[1]

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
[1] F. ~~G.~~ Required Action and associated Completion Time of Condition A, B, C, D, ~~or E~~ not met for PCIV(s) required to be OPERABLE during MODE 4 or 5 ~~or during operations with a potential for draining the reactor vessel (OPDRVs)~~. [7] <DOC M.1>	F. ~~G.~~1 [1] Initiate action to suspend operations with a potential for draining the reactor vessel ~~OPDRVs~~. [8] OR F. ~~G.~~2 Initiate action to restore valve(s) to OPERABLE status.	[Immediately Immediately] [9]

SURVEILLANCE REQUIREMENTS [1]

SURVEILLANCE		FREQUENCY
[SR 3.6.1.3.1	[------NOTE------ Only required to be met in MODES 1, 2, and 3. ------] Verify each [18] inch primary containment purge valve is sealed closed except for one purge valve in a penetration flow path while in Condition E of this LCO.	31 days]

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
[SR 3.6.1.3.~~8~~ 1 [1]	---NOTE~~S~~--- ~~1. Only required to be met in MODES 1, 2, and 3.~~ [2] ~~2~~. Not required to be met when the ~~[18]~~ inch primary containment vent and [9] purge valves are open for inerting, de-inerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open, provided the drywell vent and purge valves and their associated suppression chamber vent and purge valves are not open simultaneously [9] ------ Verify each ~~[18]~~ [9] inch primary containment vent and purge valve, except for the torus purge valve [12], is closed.	31 days]
SR 3.6.1.3.~~8~~ 2 [1]	---NOTES--- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. ------ Verify each primary containment isolation manual valve and blind flange that is located outside primary containment, and not locked, sealed, or otherwise secured [TSTF-45], and is required to be closed during accident conditions is closed.	31 days

<DOC M.2>

<4.7.A.2>

<4.7.A.2 footnote b>

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<4.7.D.2 footnote b> SR 3.6.1.3.3 [1] -----NOTES----- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for PCIVs that are open under administrative controls. --------------- Verify each primary containment manual isolation valve and blind flange that is located inside primary containment, and not locked, sealed, or otherwise secured [TSTF-45] and is required to be closed during accident conditions is closed.	Prior to entering MODE 2 or 3 from MODE 4, if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days
<4.7.D.5> <4.7.D.5.a> SR 3.6.1.3.4 [1] Verify continuity of the traversing incore probe (TIP) shear isolation valve explosive charge.	31 days
<4.7.D.3> SR 3.6.1.3.5 [1] Verify the isolation time of each power operated ~~and each~~ automatic PCIV~~s~~ [9], except for MSIVs~~,~~ [9] is within limits. (5) TSTF-46b	[In accordance with the Inservice Testing Program or 92 days] [9]

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
	SR 3.6.1.3.7 ------NOTE------ [Only required to be met in MODES 1, 2 and 3.] ------ Perform leakage rate testing for each primary containment purge valve with resilient seals.	[184 days AND Once within 92 days after opening the valve] 1
<4.6.M>	SR 3.6.1.3.8 6 11 Verify the isolation time of each MSIV is ≥ [2] 3 seconds and ≤ [8] 5 seconds. 9	[In accordance with the Inservice Testing Program or 18 months] 9
<4.7.D.2>	SR 3.6.1.3.9 7 11 Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	[18] 24 months 9
<4.7.D.4>	SR 3.6.1.3.10 8 11 Verify each reactor instrumentation line EFCV actuates ~~[on a simulated instrument line break to restrict flow to ≤ 1 gph]~~. to the isolation position on an actual or simulated instrument line break signal	[18] 24 months 9
<4.7.D.5> <4.7.D.5.b>	SR 3.6.1.3.11 9 11 Remove and test the explosive squib from each shear isolation valve of the TIP System.	[18] 24 months on a STAGGERED TEST BASIS 9

(continued)

L {CTS}

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
[SR 3.6.1.3.12	---NOTES--- [1. Only required to be met in MODES 1, 2, and 3.] 2. Results shall be evaluated against acceptance criteria of SR 3.6.1.1.1 in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions. ------ Verify the combined leakage rate for all secondary containment bypass leakage paths is ≤ [L_a] when pressurized to ≥ [psig].	---NOTE--- SR 3.0.2 is not applicable ------ In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions] 1
SR 3.6.1.3.13 46 10 1	9 Verify the combined leakage rate ~~through each~~ for all MSIV leakage paths is ≤ [11.5] scfh when tested at ≥ [28.0] psig. 25 11	[---NOTE--- SR 3.0.2 is not applicable ------] [In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions] Primary Containment Leakage Rate Testing Program 10 10

(4.7.D.6)

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.14	[-------------------NOTE------------------- Only required to be met in MODES 1, 2, and 3. --] Verify combined leakage rate of [1 gpm times the total number of PCIVs] through hydrostatically tested lines that penetrate the primary containment is not exceeded when these isolation valves are tested at ≥ [63.25] psig.	------NOTE----- SR 3.0.2 is not applicable ------------- In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions
[SR 3.6.1.3.15	[-------------------NOTE------------------- Only required to be met in MODES 1, 2, and 3. --] Verify each [ ] inch primary containment purge valve is blocked to restrict the valve from opening > [50]%.	[18] months]

1

1

1. This bracketed requirement has been deleted because it is not applicable to Quad Cities 1 and 2. The following requirements have been renumbered, where applicable, to reflect this deletion.

2. The words "in MODES 1, 2, and 3" have been deleted from ITS 3.6.1.3 ACTIONS Note 4 since there are no PCIV leakage tests required in MODES other than MODES 1, 2, and 3 for Quad Cities 1 and 2 (i.e., there are no PCIVs required to be OPERABLE in MODES other than MODES 1, 2, and 3 that have specific leakage limits). In addition, ISTS SR 3.6.1.3.2 Note 1 has been deleted for the same reason. The following Note number has been deleted since the deletion of this Note leaves only one applicable Note.

3. The words in ISTS Conditions A and B Notes and the words in ISTS Condition B have been modified to state "two or more" in lieu of "two." Some penetration flow paths at Quad Cities 1 and 2 have more than two PCIVs. This change will ensure an LCO 3.0.3 entry is not required for this design and the appropriate actions are taken consistent with a plant with only two PCIVs per penetration flow path. This change is also consistent with TSTF-207, Rev. 3.

4. The words inside the brackets have been modified to reflect the appropriate leakage category. Since there is only one category, the words "MSIV leakage rate" have been used in ISTS 3.6.1.3 Conditions A, B, and D. The PCIVs are required to be OPERABLE such that they are in the accident condition or can be automatically repositioned to the accident condition, and only MSIVs have individual leakage limits. These leakage limits are in addition to the type A, B, and C limits required by LCO 3.6.1.1, Primary Containment OPERABILITY. If a type A, B, or C limit were exceeded due to an individual valve exceeding its specific leakage limit, ISTS 3.6.1.3 ACTIONS Note 4 would require the ACTIONS of LCO 3.6.1.1 to be taken (which require primary containment to be restored within 1 hour).

   The change was made to reflect that different compensatory actions are required depending upon the cause of the inoperability. In the Quad Cities 1 and 2 ITS, ACTION A is taken if the PCIV is inoperable for reasons other than MSIV leakage; ACTION D is required if the SRs for individual MSIV leakage limits are not met. Currently (in the ISTS), Conditions A and B would only exempt purge valve leakage requirements and Condition C does not exempt any leakage requirements. If an MSIV is not meeting the leakage limits, Condition A would be entered and Required Action A.1 would be required. This Required Action allows the penetration to be isolated. However, isolating the penetration can be performed by using the leaking valve. This would not provide adequate compensatory measures to allow continued operation. When MSIV leakage is not within limits, Condition D should be entered. The Required Action for this Condition would require the leakage to be restored within

4. (continued)

limit in 8 hours consistent with the time provided in Required Actions A.1 to isolate an MSIV penetration. As discussed in the ISTS Bases, the leakage can be restored by isolating the penetration with a valve not exceeding the leakage limits. This is more restrictive than Required Action A.1, which allows isolation using the leaking valve. Condition B has also been modified to exclude MSIV leakage. This Condition is appropriate if two MSIVs will not close. As discussed above, the Required Action for Condition B would also allow the penetration to be isolated using the leaking MSIV if the bracketed phrase were not modified. This change is also consistent with TSTF-207, Rev. 3, except when plant specific differences apply or consistency errors were noted.

5. ITS 3.6.1.3 Required Action C.1 Completion Times have been modified to be consistent with approved TSTF-30, Rev. 3. The change also provides a 72 hour Completion Time for EFCVs consistent with TSTF-323.

6. Not used.

7. The words in ISTS 3.6.1.3 Condition I (ITS 3.6.1.3 Condition F), "or during operations with a potential for draining the reactor vessel (OPDRVs)," have been deleted. There are no PCIVs required to be OPERABLE in the Quad Cities 1 and 2 ITS whose Applicability is only during OPDRVs. The only PCIVs required when not in MODES 1, 2, and 3 are the RHR shutdown cooling isolation valves, and their Applicability is MODES 1, 2, 3, 4 and 5. This Condition is still applicable in MODES 4 and 5, which are the only MODES that OPDRVs can be performed. Therefore, the "during OPDRVs" Applicability is duplicative of the MODES 4 and 5 Applicability and has been deleted.

8. The acronym "OPDRVs" has been defined, consistent with the format of the ITS, since it is the first use of this term in this Specification.

9. The brackets have been removed and the proper plant specific information/value has been provided.

10. The Primary Containment Leakage Rate Testing Program has been added to ITS Section 5.5, similar to TSTF-52. The Program references the requirements of 10 CFR 50 Appendix J and approved exemptions, therefore, the Surveillances have been modified to reference the Program. This is consistent with the Current Licensing Basis and TSTF-52.

11. ISTS SR 3.6.1.3.13 (ITS SR 3.6.1.3.10), the MSIV leakage rate test, has been modified from a "per valve" basis to a "combined" leakage rate basis consistent with the current licensing basis.

12. The 18 inch torus purge valve has been excluded from the requirement in ISTS 3.6.1.3.2 (ITS 3.6.1.3.1), since it is normally open for pressure control.

<CTS>

3.6 CONTAINMENT SYSTEMS

3.6.1.4 Drywell Pressure

<3.7.6> LCO 3.6.1.4 Drywell pressure shall be ~~[~~≤ 0.75 1.5 psig~~]~~. 1

<Appl 3.7.6> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

	CONDITION	REQUIRED ACTION		COMPLETION TIME
<3.7.6 Act 2>	A. Drywell pressure not within limit.	A.1	Restore drywell pressure to within limit.	1 hour
<3.7.6 Act 2>	B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
		AND		
		B.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
<4.7.6>	SR 3.6.1.4.1	Verify drywell pressure is within limit.	12 hours

1. The brackets have been removed and the proper plant specific information/value has been provided.

c(CTS)

3.6 CONTAINMENT SYSTEMS

3.6.1.5 Drywell Air Temperature

<DOC M.1> LCO 3.6.1.5 Drywell average air temperature shall be ≤ ~~[135]~~ 150 [1] °F.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<DOC M.1>	A. Drywell average air temperature not within limit.	A.1 Restore drywell average air temperature to within limit.	8 hours
<DOC M.1>	B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
<DOC M.1>	SR 3.6.1.5.1 Verify drywell average air temperature is within limit.	24 hours

1. The brackets have been removed and the proper plant specific information/value has been provided.

<CTS>

Low Set Relief 1

3.6 CONTAINMENT SYSTEMS 1

3.6.1.6 Low-~~Low~~ Set (LLS) Relief Valves

<3.6.F> LCO 3.6.1.6 The LLS low set relief 1 function of ~~[four]~~ two 2 ~~safety~~ 1 relief valves shall be OPERABLE.

<Appl 3.6.F> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.F Act 2> A. One LLS low set relief valve inoperable.	A.1 Restore LLS low set relief valve to OPERABLE status.	14 days 1
<3.6.F Act 2> B. Required Action and associated Completion Time of Condition A not met. OR <3.6.F Act 3> Two ~~or more~~ 2 LLS low set relief 1 valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

<CTS>

Low Set Relief 1

SURVEILLANCE REQUIREMENTS

DOC A.3

SURVEILLANCE		FREQUENCY
SR 3.6.1.6.1	------------------NOTE------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify each LLS valve opens when manually actuated. low set relief 1	24 2 ~~18~~ months ~~on a STAGGERED TEST BASIS for each valve solenoid~~ 3
SR 3.6.1.6.2	------------------NOTE------------------ Valve actuation may be excluded. -- Verify ~~the LLS System~~ actuates on an actual or simulated automatic initiation signal. 1 each low set relief valve	24 4 ~~18~~ months

DOC A.3

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.6.1.6 - LOW SET RELIEF VALVES

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. The brackets have been removed and the proper plant specific value/nomenclature has been provided.

3. The bracketed information has been deleted since it does not apply to Quad Cities 1 and 2.

4. The 18 month Frequency of ISTS SR 3.6.1.6.2 has been changed to 24 months consistent with the Quad Cities 1 and 2 fuel cycle.

<CTS>

3.6 CONTAINMENT SYSTEMS

3.6.1.7 Reactor Building-to-Suppression Chamber Vacuum Breakers

<3.7.F> LCO 3.6.1.7 Each reactor building-to-suppression chamber vacuum breaker shall be OPERABLE.

<Appl 3.7.F> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each line.

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.7.F Act 2>	A. One or more lines with one reactor building-to-suppression chamber vacuum breaker not closed.	A.1 Close the open vacuum breaker.	~~12~~ hours 7 days 3
<3.7.F Act 2>	B. One or more lines with two reactor building-to-suppression chamber vacuum breakers not closed.	B.1 Close one open vacuum breaker.	1 hour
<3.7.F Act 1>	C. One line with one or more reactor building-to-suppression chamber vacuum breakers inoperable for opening.	C.1 Restore the vacuum breaker(s) to OPERABLE status.	~~72~~ hours 7 days 3

(continued)

<CTS>

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME
<DOC L.1>	D. Two $\boxed{\text{or more}}$ [1] lines with one or more reactor building-to-suppression chamber vacuum breakers inoperable for opening.	D.1	Restore all vacuum breakers in {one} [2] line to OPERABLE status.	1 hour
<3.7.F Act 1>	E. Required Action and Associated Completion Time not met.	E.1	Be in MODE 3.	12 hours
		<u>AND</u>		
<3.7.F Act 2>		E.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
<4.7.F.1>	SR 3.6.1.7.1	-------------------NOTES------------------- 1. Not required to be met for vacuum breakers that are open during Surveillances. 2. Not required to be met for vacuum breakers open when performing their intended function. --- Verify each vacuum breaker is closed.	14 days
<4.7.F.2.a>	SR 3.6.1.7.2	Perform a functional test of each vacuum breaker.	{92} [2] days

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
<4.7.F.2.b> SR 3.6.1.7.3	Verify the opening setpoint of each vacuum breaker is ≤ ~~[~~0.5~~]~~ psid. 2	~~[18]~~ 24 months 2

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.6.1.7 - REACTOR BUILDING-TO-SUPPRESSION CHAMBER VACUUM BREAKERS

1. The brackets have been removed and the information deleted since it does not apply to Quad Cities 1 and 2.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. The Completion Time has been revised to reflect the current licensing basis reflected in Technical Specifications.

3.6 CONTAINMENT SYSTEMS

3.6.1.8 Suppression Chamber-to-Drywell Vacuum Breakers

⟨3.7.E⟩

LCO 3.6.1.8 [Nine] suppression chamber-to-drywell vacuum breakers shall be OPERABLE for opening. 1

AND

[Twelve] suppression chamber-to-drywell vacuum breakers shall be closed, ~~except when performing their intended function~~. 1 2

⟨Appl 3.7.E⟩

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
⟨3.7.E Act 1⟩	A. One required suppression chamber-to-drywell vacuum breaker inoperable for opening.	A.1 Restore one vacuum breaker to OPERABLE status.	72 hours
⟨3.7.E Act 2⟩	B. One suppression chamber-to-drywell vacuum breaker not closed.	B.1 Close the open vacuum breaker.	2 hours 4 3
⟨3.7.E Act 1⟩ ⟨3.7.E Act 2⟩	C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. **AND** C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.8.1 <DOC L.3> <DOC A.2> <4.7.E.1>	----NOTE(S)---- [2] 1. Not required to be met for vacuum breakers that are open during Surveillances. [2] 2. Not required to be met for vacuum breakers open when performing their intended function. [2] Verify each vacuum breaker is closed.	14 days ~~AND~~ ~~Within 2 hours after any discharge of steam to the suppression chamber from the safety/relief valves (S/RVs) or any operation that causes the drywell-to-suppression chamber differential pressure to be reduced by ≥ [0.5] psid~~ [4]

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
⟨4.7.E.2.a⟩ SR 3.6.1.8.2	Perform a functional test of each required vacuum breaker.	31 days AND Within 12 hours after any discharge of steam to the suppression chamber from the ~~S/RVs~~ relief valves [5] ~~AND~~ ~~Within 12 hours following an operation that causes any of the vacuum breakers to open~~ [6]
⟨4.7.E.2.c⟩ SR 3.6.1.8.3	Verify the opening setpoint of each required vacuum breaker is ≤ ~~[0.5]~~ psid. [1]	~~[18]~~ 24 months [1]

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. A portion of the second part of the LCO statement ("except when performing their intended function") has been moved to the Surveillance (SR 3.6.1.8.1) in the form of a Note. The location of the Note is consistent with the BWR/4 ISTS SR 3.6.1.7.1 for reactor building-to-suppression chamber vacuum breakers. Also, the existing Note and the new Note to SR 3.6.1.8.1 have been numbered for clarity.

3. The Completion Time of Required Action B.1 has been extended from 2 to 4 hours consistent with existing requirements. Entry into ACTION B will be required when SR 3.6.1.8.1 is not satisfied or between surveillances as required by SR 3.0.1. The 4 hours is needed to prepare and perform the alternate verification of valve position (total leakage between the suppression chamber and drywell). This 4 hour allowance will not be taken if it is known that the leakage limit is not met. In this case, entry into ITS 3.6.1.1 ACTION A will be required.

4. The second Frequency to ISTS SR 3.6.1.8.1 requires the vacuum breakers to be verified closed after they may have been opened. This Frequency is not needed and has not been included in ITS SR 3.6.1.8.1. Surveillances must be continually met (per SR 3.0.1), thus if the vacuum breakers are open and the Surveillance is not due yet, the SR would still be considered not met, and appropriate ACTIONS taken. There are many other instances where valves are required to be closed, and verified closed on a periodic basis. If these other valves are cycled (e.g., ECCS valves) plant administrative controls ensure they are left in the correct position; a special Frequency of the Surveillance is not required. In addition, these vacuum breakers have local position indication with alarms in the control room, which are monitored by control room operators. If conditions exist for the vacuum breakers to be potentially opened (e.g., venting the drywell), control room operators would be alert to the possibility and ensure the vacuum breakers were closed at the completion of the evolution. Also, this Surveillance Frequency is not required in the current Quad Cities 1 and 2 Technical Specifications.

5. The proper plant specific information/nomenclature/value has been provided.

6. The third Frequency to ISTS SR 3.6.1.8.2 requires a functional test of the vacuum breakers (i.e., cycle the vacuum breakers) within 12 hours after the vacuum breakers have cycled. In a September 8, 1992 memorandum to C.I. Grimes from C.E. McCracken, the only basis for this Frequency is given as..."in case the event caused damage to one or more vacuum breakers."

6. (continued)

Since the vacuum breakers are designed to operate and are assumed to function after a LOCA blowdown, their operation as designed after some steam release or change in internal pressure should not raise questions regarding immediate OPERABILITY of the vacuum breakers. In addition, local position indication and redundant control room alarms are provided for each vacuum breaker such that the control room operators would be alerted to the possibility of a stuck open vacuum breaker and would take the appropriate action (e.g., close the vacuum breaker) to ensure isolation capability is maintained. Therefore, this Frequency, which is not required in the current Technical Specifications for Quad Cities 1 and 2, has not been added to the Quad Cities 1 and 2 ITS.

3.6 CONTAINMENT SYSTEMS

3.6.1.9 Main Steam Isolation Valve (MSIV) Leakage Control System (LCS)

LCO 3.6.1.9 Two MSIV LCS subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One MSIV LCS subsystem inoperable.	A.1	Restore MSIV LCS subsystem to OPERABLE status.	30 days
B. Two MSIV LCS subsystems inoperable.	B.1	Restore one MSIV LCS subsystem to OPERABLE status.	7 days
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours
	AND		
	C.2	Be in MODE 4.	36 hours

1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.9.1	Operate each MSIV LCS blower ≥ [15] minutes.	31 days
SR 3.6.1.9.2	Verify electrical continuity of each inboard MSIV LCS subsystem heater element circuitry.	31 days
SR 3.6.1.9.3	Perform a system functional test of each MSIV LCS subsystem.	[18] months

1

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ISTS: 3.6.1.9 - MAIN STEAM ISOLATION VALVE (MSIV) LEAKAGE CONTROL SYSTEM (LCS)

1. The Quad Cities 1 and 2 design does not include a Main Steam Isolation Valve (MSIV) Leakage Control System (LCS). Therefore, this Specification has been deleted.

<CTS>

3.6 CONTAINMENT SYSTEMS

3.6.2.1 Suppression Pool Average Temperature

TSTF-206 — {with THERMAL POWER > 1% RTP}

LCO 3.6.2.1 Suppression pool average temperature shall be:

<3.7.K.2> 1

a. ≤ {95}°F [when any OPERABLE intermediate range monitor (IRM) channel is > [25/40] divisions of full scale on Range 7] and no testing that adds heat to the suppression pool is being performed;

b. ≤ {105}°F [when any OPERABLE IRM channel is > [25/40] divisions of full scale on Range 7] and testing that adds heat to the suppression pool is being performed; and

c. ≤ {110}°F [when all OPERABLE IRM channels are ≤ [25/40] divisions of full scale on Range 7].

{with THERMAL POWER ≤ 1% RTP} — TSTF-206

<Appl 3.7.K> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<4.7.K.2.b> <3.7.K Act 2> <3.7.K Act 3> A. Suppression pool average temperature > {95}°F but ≤ {110}°F. 1 AND 1 ~~Any OPERABLE IRM channel > [25/40] divisions of full scale on Range 7~~ {THERMAL POWER > 1% RTP} TSTF-206 AND Not performing testing that adds heat to the suppression pool.	A.1 Verify suppression pool average temperature ≤ {110}°F. 1 AND A.2 Restore suppression pool average temperature to ≤ {95}°F. 1	Once per hour 24 hours

(continued)

ε <CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.7.K Act 2> B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER ~~[until all OPERABLE IRM channels ≤ [25/40] divisions of full scale on Range 7]~~ to ≤ 1% RTP 1 TSTF-206	12 hours 1
<3.7.K Act 3> C. Suppression pool average temperature 1 [> [105]°F. AND 1 ~~[Any OPERABLE IRM channel > [25/40] divisions of full scale on Range 7]~~ THERMAL POWER > 1% RTP 1 TSTF-206 AND Performing testing that adds heat to the suppression pool.	C.1 Suspend all testing that adds heat to the suppression pool.	Immediately
<3.7.K Act 4> D. Suppression pool average temperature 1 [> [110]°F but ≤ [120]°F. <4.7.K.2.c>	D.1 Place the reactor mode switch in the shutdown position. AND D.2 Verify suppression pool average temperature ≤ [120]°F.] 1 AND D.3 Be in MODE 4.	Immediately Once per 30 minutes 36 hours

(continued)

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME
<3.7.K.2.c> <3.7.K Act 5>	E. Suppression pool average temperature > ~~[~~120~~]~~°F. $\boxed{1}$	E.1	Depressurize the reactor vessel to < ~~[200]~~ 150 psig. $\boxed{1}$	12 hours
		AND		
		E.2	Be in MODE 4.	~~[~~36 hours~~]~~ $\boxed{1}$

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
<4.7.K.2> <4.7.K.2.a>	SR 3.6.2.1.1	Verify suppression pool average temperature is within the applicable limits.	24 hours AND 5 minutes when performing testing that adds heat to the suppression pool

1. The brackets have been removed and the proper plant specific information/value has been provided.

3.6 CONTAINMENT SYSTEMS

3.6.2.2 Suppression Pool Water Level

⟨3.7.K.1⟩ ⟨3.5.c.1⟩

LCO 3.6.2.2 Suppression pool water level shall be ≥ 14 ft 1 inch ~~[12 ft 2 inches]~~ and ≤ ~~[12 ft 6 inches]~~ 14 ft 5 inches $\boxed{1}$

⟨Appl 3.7.K⟩ ⟨Appl 3.5.c⟩

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
⟨3.7.K Act 1⟩ ⟨3.5.C Act 1⟩ A. Suppression pool water level not within limits.	A.1	Restore suppression pool water level to within limits.	2 hours
⟨3.7.K Act 1⟩ ⟨3.5.C Act 1⟩ B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
⟨4.5.c.1⟩ SR 3.6.2.2.1	Verify suppression pool water level is within limits.	24 hours

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.6.2.2 - SUPPRESSION POOL WATER LEVEL

1. The brackets have been removed and the proper plant specific information/value has been provided.

<CTS>

3.6 CONTAINMENT SYSTEMS

3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

<3.7.M> LCO 3.6.2.3 Two RHR suppression pool cooling subsystems shall be OPERABLE.

<Appl 3.7.M> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
<3.7.M Act 1>	A. One RHR suppression pool cooling subsystem inoperable.	A.1 Restore RHR suppression pool cooling subsystem to OPERABLE status.	7 days	
<3.7.M Act 1>	~~B.~~ C. Required Action and associated Completion Time ~~of Condition A~~ or B not met. ~~OR~~ [5]	~~B.1~~ C.1 Be in MODE 3. AND ~~B.2~~ C.2 Be in MODE 4.	12 hours 36 hours	TSTF-230
<3.7.M Act 2> <DOC L.1>	B. Two RHR suppression pool cooling subsystems inoperable.	B.1 Restore one RHR suppression pool cooling subsystem to OPERABLE status.	8 hours	TSTF-230

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
⟨4.7.M.1⟩	SR 3.6.2.3.1 [1] [2] [1] Verify each RHR suppression pool cooling subsystem 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 or can be aligned to the correct position.	31 days
⟨4.7.M.2⟩	SR 3.6.2.3.2 [4] Verify each required [3] RHR pump develops a flow rate ≥ ~~[7700]~~ 5000 gpm through the associated heat exchanger while operating in the suppression pool cooling mode.	[4] In accordance with the Inservice Testing Program ~~or 92 days~~

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.6.2.3 - RHR SUPPRESSION POOL COOLING

1. Editorial change made to be consistent with other similar requirements in the ITS.

2. The Quad Cities 1 and 2 design does not include any automatically actuated RHR suppression pool cooling valves. The RHR suppression pool cooling mode is manually actuated. Therefore, the word "automatic" in ITS SR 3.6.2.3.1 has been deleted.

3. The Quad Cities 1 and 2 design only requires one of the two RHR pumps in a suppression pool cooling subsystem. Therefore, ISTS SR 3.6.2.3.2 has been modified to only require the "required" RHR pumps to be tested. This change is consistent with the use of the word "required" in the ITS.

4. The brackets have been removed and the proper plant specific information/value has been provided.

5. The words "of Condition A or B" (as modified by TSTF-230) have been deleted to be consistent with all other similar conditions in the ITS. The format of the ITS is not to use the term "of Condition X" in a Condition, when the Condition applies to all Conditions previous to it and it is the last Condition in the ACTIONS Table.

<CTS>

3.6 CONTAINMENT SYSTEMS

3.6.2.4 Residual Heat Removal (RHR) Suppression Pool Spray

<3.7.L> LCO 3.6.2.4 Two RHR suppression pool spray subsystems shall be OPERABLE.

<Appl 3.7.L> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.7.L Act 1>	A. One RHR suppression pool spray subsystem inoperable.	A.1 Restore RHR suppression pool spray subsystem to OPERABLE status.	7 days
<3.7.L Act 2>	B. Two RHR suppression pool spray subsystems inoperable.	B.1 Restore one RHR suppression pool spray subsystem to OPERABLE status.	8 hours
<3.7.L Act 1>	C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours
<3.7.L Act 2>		<u>AND</u> C.2 Be in MODE 4.	36 hours

⟨CTS⟩

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.4.1	Verify each RHR suppression pool spray subsystem 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 or can be aligned to the correct position.	31 days
~~SR 3.6.2.4.2~~	~~Verify each RHR pump develops a flow rate ≥ [400] gpm through the heat exchanger while operating in the suppression pool spray mode.~~	~~In accordance with the Inservice Testing Program or 92 days~~
SR 3.6.2.4.2	Verify each suppression pool spray nozzle is unobstructed.	5 years

⟨4.7.L.1⟩ [1] [2] [3] [4] ⟨DOC M1⟩

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.6.2.4 - RHR SUPPRESSION POOL SPRAY

1. The Quad Cities 1 and 2 design does not include an automatically actuated RHR Suppression Pool Spray System; the system is entirely manually actuated. Therefore, the word "automatic" has been deleted from the valve position check Surveillance (ITS SR 3.6.2.4.1).

2. Editorial change made to be consistent with other similar specifications.

3. The bracketed requirement has been deleted. The current licensing basis for Quad Cities 1 and 2 does not require a suppression pool spray flow rate verification.

4. A new Surveillance was added which verifies each suppression pool spray nozzle is unobstructed every 5 years. This Surveillance is required to ensure that when a suppression pool spray subsystem is required per its design function that it will perform as designed. If the spray nozzles are obstructed, then their design function may not be met. The 5 year Frequency is consistent with the current requirement for verifying that the drywell spray nozzles remain unobstructed.

<(CTS)>

3.6 CONTAINMENT SYSTEMS

3.6.2.5 Drywell-to-Suppression Chamber Differential Pressure

<3.7.H> LCO 3.6.2.5 The drywell pressure shall be maintained ≥ ~~[1.5]~~ 1.0 psid above the pressure of the suppression chamber. [1]

<Appl 3.7.H> APPLICABILITY: MODE 1 during the time period:

a. From [24] hours after THERMAL POWER is > [15]% RTP following startup, to

b. [24] hours prior to reducing THERMAL POWER to < [15]% RTP prior to the next scheduled reactor shutdown. [1]

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.7.H Act 1> A. Drywell-to-suppression chamber differential pressure not within limit.	A.1 Restore differential pressure to within limit.	~~8~~ 24 hours [3]
<3.7.H Act 1> B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ [15]% RTP. [1]	~~12~~ 8 hours [3]

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<4.7.H.1> SR 3.6.2.5.1 Verify drywell-to-suppression chamber differential pressure is within limit.	12 hours

<3.7.H footnote a> NOTE [2]
Not required to be met for up to 4 hours during performance of required Surveillances.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.6.2.5 - DRYWELL-TO-SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. A Note has been added providing a period of up to 4 hours when LCO 3.6.2.5 is not required to be met to allow performance of required Surveillances that reduce the differential pressure. This allowance was provided as footnote a for CTS 3.7.H. This change is consistent with the current licensing basis.

3. The Completion Time has been revised to reflect the current licensing basis in accordance with Amendments 165 and 161, dated November 27, 1995.

1

3.6 CONTAINMENT SYSTEMS

3.6.3.1 Primary Containment Hydrogen Recombiners (if permanently installed)

LCO 3.6.3.1 Two primary containment hydrogen recombiners shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One primary containment hydrogen recombiner inoperable.	A.1	-------NOTE------- LCO 3.0.4 is not applicable. ------------------ Restore primary containment hydrogen recombiner to OPERABLE status.	30 days
[B. Two primary containment hydrogen recombiners inoperable.	B.1	Verify by administrative means that the hydrogen control function is maintained.	1 hour AND Once per 12 hours thereafter
	AND		
	B.2	Restore one primary containment hydrogen recombiner to OPERABLE status.	7 days]

(continued)

[1]

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.3.1.1	Perform a system functional test for each primary containment hydrogen recombiner.	[18] months
SR 3.6.3.1.2	Visually examine each primary containment hydrogen recombiner enclosure and verify there is no evidence of abnormal conditions.	[18] months
SR 3.6.3.1.3	Perform a resistance to ground test for each heater phase.	[18] months

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ISTS: 3.6.3.1 - PRIMARY CONTAINMENT HYDROGEN RECOMBINERS

1. The Quad Cities 1 and 2 design does not include Primary Containment Hydrogen Recombiners. Therefore, this Specification has been deleted.

3.6 CONTAINMENT SYSTEMS

3.6.3.2 [Drywell Cooling System Fans]

1

LCO 3.6.3.2 Two [drywell cooling system fans] shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One [required] [drywell cooling system fan] inoperable.	A.1 ---------NOTE--------- LCO 3.0.4 is not applicable. ------------------------- Restore [required] [drywell cooling system fan] to OPERABLE status.	30 days
B. Two [required] [drywell cooling system fans] inoperable.	B.1 Verify by administrative means that the hydrogen control function is maintained.	1 hour **AND** Once per 12 hours thereafter
	AND B.2 Restore one [required] [drywell cooling system fan] to OPERABLE status.	7 days
C. Required Action and Associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.3.2.1	Operate each [required] [drywell cooling system fan] for ≥ [15] minutes.	92 days
[SR 3.6.3.2.2	Verify each [required] [drywell cooling system fan] flow rate is ≥ [500] scfm.	[18] months]

1

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ISTS: 3.6.3.2 - DRYWELL COOLING SYSTEM FANS

1. The current Quad Cities 1 and 2 licensing basis does not include Technical Specification requirements for Drywell Cooling System fans (i.e., hydrogen mixing fans) since the hydrogen control analysis does not assume the fans function to mix the primary containment atmosphere (i.e., the atmosphere is mixed by natural convection). In addition, the fans are automatically tripped on a LOCA signal. Therefore, this Specification has been deleted.

<CTS>

3.6 CONTAINMENT SYSTEMS

3.6.3.3 Primary Containment Oxygen Concentration

1 1

<3.7.J> LCO 3.6.3.3 The primary containment oxygen concentration shall be < 4.0 volume percent.

<Appl 3.7.J> APPLICABILITY: MODE 1 during the time period:

a. From [24] hours after THERMAL POWER is > [15]% RTP following startup, to 2

b. [24] hours prior to reducing THERMAL POWER to < [15]% RTP prior to the next scheduled reactor shutdown.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.7.J Act>	A. Primary containment oxygen concentration not within limit.	A.1 Restore oxygen concentration to within limit.	24 hours
<3.7.J Act>	B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ [15]% RTP.	8 hours 2

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
<4.7.J>	SR 3.6.3.3.1 Verify primary containment oxygen concentration is within limits.	7 days

1 1

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.6.3.1 - PRIMARY CONTAINMENT OXYGEN CONCENTRATION

1. The Specification has been renumbered due to the deletion of ISTS 3.6.3.1 and 3.6.3.2.

2. The brackets have been removed and the proper plant specific information/value has been provided.

1

3.6 CONTAINMENT SYSTEMS

3.6.3.4 Containment Atmosphere Dilution (CAD) System

LCO 3.6.3.4 Two CAD subsystems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CAD subsystem inoperable.	A.1	---------NOTE--------- LCO 3.0.4 is not applicable. ---------------------- Restore CAD subsystem to OPERABLE status.	30 days
B. Two CAD subsystems inoperable.	B.1	Verify by administrative means that the hydrogen control function is maintained.	1 hour AND Once per 12 hours thereafter
	AND B.2	Restore one CAD subsystem to OPERABLE status.	7 days
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours

[1]

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.3.4.1	Verify ≥ [4350] gal of liquid nitrogen are contained in the CAD System.	31 days
SR 3.6.3.4.2	Verify each CAD subsystem 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 or can be aligned to the correct position.	31 days

1. NUREG-1433, Specification 3.6.3.4, "Containment Atmosphere Dilution (CAD) System," is not included in the Quad Cities 1 and 2 ITS. This Specification is deleted since the current licensing basis for Quad Cities 1 and 2, as reflected in the CTS, do not include requirements for a CAD System. The NRC, in an SER dated June 28, 1996, found the deletion of CTS 3.7.I, "Primary Containment Nitrogen System," and the relocation of the Nitrogen System requirements to the UFSAR to be acceptable for Dresden 2 and 3 and also consistent with the CTS for Quad Cities. The Nitrogen System supports the requirements for primary containment oxygen concentration, which has requirements specified in CTS 3/4.7.J (ITS 3.6.3.1). The Nitrogen System also performs the CAD System function to maintain post-accident combustible gas concentrations within the primary containment at or below the flammability limits by purging the containment atmosphere with nitrogen. The NRC determined that licensee controlled procedures and administrative controls are adequate to ensure Nitrogen System operability. Thus, the Nitrogen System will maintain the containment in an inerted condition as required by CTS 3/4.7.J (ITS 3.6.3.1) and remain capable of purging the containment with nitrogen as necessary under accident conditions. Therefore, consistent with the current licensing basis, CAD System requirements are not included in the ITS.

<CTS>

3.6 CONTAINMENT SYSTEMS

3.6.4.1 [Secondary] Containment

<3.7.N> LCO 3.6.4.1 The [secondary] containment shall be OPERABLE.

<Appl 3.7.N> APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the [secondary] containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

$\boxed{1}$

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.7.N Act 1>	A. [Secondary] containment inoperable in MODE 1, 2, or 3.	A.1 Restore [secondary] containment to OPERABLE status.	4 hours $\boxed{1}$
<3.7.N Act 1>	B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
<3.7.N Act 2>	C. [Secondary] containment inoperable during movement of irradiated fuel assemblies in the [secondary] containment, during CORE ALTERATIONS, or during OPDRVs.	C.1 ---------NOTE--------- LCO 3.0.3 is not applicable. ------------------------- Suspend movement of irradiated fuel assemblies in the [secondary] containment. AND	Immediately $\boxed{1}$ (continued)

<CTS>

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
<3.7.N Act 2> C. (continued)	C.2	Suspend CORE ALTERATIONS.	Immediately
	AND		
	C.3	Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
<4.7.N.1>	SR 3.6.4.1.1	Verify ~~secondary~~ containment vacuum is ≥ ~~[0.25]~~ .10 inch of vacuum water gauge.	24 hours 1
	~~SR 3.6.4.1.2~~	~~Verify all [secondary] containment equipment hatches are closed and sealed.~~	~~31 days~~ 2
<4.7.N.2>	SR 3.6.4.1.~~3~~ 2 2	Verify ~~each~~ one ~~secondary~~ containment access door in each access opening is closed~~, except when the access opening is being used for entry and exit[, then at least one door shall be closed]~~. 1	31 days TSTF-18
	~~SR 3.6.4.1.4~~	~~Verify each standby gas treatment (SGT) subsystem will draw down the [secondary] containment to ≥ [0.25] inch of vacuum water gauge in ≤ [120] seconds.~~	~~[18] months on a STAGGERED TEST BASIS~~ 5

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
<4.7.N.3> 2	SR 3.6.4.1.3 3	the secondary containment can be maintained Verify each SGT subsystem can maintain ≥ [0.25] inch of vacuum water gauge in the [secondary] containment for 1 hour, at a flow rate ≤ [4000] cfm. using one SGT subsystem 4	24 [18] months on a STAGGERED TEST BASIS for each SGT subsystem 1 4

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.6.4.1 - SECONDARY CONTAINMENT

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. ISTS SR 3.6.4.1.2, which verifies all secondary containment equipment hatches are closed and sealed, is not retained in the ITS. The Surveillance Requirement was not added during the Technical Specification Upgrade Program, in accordance with Amendment 171 and 167 respectively, issued by the NRC on November 27, 1995. The following requirements have been revised or renumbered, where applicable, to reflect this deletion.

3. Not used.

4. ISTS SR 3.6.1.4.5 is a test that ensures the Secondary Containment is Operable; the leak tightness of the Secondary Containment boundary is within the assumptions of the accident analyses. However, it is written in such a manner it implies that if a SGT subsystem is inoperable, the SR is failed ("Verify each standby gas treatment (SGT) subsystem can..."). As stated above, this is not the intent of the SR. Therefore, to ensure this misinterpretation cannot occur, the SR has been rephrased to more clearly convey the original intent of the SR, to verify the Secondary Containment is Operable. With the new wording, if a SGT subsystem is inoperable, ITS SR 3.6.4.1.3 will still be met and only the SGT System Specification, LCO 3.6.4.3, will be required to be entered. The SR will still ensure each SGT subsystem is used (on a STAGGERED TEST BASIS) to perform the SR. This change is also consistent with TSTF-322.

5. The bracketed Surveillance (ISTS SR 3.6.4.1.4), the drawdown test, has been deleted consistent with the current licensing basis. The analysis does not assume an explicit drawdown time. The subsequent SR has been renumbered to reflect this deletion.

<CTS>

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

LCO 3.6.4.2 Each SCIV shall be OPERABLE.

<Appl 3.7.D>

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the [secondary] containment, 1
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

-----NOTES-----

<DOC L.1> 1. Penetration flow paths may be unisolated intermittently under administrative controls.

<DOC A.3> 2. Separate Condition entry is allowed for each penetration flow path.

<DOC A.3> 3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIVs.

<3.7.D Act>
<3.7.D Act 2+3>

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one SCIV inoperable.	A.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. AND	8 hours
		(continued)

ε⟨CTS⟩

ACTIONS

⟨3.7.0 Act⟩

⟨3.7.0 Act 2&3⟩

⟨4.7.N.2.b⟩

⟨4.7.N.2.b footnote *⟩

⟨DOC L.2⟩

⟨3.7.0 Act⟩

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 ① ---------NOTE--------- ⑤ 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.	TSTF-269 Once per 31 days
B. ---------NOTE--------- Only applicable to penetration flow paths with two isolation valves. --------------------- One or more penetration flow paths with two SCIVs 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.	4 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

(continued)

ε(CTS)

ACTIONS (continued)

⟨3.7.0 Act⟩

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies in the [secondary] containment, during CORE ALTERATIONS, or during OPDRVs.	D.1 -----NOTE----- LCO 3.0.3 is not applicable. ------------------ Suspend movement of irradiated fuel assemblies in the [secondary] containment.	Immediately 1
	AND	
	D.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	D.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
⟨4.7.N.2.b⟩ ⟨4.7.N.2.b footnote a⟩ SR 3.6.4.2.1	-------------------NOTES------------------- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for SCIVs that are open under administrative controls. -- Verify each secondary containment isolation manual valve and blind flange that is and not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed.	TSTF-45 Rev.2 31 days
⟨DOC M.1⟩ SR 3.6.4.2.2 TSTF-46 ①	Verify the isolation time of each power operated ~~and each~~ automatic SCIV is within limits.	1 ~~[In accordance with the Inservice Testing Program or]~~ 92 days
⟨4.7.0.2⟩ SR 3.6.4.2.3	Verify each automatic SCIV actuates to the isolation position on an actual or simulated actuation signal.	~~[18]~~ 24 months 1

1. The brackets have been removed and the proper plant specific information/value has been provided.

ε⟨CTS⟩

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

⟨3.7.P⟩ LCO 3.6.4.3 ⁑Two⁑ SGT subsystems shall be OPERABLE. [1]

⟨Appl 3.7.P⟩ APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the ⁑secondary⁑ containment,
During CORE ALTERATIONS,
⟨Appl 3.7.P footnote⟩ During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
⟨3.7.P Act 1⟩ A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
⟨3.7.P Act 1.c⟩ B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
⟨3.7.P Act 1.b⟩ C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the ⁑secondary⁑ containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ---------------------------- C.1 Place OPERABLE SGT subsystem in operation. OR	Immediately [1]

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued) ⟨3.7.P Act 1.b⟩	C.2.1 Suspend movement of irradiated fuel assemblies in *secondary* containment.	Immediately
	AND	
	C.2.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	C.2.3 Initiate action to suspend OPDRVs.	Immediately
⟨3.7.P Act 2⟩ D. Two SGT subsystems inoperable in MODE 1, 2, or 3.	D.1 ~~Enter LCO 3.0.3~~ Restore one SGT subsystem to OPERABLE status	~~Immediately~~ 1 hour
E. Required Action and associated Completion Time of Condition D not met. ⟨3.7.P Act 2⟩	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	12 hours 36 hours
2 F ~~E~~. Two SGT subsystems inoperable during movement of irradiated fuel assemblies in the *secondary* containment, during CORE ALTERATIONS, or during OPDRVs.	F ~~E~~.1 --------NOTE-------- LCO 3.0.3 is not applicable. ----------------- Suspend movement of irradiated fuel assemblies in *secondary* containment.	Immediately
	AND	
		(continued)

1 2 1 2

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E F (continued) 2 ⟨3.7.P.3⟩	E F.2	Suspend CORE ALTERATIONS.	Immediately
	AND		
	E F.3	Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
⟨4.7.P.1⟩	SR 3.6.4.3.1	Operate each SGT subsystem for ≥ [10] continuous hours [with heaters operating].	31 days	1
⟨4.7.P.2⟩ ⟨DOC A.2⟩	SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP	
⟨4.7.P.4.b⟩	SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	~~[18]~~ 24 months	1
	[SR 3.6.4.3.4	Verify each SGT filter cooler bypass damper can be opened and the fan started.	[18] months]	3

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.6.4.3 - STANDBY GAS TREATMENT (SGT) SYSTEM

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. ISTS 3.6.4.3 ACTION D, which requires an LCO 3.0.3 entry when both SGT subsystems are inoperable in MODE 1, 2, or 3, has been replaced with two Actions: ITS 3.6.4.3 ACTION D, which allows 1 hour to restore one SGT subsystem when both are inoperable, and ITS 3.6.4.3 ACTION E, which requires a plant shutdown to MODE 4 when the requirements of ACTION D are not met. These two ACTIONS are consistent with the CTS, and were recently approved by the NRC in Amendments 171 and 167. Due to this change, the following Action was renumbered.

3. The bracketed requirement is deleted. The SGT subsystem arrangement to ensure the removal of decay heat from an idle train consists of a flow path containing an automatically actuated damper, in each subsystem, and a common, locked open, electrically disconnected crosstie valve. Operability of the automatic damper is verified within the performance of ITS SR 3.6.4.3.3. Operation of the common crosstie valve is controlled in accordance with plant procedures. This change is consistent with the current licensing basis.

All changes are [1] unless otherwise indicated

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.1 Primary Containment

BASES

BACKGROUND

The function of the primary (Loss of Coolant) containment is to isolate and contain fission products released from the Reactor Primary System following a Design Basis Accident (DBA) (LOCA) and to confine the postulated release of radioactive material. [6] The primary containment consists of a steel ~~lined~~ (pressure vessel enclosed in), reinforced concrete ~~vessel~~, ~~which~~ (a drywell, which is) surrounds the Reactor Primary System (and a suppression chamber, which is a steel torus-shaped pressure vessel, connected by vent pipes. The primary containment) and provides an essentially leak tight barrier against an uncontrolled release of radioactive material to the environment.

The isolation devices for the penetrations in the primary 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

   2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)";

b. The primary containment air lock is OPERABLE, except as provided in LCO 3.6.1.2, "Primary Containment Air Lock";

c. All equipment hatches are closed~~;~~ and sealed. [2] and

d. ~~The pressurized sealing mechanism associated with a penetration is OPERABLE, except as provided in LCO 3.6.1.[ ].~~

d. The sealing mechanism associated with each primary containment penetration (e.g., welds, bellows, or O-rings) is OPERABLE. [1]

This Specification ensures that the performance of the primary containment, in the event of a DBA (Design Basis Accident) [2], meets the assumptions used in the safety analyses of References 1 and 2. SR 3.6.1.1.1 leakage rate requirements are in

(continued)

BASES

BACKGROUND (continued)

conformance with 10 CFR 50, Appendix J, Option B (Ref. 3), as modified by approved exemptions. 3

APPLICABLE SAFETY ANALYSES

The safety design basis for the primary containment is that it must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage.

Analytical methods and assumptions involving the primary containment are presented in References 1 and 2. The safety analyses assume a nonmechanistic fission product release following a DBA, which forms the basis for determination of offsite doses. The fission product release is, in turn, based on an assumed leakage rate from the primary containment. OPERABILITY of the primary containment ensures that the leakage rate assumed in the safety analyses is not exceeded.

4 1.0 4 design basis LOCA 1

The maximum allowable leakage rate for the primary containment (L_a) is [1.2]% by weight of the containment air per 24 hours at the ~~maximum~~ calculated peak containment pressure (P_a) of [57.5] 48 psig, ~~or [0.84]% by weight of the containment air per 24 hours at the reduced pressure of P_t ([28.8] psig)~~ ~~(Ref. 1)~~ (Ref. 2). 5 1

Primary containment satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). 1

LCO

Primary 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,~~ Primary Containment Leakage Rate Testing Program leakage test. At this time, ~~the combined Type B and C~~ applicable leakage limits must be met ~~< 0.6 L_a, and the overall Type A leakage must be < 0.75 L_a~~. In addition, the leakage from the drywell to suppression chamber must be limited to ensure the primary containment pressure and temperature do not exceed design limits. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is 3

(continued)

BASES

LCO (continued)

structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

Individual leakage rates specified for the primary containment air lock are addressed in LCO 3.6.1.2.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, primary containment is not required to be OPERABLE in MODES 4 and 5 to prevent leakage of radioactive material from primary containment.

ACTIONS

A.1

In the event primary containment is inoperable, primary 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 primary containment OPERABILITY during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring primary containment OPERABILITY) occurring during periods where primary containment is inoperable is minimal.

B.1 and B.2

If primary 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 12 hours and to MODE 4 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.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.1

the Primary Containment Leakage Rate Testing Program 3

Maintaining the primary containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of ~~10 CFR 50, Appendix J (Ref. 3), as modified by approved exemptions~~. Failure to meet air lock leakage ~~testing~~ limit (SR 3.6.1.2.1), ~~[secondary containment bypass leakage (SR 3.6.1.3.12),]~~ ~~[resilient seal primary containment purge valve leakage testing (SR 3.6.1.3.7),]~~ or main steam isolation valve leakage (SR 3.6.1.3.~~13~~ 10) does not necessarily result in a failure of this SR. The impact of the failure to meet these SRs must be evaluated against the Type A, B, and C acceptance criteria of ~~10 CFR 50, Appendix J, as modified by approved exemptions (Ref. 3)~~. # As left leakage prior to the first startup after performing a required ~~10 CFR 50, Appendix J~~ leakage test is required to be < 0.6 L_a for combined Type B and C leakage, and ≤ 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 ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. The Frequency is required by the ~~10 CFR 50, Appendix J (Ref. 3), as modified by approved exemptions~~. ~~Thus, SR 3.0.2 (which allows Frequency extensions) does not apply.~~

3 6 3 10 3 6 1 3

Primary Containment Leakage Rate Testing Program

SR 3.6.1.1.2

3

Insert SR 3.6.1.1.2

~~Maintaining the pressure suppression function of primary containment requires limiting the leakage from the drywell to the suppression chamber. Thus, if an event were to occur that pressurized the drywell, the steam would be directed through the downcomers into the suppression pool. This SR measures drywell to suppression chamber differential pressure during a [10] minute period to ensure that the leakage paths that would bypass the suppression pool are within allowable limits.~~

~~Satisfactory performance of this SR can be achieved by establishing a known differential pressure between the drywell and the suppression chamber and verifying that the pressure in either the suppression chamber or the drywell does not change by more than [0.25] inch of water per minute over a 10 minute period. The leakage test is performed every [18 months]. The [18 month] Frequency was developed~~

(continued)

3

Insert SR 3.6.1.1.2

SR 3.6.1.1.2

The analyses results in Reference 4 are based on a maximum drywell-to-suppression chamber bypass leakage. This Surveillance ensures that the actual bypass leakage is less than or equal to the acceptable $A/\sqrt{k}$ design value of 0.18 ft^2 assumed in the safety analysis. For example, with a typical loss factor of 3 or greater, the maximum allowable leakage area would be approximately 0.3 ft^2, corresponding to a 8-in line size.

As left bypass leakage, prior to the first startup after performing a required bypass leakage test, is required to be $\leq$ 2% of the drywell-to-suppression chamber bypass leakage limit. At all other times between required leakage rate tests, the acceptance criteria is based on design $A/\sqrt{k}$. At the design $A/\sqrt{k}$ the containment temperature and pressurization response are bounded by the assumptions of the safety analysis. The leakage test is performed every 24 months, consistent with the difficulty of performing the test, risk of high radiation exposure, and the remote possibility of a component failure that is not identified by some other drywell or primary containment SR.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.2 (continued)

~~considering it is prudent that this Surveillance be performed during a unit outage and also in view of the fact that component failures that might have affected this test are identified by other primary containment SRs. Two consecutive test failures, however, would indicate unexpected primary containment degradation; in this event, as the Note indicates, increasing the Frequency to once every [9 months] is required until the situation is remedied as evidenced by passing two consecutive tests.~~ 3

REFERENCES

1. U FSAR, Section ~~[6.2]~~ .1 . 1
2. FSAR, Section ~~[15.1.39]~~. 15.6.5 4
3. 10 CFR 50, Appendix J. Option B 1
4. UFSAR, Section 6.2.1.2.4.1 1

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.1.1 - PRIMARY CONTAINMENT

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. This bracketed requirement/information has been deleted because it is not applicable to Quad Cities 1 and 2.

3. Changes have been made to reflect those changes made to the Specification.

4. The brackets have been removed and the proper plant specific information/value has been provided.

5. The alternate allowance is not included in the Specifications and therefore has been deleted.

6. Editorial change made for enhanced clarity.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.2 Primary Containment Air Lock

BASES

BACKGROUND

One double door primary containment air lock has been built into the primary containment to provide personnel access to the drywell and to provide primary containment isolation during the process of personnel entering and exiting the drywell. The air lock is designed to withstand the same loads, temperatures, and peak design internal and external pressures as the primary containment (Ref. 1). As part of the primary containment, the air lock limits the release of radioactive material to the environment during normal unit operation and through a range of transients and accidents up to and including postulated Design Basis Accidents (DBAs).

Each air lock door has been designed and tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a DBA in primary containment. 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 primary containment internal pressure results in increased sealing force on each door). a [1]

Each air lock is nominally a right circular cylinder, approximately 10 ft in diameter, with doors at each end that are interlocked to prevent simultaneous opening. The air lock is provided with ~~limit switches~~ gear-driven position indicators [1] on both doors that provide ~~control room~~ local [1] indication of door position. ~~Additionally, control room indication is provided to alert the operator whenever an air lock interlock mechanism is defeated.~~ [2] During periods when primary containment is not required to be OPERABLE, the air lock interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent primary containment entry is necessary. Under some conditions as allowed by this LCO, the primary containment may be accessed through the air lock, when the interlock mechanism has failed, by manually performing the interlock function.

The primary containment air lock forms part of the primary containment pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining primary

(continued)

BASES

BACKGROUND (continued)

containment leakage rate to within limits 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 FSAR safety analysis. [3]

APPLICABLE SAFETY ANALYSES

[1]
design basis LOCA

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (L_a) of 1.2% 1.0 [1] by weight of the containment air mass per 24 hours at the ~~calculated~~ ~~maximum~~ peak containment pressure (P_a) of ~~57.5~~ 48 psig (Ref. 2). [1] This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

Primary containment air lock OPERABILITY is also required to minimize the amount of fission product gases that may escape primary containment through the air lock and contaminate and pressurize the secondary containment.

The primary containment air lock satisfies Criterion 3 of [1] 10 CFR 50.36(c)(2)(ii) ~~the NRC Policy Statement~~. [1]

LCO

As part of the primary containment pressure boundary [3], the air lock's [4] safety function is related to control of containment leakage rates [4] following a DBA. Thus, the air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

The primary containment 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 to be opened at a time. This provision ensures that a gross breach of primary containment does not exist when primary containment is required to be

(continued)

BASES

the [4]

LCO (continued)

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 ~~and~~ exit from primary containment.

or [3]

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the primary containment air lock is not required to be OPERABLE in MODES 4 and 5 to prevent leakage of radioactive material from primary containment.

ACTIONS

The ACTIONS are modified by Note 1, which allows entry and exit to perform repairs of the affected air lock component. If the outer door is inoperable, then it may be easily accessed to repair. If the inner door is the one that is inoperable, however, then a short time exists when the containment boundary is not intact (during access through the outer door). The ~~ability~~ to open the OPERABLE door, even if it means the primary containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the primary containment during the short time in which the OPERABLE door is expected to be open. ~~The OPERABLE door must be immediately closed after each entry and exit.~~

[3] OPERABLE

allowance [3]

[3] The required administrative controls consist of stationing a dedicated individual to assure closure of the OPERABLE door except during entry and exit, and to assure the OPERABLE door is relocked after completion of the containment entry and exit.

[3]

The ACTIONS are modified by a second Note, which ensures appropriate remedial measures are taken when necessary. Pursuant to LCO 3.0.6, actions are not required, even if primary containment is exceeding ~~its~~ leakage ~~limit~~. Therefore, the Note is added to require ACTIONS for LCO 3.6.1.1, "Primary Containment," to be taken in this event.

if air lock leakage results in exceeding overall containment leakage rate acceptance criteria

L_a [3]

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

With one primary containment air lock door inoperable, the OPERABLE door must be verified closed (Required Action A.1) in the air lock. This ensures that a leak tight primary

(continued)

BASES

ACTIONS

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

containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1.1, which requires that primary containment be restored to OPERABLE status within 1 hour.

In addition, the air lock penetration must be isolated by locking closed the 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 that the OPERABLE door is being maintained closed.

3

penetration

Required Action A.3 ensures that the air lock ~~with an inoperable door~~ has been isolated by the use of a locked closed OPERABLE air lock door. This ensures that an acceptable primary 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 or areas with limited access due to inerting and allows these doors to be verified locked closed by use of administrative controls. Allowing verification by administrative controls 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.

3 given 3

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the air lock are inoperable. With both doors in the 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. Primary containment entry may be required to perform Technical Specifications (TS) Surveillances and Required

3 4

(continued)

BASES

ACTIONS

The required administrative controls consist of stationing a dedicated individual to assure closure of the OPERABLE door except during entry and exit, and to assure the OPERABLE door is relocked after completion of the containment entry and exit.

3

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

5

Actions, as well as other activities on equipment inside primary 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-related activities) if the primary 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 primary containment during the short time that the OPERABLE door is expected to be open. 5

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

With an air lock interlock mechanism inoperable, 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 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the air lock are inoperable. With both doors in the 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 primary 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 or areas with limited access due to inerting and that allows these doors to be verified locked closed by use of administrative controls. Allowing verification by administrative controls 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.

(continued)

BASES

ACTIONS (continued)

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

If the air lock is inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be immediately initiated to evaluate containment overall leakage rates using current air lock leakage test results. An evaluation is acceptable since it is overly conservative to immediately declare the primary 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)~~, primary containment remains OPERABLE, yet only 1 hour (according to LCO 3.6.1.1) would be provided to restore the air lock door to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits. 1

Required Action C.2 requires that one door in the primary containment air lock must be verified closed. This action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.1.1, which require that primary containment be restored to OPERABLE status within 1 hour.

(Required Action C.3) 3

Additionally, the air lock must be restored to OPERABLE status within 24 hours. The 24 hour Completion Time is reasonable for restoring an inoperable air lock to OPERABLE status considering that at least one door is maintained closed in the air lock.

D.1 and D.2

If the inoperable primary containment air lock cannot be restored to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 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.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.1.2.1

the 3

3

Maintaining primary containment air locks OPERABLE requires compliance with the leakage rate test requirements of ~~10 CFR 50, Appendix J (Ref. 2), as modified by approved exemptions~~. This SR reflects the leakage rate testing requirements with respect to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and primary containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall primary containment leakage rate. The Frequency is required by ~~10 CFR 50, Appendix J (Ref. 2), as modified by approved exemptions~~. ~~Thus, SR 3.0.2 (which allows Frequency extensions) does not apply.~~

6

the Primary Containment Leakage Rate Testing Program

7

6

two s. Note 1

The SR has been modified by ~~a~~ Note ~~that~~ 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.

6

Insert SR 3.6.1.2.1

6

SR 3.6.1.2.2

The air lock interlock mechanism is designed to prevent simultaneous opening of both doors in the air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident primary containment pressure, closure of either door will support primary containment OPERABILITY. Thus, the interlock feature supports primary 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 will function as designed and that simultaneous inner and outer door opening will not inadvertently occur. ~~Due to the purely mechanical nature of this interlock, and given that~~ the interlock ~~mechanism~~ is ~~only~~ challenged when primary ~~containment is entered~~, this test is only required to be performed ~~upon entering primary containment, but is not required more frequently than 184 days when primary containment is de-inerted~~. The ~~184 day~~ Frequency is based on engineering judgment and is considered adequate ~~in view of other administrative controls~~

8

Insert SR 3.6.1.2.2-1

4 the

TSTF-17

not normally

air lock door

4

Insert SR 3.6.1.2.2-2

24 month

Insert SR 3.6.1.2.2-3

(continued)

6

Insert SR 3.6.1.2.1

Note 2 has been added to this SR, requiring the results to be evaluated against the acceptance criteria which are applicable to SR 3.6.1.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Types B and C primary containment leakage rate.

TSTF-17

Insert SR 3.6.1.2.2-1

used for entry and exit (procedures require strict adherence to single door opening)

TSTF-17

Insert SR 3.6.1.2.2-2

every 24 months. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, and the potential for loss of primary containment OPERABILITY if the Surveillance were performed with the reactor at power. The 24 month Frequency for the interlock is justified based on generic operating experience. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency. 8

TSTF-17

Insert SR 3.6.1.2.2-3

given that the interlock is not challenged during the use of the air lock. 4

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.2.2 (continued)

[such as indications of interlock mechanism status, available to operations personnel]. TSTF-17

REFERENCES

u 1. FSAR, Section [3.8.2.8.2.2]. 6.2.1.2.1

1 2. 10 CFR 50, Appendix J. 7

u 2 3. FSAR, Section [6.2]. 15.6.5

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.1.2 - PRIMARY CONTAINMENT AIR LOCK

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. This bracketed requirement/information has been deleted because it is not applicable to Quad Cities 1 and 2.

3. Editorial change made for enhanced clarity.

4. Typographical/grammatical error corrected.

5. These words have been deleted since the primary containment may need to be entered for reasons related to TS that are not specifically on "equipment." This could include sampling and inspections. The intent has not changed in that it must still be related to TS.

6. Changes have been made to reflect those changes made to the Specification.

7. The brackets have been removed and the proper plant specific information/value has been provided.

8. The change has been made for consistency with similar phrases in other parts of the Bases. The phrase "Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency" is generally used to describe why a 24 month Frequency is acceptable, and in almost all cases, the current Frequency in the CTS is 18 months. For this Surveillance, the CTS Frequency could be as long as 18 months, therefore using these words is consistent with similar phrases in other parts of the Bases.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.3 Primary Containment Isolation Valves (PCIVs)

BASES

BACKGROUND

The function of the PCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) to within limits. Primary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a DBA.

The OPERABILITY requirements for PCIVs help ensure that an adequate primary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. Therefore, the OPERABILITY requirements provide assurance that primary containment function assumed in the safety analyses will be maintained. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges 1 (which include plugs and caps as listed in Reference 1) and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration 1 , except for penetrations isolated by excess flow check valves, 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.

The reactor building-to-suppression chamber vacuum breakers serve a dual function, one of which is primary containment isolation. However, since the other safety function of the vacuum breakers would not be available if the normal PCIV actions were taken, the PCIV OPERABILITY requirements are not applicable to the reactor building-to-suppression chamber vacuum breakers valves. Similar surveillance requirements in the LCO for reactor building-to-suppression chamber vacuum breakers provide assurance that the isolation capability is available without conflicting with the vacuum relief function.

(continued)

All changes are [1] unless otherwise indicated

except for torus purge valve 1601-56. This valve is normally open for pressure control. This is acceptable since this valve and other vent and purge valves are designed to automatically close on LOCA conditions.

BASES

valves — 2, 6 and — [2] — valves — [2] — vent and

BACKGROUND (continued) [1]

The primary containment purge ~~lines~~ are ~~[18]~~ inches in diameter; vent ~~lines~~ are ~~[18]~~ inches in diameter. The ~~[18]~~ inch primary containment purge valves are normally maintained closed in MODES 1, 2, and 3 to ensure the primary containment boundary is maintained. The isolation valves on the ~~[18]~~ inch vent lines have ~~[2]~~ inch bypass lines around them for use during normal reactor operation. ~~Two additional redundant excess flow isolation dampers are provided on the vent line upstream of the Standby Gas Treatment (SGT) System filter trains. These isolation dampers, together with the PCIVs,~~ will prevent high pressure ~~from reaching the SGT System filter trains~~ in the unlikely event of a loss of coolant accident (LOCA) during venting. ~~Closure of the excess flow isolation dampers will not prevent the SGT System from performing its design function (that is, to maintain a negative pressure in the secondary containment). To ensure that a vent path is available, a [2] inch bypass line is provided around the dampers.~~

from the suppression chamber and drywell

[2]

Use of the 2 inch vent valves

Standby Gas Treatment

and the Reactor Building Ventilation system

APPLICABLE SAFETY ANALYSES

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

(1)

The DBAs that result in a release of radioactive material ~~within primary containment~~ are a LOCA ~~and~~ a main steam line break (MSLB). In the analysis for each of these accidents, it is assumed that PCIVs are either closed or close within the required isolation times following event initiation. This ensures that potential paths to the environment through PCIVs (including primary containment purge valves) are minimized. Of the events analyzed in Reference ~~4~~, the MSLB is the most limiting event due to radiological consequences. The closure time of the main steam isolation valves (MSIVs) is a significant variable from a radiological standpoint. The MSIVs are required to close within 3 to 5 seconds since the 5 second closure time is assumed in the analysis. ~~The safety analyses assume that the purge valves were closed at event initiation.~~ Likewise, it is assumed that the primary

for which the consequences are mitigated by PCIVs

(Refs 2 and 3, respectively)

(4)

LOCA

(Ref. 3)

the 3 second closure time is assumed in the MSIV closure (the most severe overpressurization transient) analysis (Ref. 5) and

(continued)

BASES

isolates 1

APPLICABLE SAFETY ANALYSES (continued)

containment ~~is isolated~~ such that release of fission products to the environment is controlled.

The DBA analysis assumes that ~~within 60 seconds of the accident,~~ isolation of the primary containment is complete and leakage is terminated, except for the maximum allowable leakage rate, L. ~~The primary containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and PCIV stroke times.~~ 3

, prior to fuel damage 3

[The single failure criterion ~~required to be imposed in the conduct of unit safety analyses~~ was considered in the original design of the primary containment vent and purge valves. Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred.] 2

~~[The primary containment 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, and 3. In this case, the single failure criterion remains applicable to the primary containment purge valve due to failure in the control circuit associated with each valve. The primary containment purge valve design precludes a single failure from compromising the primary containment boundary as long as the system is operated in accordance with this LCO.]~~ 4

PCIVs satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). 1

LCO

PCIVs form a part of the primary containment boundary. The PCIV safety function is related to minimizing the loss of reactor coolant inventory and establishing the primary containment boundary during a DBA.

The power operated, automatic isolation valves are required to have isolation times within limits and actuate on an automatic isolation signal. ~~The [18] inch purge valves must be maintained sealed closed [or blocked to prevent full opening].~~ 1 6 While the reactor building-to-suppression chamber vacuum breakers isolate primary containment penetrations, they are excluded from this Specification. Controls on their isolation function are adequately addressed in LCO

(continued)

Normally closed automatic PCIVs which are required by design (e.g., to meet 10 CFR 50 Appendix R requirements) to be de-activated and closed are considered OPERABLE when the valves are closed and de-activated. 12

BASES

the Technical Requirements Manual (Ref. 1)

LCO (continued)

3.6.1.7, "Reactor Building-to-Suppression Chamber Vacuum Breakers." The valves covered by this LCO are listed with their associated stroke times in ~~Reference 2~~.

12 manual

and blind flanges are in place 12 the under

The normally closed PCIVs are considered OPERABLE when ~~manual~~ valves are closed or open ~~in accordance with appropriate~~ administrative controls, ~~automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact~~. These passive isolation valves and devices are those listed in Reference ~~2~~ 1. 12 5

~~Purge valves with resilient seals, secondary bypass valves, MSIVs, and hydrostatically tested valves~~ must meet additional leakage rate requirements. Other PCIV leakage rates are addressed by LCO 3.6.1.1, "Primary Containment," as Type B or C testing. 6

This LCO provides assurance that the PCIVs will perform their designed safety functions to minimize the loss of reactor coolant inventory and establish the primary containment boundary during accidents.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, most PCIVs are not required to be OPERABLE ~~and the primary containment purge valves are not required to be sealed closed~~ in MODES 4 and 5. Certain valves, however, are required to be OPERABLE to prevent inadvertent reactor vessel draindown. These valves are those whose associated instrumentation is required to be OPERABLE per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation." (This does not include the valves that isolate the associated instrumentation.) 6

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow path(s) ~~(except for purge valve flow path(s))~~ to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous 6

(continued)

BASES

ACTIONS (continued)

communication with the control room. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated. ~~Due to the size of the primary containment purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, the penetration flow path containing these valves is not allowed to 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 3.6.1.3.1.~~ 6

A second Note has been added to provide clarification that, for the purpose of 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 PCIV. Complying with the Required Actions may allow for continued operation, and subsequent inoperable PCIVs are governed by subsequent Condition entry and application of associated Required Actions.

The ACTIONS are modified by Notes 3 and 4. Note 3 ensures that appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by an inoperable PCIV (e.g., an Emergency Core Cooling System subsystem is inoperable due to a failed open test return valve). Note 4 ensures appropriate remedial actions are taken when the primary containment leakage limits are exceeded. Pursuant to LCO 3.0.6, these actions are not required even when the associated LCO is not met. Therefore, Notes 3 and 4 are added to require the proper actions be taken.

A.1 and A.2

With one or more penetration MSIV flow paths with rate one PCIV inoperable [except for ~~purge valve~~ leakage not within limit], the affected penetration flow paths 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, a blind flange, and a check valve with flow through the valve secured. For a penetration isolated in accordance with Required Action A.1, 2

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

the device used to isolate the penetration should be the closest available valve to the primary containment. The Required Action must be completed within the 4 hour Completion Time (8 hours for main steam lines). The Completion Time of 4 hours is reasonable considering the time required to isolate the penetration and the relative importance of supporting primary containment OPERABILITY during MODES 1, 2, and 3. For main steam lines, an 8 hour Completion Time is allowed. The Completion Time of 8 hours for the main steam lines allows a period of time to restore the MSIVs to OPERABLE status given the fact that MSIV closure will result in isolation of the main steam line(s) and a potential for plant shutdown.

For affected penetrations that have been isolated in accordance with Required Action A.1, the affected penetration flow path(s) must be verified to be isolated on a periodic basis. This is necessary to ensure that primary 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 that those devices outside primary [5] containment and capable of potentially being mispositioned are in the correct position. The Completion Time of "once per 31 days for isolation devices outside primary containment" is appropriate because the devices are operated under administrative controls and the probability of their misalignment is low. For the devices inside primary containment, the time period specified "prior to entering MODE 2 or 3 from MODE 4, ~~if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days~~ [5]" is based on engineering judgment and is considered reasonable in view of the inaccessibility of the devices and the existence of [7] other administrative controls ensuring that device misalignment is an unlikely possibility.

Condition A is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with two or more [6] PCIVs. For penetration flow paths with one PCIV, Condition C provides the appropriate Required Actions.

Required Action A.2 is modified by ~~a~~ two Note(s) ~~that~~ Note 1 applies to isolation devices located in high radiation areas, [B] and TSTF-269

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

allows them 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 low.

Insert A.1 and A.2

TSTF-269

9

9

B.1

With one or more penetration flow paths with two or more PCIVs inoperable except for MSIV leakage, either the inoperable PCIVs must be restored to OPERABLE status or 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 3.6.1.1.

6

6

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two or more PCIVs. For penetration flow paths with one PCIV, Condition C provides the appropriate Required Actions.

6

C.1 and C.2

With one or more penetration flow paths with one PCIV 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) hours ~~Completion Time~~ 4 hours except for excess flow check valve (EFCV) lines and penetrations with a closed system and 72 hours for EFCVs and penetrations with a closed system. The Completion Time of ~~(4) hours~~ 72 hours is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of supporting primary containment OPERABILITY during

11

5 11

72

TSTF-30

TSTF-30

5

(continued)

The Completion Time of 4 hours for valves other than EFCVs and in penetrations with a closed system is reasonable considering the time required to isolate the penetration and the relative importance of supporting primary containment OPERABILITY during MODES 1, 2, and 3.

for a penetration with a closed system

11

TSTF-269

Insert A.1 and A.2

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.

BASES

ACTIONS

C.1 and C.2 (continued)

72 [11] For EFCVs [11] also

MODES 1, 2, and 3. The Completion Time of 12 hours is reasonable considering the instrument and the small pipe diameter of penetration (hence, reliability) to act as a penetration isolation boundary and the small pipe diameter of the affected penetrations. In the event the affected penetration flow path is isolated in accordance with Required Action C.1, the affected penetration must be verified to be isolated on a periodic basis. This is necessary to ensure that primary containment penetrations required to be isolated following an accident are isolated. The Completion Time of once per 31 days ~~for verifying each affected penetration is isolated~~ is appropriate because the ~~valves~~ devices [7] are operated under administrative controls and the probability of their misalignment is low.

The closed system must meet the requirements of Reference 6.

TSTF-30

This Required Action does not require any testing or valve manipulation. Rather, it involves verification that these devices outside containment and capable of potentially being mispositioned are in the correct position

[10]

[7]

Condition C is modified by a Note indicating that this Condition is only applicable to penetration flow paths with only one or more [6] PCIV. For penetration flow paths with two PCIVs, Conditions A and B provide the appropriate Required Actions.

This Note is necessary since this Condition is written specifically to address those penetrations with a single PCIV.

[5]

Required Action C.2 is modified by ~~a~~ two Note~~s~~ ~~that~~ Note 1 applies to ~~valves and blind flanges~~ isolation devices [5] located in high radiation areas and allows them 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 valves~~, once they have been verified to be in the proper position, is low.

TSTF-269

Insert ACTION C-2 TSTF-269

D.1

With ~~the secondary containment bypass leakage rate or~~ MSIV leakage rate not within limit, the assumptions of the safety analysis may not be met. Therefore, the leakage must be restored to within limit within (8) hours. Restoration can be accomplished by isolating the penetration that caused the limit to be exceeded by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. When a penetration is isolated, the leakage rate for the isolated penetration is assumed to be the actual pathway leakage through the isolation device. If two isolation devices are used to isolate the penetration, the leakage rate is assumed

[6]

(continued)

TSTF-269

Insert ACTION C-2

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.

allows a reasonable period of time to restore MSIV leakage and is acceptable given the fact that MSIV closure will result in isolation of the main steam line(s) and a potential for plant shutdown

BASES

ACTIONS

D.1 (continued) 8

to be the lesser actual pathway leakage of the two devices. The 8 hour Completion Time ~~is reasonable considering the time required to restore the leakage by isolating the penetration and the relative importance of secondary containment bypass leakage to the overall containment function.~~ 6

~~E.1, E.2, and E.3~~

~~In the event one or more containment purge valves are not within the purge valve leakage limits, purge valve leakage must be restored to within limits or the affected penetration 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]. If a purge valve with resilient seals is utilized to satisfy Required Action E.1, it must have been demonstrated to meet the leakage requirements of SR 3.6.1.3.7. The specified Completion Time is reasonable, considering that one containment purge valve remains closed so that a gross breach of containment does not exist.~~ 6

~~In accordance with Required Action E.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 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 2 or 3 from MODE 4 if 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.~~

(continued)

BASES

ACTIONS

6

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

For the containment purge valve with resilient seal that is isolated in accordance with Required Action E.1, SR 3.6.1.3.7 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 3.6.1.3.7 is 184 days. 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 to be acceptable based on operating experience.

6
TSTF-269 changes not shown

E ~~F~~.1 and ~~F~~.2 6

If any Required Action and associated Completion Time cannot be met in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 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.

F 6 6

~~G~~.1, ~~H.1, I.1,~~ and ~~G~~.2 F

6
for PCIV(s) required OPERABLE in MODE 4 or 5

If any Required Action and associated Completion Time cannot be met, the unit must be placed in a condition in which the LCO does not apply. ~~If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe condition. Also, if applicable,~~ action must be immediately initiated to suspend operations with a potential for draining the reactor vessel (OPDRVs) to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended ~~and valve(s) are restored to OPERABLE status~~. If suspending an OPDRV would result in closing the residual heat removal (RHR) shutdown cooling isolation valves, an alternative Required Action is provided to

2

7

(continued)

BASES

6 F

ACTIONS

F

E.1, H.1, I.1, and J.2 (continued)

immediately initiate action to restore the valve(s) to OPERABLE status. This allows RHR to remain in service while actions are being taken to restore the valve.

shutdown cooling

2 7

SURVEILLANCE REQUIREMENTS

6

SR 3.6.1.3.1

10

TSTF-30 changes not shown

Each [18] inch primary containment purge valve is required to be verified sealed closed at 31 day intervals. This SR is designed to ensure that a gross breach of primary containment is not caused by an inadvertent or spurious opening of a primary 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. Primary containment purge valves that are sealed closed must have motive power to the valve operator removed. This can be accomplished by de-energizing the source of electric power or removing the air supply to the valve operator. In this application, the term "sealed" has no connotation of leak tightness. The 31 day Frequency is a result of an NRC initiative, Generic Issue B-24 (Ref. 4), related to primary containment purge valve use during unit operations.

This SR allows a valve that is open under administrative controls to not meet the SR during the time the valve is open. Opening a purge valve under administrative controls is restricted to one valve in a penetration flow path at a given time (refer to discussion for Note 1 of the ACTIONS) in order to effect repairs to that valve. This allows one purge valve to be opened without resulting in a failure of the Surveillance and resultant entry into the ACTIONS for this purge valve, provided the stated restrictions are met. Condition E must be entered during this allowance, and the valve opened only as necessary for effecting repairs. Each purge valve in the penetration flow path may be alternately opened, provided one remains sealed closed, if necessary, to complete repairs on the penetration.

The SR is modified by a Note stating that primary containment purge valves are only required to be sealed closed in MODES 1, 2, and 3. If a LOCA inside primary

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.1 (continued)

containment occurs in these MODES, the purge valves may not be capable of closing before the pressure pulse affects systems downstream of the purge valves or the release of radioactive material will exceed limits prior to the closing of the purge valves. At other times when the purge valves are required to be capable of closing (e.g., during handling of irradiated fuel), pressurization concerns are not present and the purge valves are allowed to be open.

6

The torus purge valve, 1601-56, is normally open for pressure control, therefore the valve is excluded from this SR. However this is acceptable since this valve is designed to automatically close on LOCA conditions.

1

SR 3.6.1.3.2 ① 6 2

vent or 18 inch 5 ed ed vent and 8 2

This SR ensures that the primary containment purge 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 also modified by a Note (Note 1), stating that primary containment purge valves are only required to be closed in MODES 1, 2, and 3. If a LOCA inside primary containment occurs in these MODES, the purge valves may not be capable of closing before the pressure pulse affects systems downstream of the purge valves, or the release of radioactive material will exceed limits prior to the purge valves closing. At other times when the purge valves are required to be capable of closing (e.g., during handling of irradiated fuel), pressurization concerns are not present and the purge valves are allowed to be open.] The SR is modified by a Note (Note 2) stating that the SR is not required to be met when the purge valves are open for the stated reasons. The Note states that these valves may be opened for inerting, de-inerting, pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open. The [18] inch purge 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 PCIV requirements discussed in SR 3.6.1.3.8.

6 2 4 2 ventor

Vent and 2

② 6

provided the drywell vent and purge valves and their associated suppression chamber vent and purge valves are closed 6

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.3 (2) 6

This SR verifies that each primary containment isolation manual valve and blind flange that is located outside primary containment and is 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 primary containment boundary is within design limits.

and is not locked, sealed, or otherwise secured

TSTF-45

(3) 7

This SR does not require any testing or valve manipulation. Rather, it involves verification that those PCIVs outside primary containment, and capable of being mispositioned, are in the correct position. Since verification of ~~valve~~ 5 position for PCIVs outside primary containment is relatively easy, the 31 day Frequency was chosen to provide added assurance that the PCIVs are in the correct positions.

TSTF-45

This SR does not apply to valves and blind flanges that are locked, sealed, or otherwise secured in the closed position, since these valves were verified to be in the correct position upon locking, sealing, or securing.

7

Two Notes have been added to this SR. The first Note allows valves and blind flanges located in high radiation areas to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable since ~~the primary containment is inerted and~~ access to these areas is typically restricted ~~during MODES 1, 2, and 3~~ for ALARA reasons. Therefore, the probability of misalignment of these PCIVs, once they have been verified to be in the proper position, is low. 7 A second Note has been included to clarify that PCIVs ~~that are~~ 7 open under administrative controls are not required to meet the SR during the time that the PCIVs are open.

TSTF-45

These controls consists of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated.

7

SR 3.6.1.3.4 (3) 6

and not locked, sealed, or otherwise secured

This SR verifies that each primary containment manual isolation valve and blind flange ~~that is~~ located inside primary containment and is required to be closed during accident conditions is closed. 7 The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the primary containment boundary is within design limits. For PCIVs inside primary containment, the Frequency ~~defined~~ of 7 ~~as~~ "prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4, if not performed within the previous 92 days" is appropriate since these PCIVs are operated under administrative controls and the probability of their misalignment is low. 7

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.6 (continued)

3 6

Two Notes have been added to this SR. The first Note allows valves and blind flanges located in high radiation areas to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable since the primary containment is inerted and access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these PCIVs, once they have been verified to be in their proper position, is low. A second Note has been included to clarify that PCIVs that are open under administrative controls are not required to meet the SR during the time that the PCIVs are open.

These controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way the penetration can be rapidly isolated when a need for primary containment isolation is indicated

7

SR 3.6.1.3.9 4 6

The traversing incore probe (TIP) shear isolation valves are actuated by explosive charges. Surveillance of explosive charge continuity provides assurance that TIP valves will actuate when required. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience that has demonstrated the reliability of the explosive charge continuity.

SR 3.6.1.3.8 5 6

9 TSTF-46

Verifying the isolation time of each power operated ~~and each~~ automatic PCIV is within limits is required to demonstrate OPERABILITY. MSIVs may be excluded from this SR since MSIV full closure isolation time is demonstrated by SR 3.6.1.3.8. 6 6 The isolation time test ensures that ~~the~~ each 5 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~~ is in accordance with the requirements of the Inservice Testing Program ~~or 92 days~~. 15 2 2 3

6

~~SR 3.6.1.3.7~~

~~For primary containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J (Ref. 3), is required to ensure~~

(continued)

BASES

6

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.7 (continued)

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 primary containment and the environment), a Frequency of 184 days was established.

Additionally, this SR must be performed once within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that which occurs 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.

The SR is modified by a Note stating that the primary containment purge valves are only required to meet leakage rate testing requirements in MODES 1, 2, and 3. If a LOCA inside primary containment occurs in these MODES, purge valve leakage must be minimized to ensure offsite radiological release is within limits. At other times when the purge valves are required to be capable of closing (e.g., during handling of irradiated fuel), pressurization concerns are not present and the purge valves are not required to meet any specific leakage criteria.

SR 3.6.1.3.~~8~~ 7 6

Verifying that the isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The isolation time test ensures that the MSIV will isolate in a time period that does not exceed the times assumed in the DBA and transient analyses. This ensures that the calculated radiological consequences of these events remain within 10 CFR 100 limits. The Frequency of this SR is ~~(~~in accordance with the requirements of the Inservice Testing Program ~~or 18 months)~~.

1

2

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.9 7 6

Automatic PCIVs close on a primary containment isolation signal to prevent leakage of radioactive material from primary containment following a DBA. This SR ensures that each automatic PCIV will actuate to its isolation position on a primary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in ~~SR 3.3.6.3.7~~ LCO 3.3.6.1, "Primary Containment Isolation Instrumentation," 5 overlaps this SR to provide complete testing of the safety function. The ~~18~~ 24 2 month Frequency was developed considering it is prudent that this Surveillance be performed only during a unit outage since isolation of penetrations would eliminate cooling water flow and disrupt the normal operation of many critical components. Operating experience has shown that these components usually pass this Surveillance when performed at the ~~18~~ 24 2 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.10 8 6

This SR requires a demonstration that each reactor instrumentation line excess flow check valve (EFCV) is OPERABLE by verifying that the valve ~~reduces flow to ≤ 1 gph on a simulated instrument line break~~ actuates to the isolation position on an actual or simulated instrument line break condition 2. This test is performed by blowing down the instrument line during a inservice leak or hydrostatic test and verifying a distinctive "click" when the poppet valve seats or a quick reduction in flow 1. This SR provides assurance that the instrumentation line EFCVs will perform ~~so that predicted radiological consequences will not be exceeded during the postulated instrument line break event evaluated in Reference 6.~~ as designed. 1 The ~~18~~ 24 2 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the ~~18~~ month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.11 9 6

The TIP shear isolation valves are actuated by explosive charges. An in place functional test is not possible with this design. The explosive squib is removed and tested to provide assurance that the valves will actuate when

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.11 9 (continued)

6

required. The replacement charge for the explosive squib shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of the batch successfully fired. The Frequency of 18 24 months on a STAGGERED TEST BASIS is considered adequate given the administrative controls on replacement charges and the frequent checks of circuit continuity (SR 3.6.1.3.6 4).

1

6

6

Other administrative controls, such as those that limit the shelf life and operating life, as applicable, of the explosive charges, must be followed.

1

~~SR 3.6.1.3.12~~

~~This SR ensures that the leakage rate of secondary containment bypass leakage paths is less than the specified leakage rate. This provides assurance that the assumptions in the radiological evaluations of Reference 7 are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves) unless the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. This method of quantifying maximum pathway leakage is only to be used for this SR (i.e., Appendix J maximum pathway leakage limits are to be quantified in accordance with Appendix J). The Frequency is required by 10 CFR 50, Appendix J, as modified by approved exemptions (and therefore, the Frequency extensions of SR 3.0.2 may not be applied), since the testing is an Appendix J, Type C test. This SR simply imposes additional acceptance criteria. Note 1 is added to this SR which states that these valves are only required to meet this leakage limit in MODES 1, 2, and 3. In the other conditions, the Reactor Coolant System is not pressurized and specific primary containment leakage limits are not required.~~

~~[Bypass leakage is considered part of L_a. [Reviewer's Note: Unless specifically exempted].]~~

(continued)

TSTF-30 changes not shown 1

BASES

6

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.~~12~~ 10

3

The analyses in References 2 and ~~5~~ 3 are based on leakage that is less than the specified leakage rate. ~~Leakage through each MSIV must be ≤ [11.5] scfh when tested at ≥ P_t ([28.8] psig). The MSIV leakage rate must be verified to be in accordance with the leakage test requirements of 10 CFR 50, Appendix J (Ref. 3), as modified by approved exemptions.~~ ~~Note 1 is added to this SR which states that these valves are only required to meet this leakage limit in MODES 1, 2, and 3. In the other conditions, the Reactor Coolant System is not pressurized and specific primary containment leakage limits are not required.~~ This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate. The Frequency is required by ~~10 CFR 50, Appendix J, as modified by approved exemptions; thus, SR 3.0.2 (which allows Frequency extensions) does not apply.~~ 1

Insert SR 3.6.1.3.10 6

6

8

6

the Primary Containment Leakage Rate Testing Program

MSIV leakage is considered part of La. 1

6

~~SR 3.6.1.3.14~~

~~Surveillance of hydrostatically tested lines provides assurance that the calculation assumptions of Reference 2 are met. The combined leakage rates must be demonstrated in accordance with the leakage rate test Frequency of 10 CFR 50, Appendix J (Ref. 3), as modified by approved exemptions; thus SR 3.0.2 (which allows Frequency extensions) does not apply.~~

~~[This SR has been modified by a Note that states that these valves are only required to meet the combined leakage rate in MODES 1, 2, and 3, since this is when the Reactor Coolant System is pressurized and primary containment is required. In some instances, the valves are required to be capable of automatically closing during MODES other than MODES 1, 2, and 3. However, specific leakage limits are not applicable in these other MODES or conditions.]~~

6

~~[SR 3.6.1.3.15~~

~~[Reviewer's Note: This SR is only required for those plants with purge valves with resilient seals allowed to be open during [MODE 1, 2, 3, or 4] and having blocking devices that are not permanently installed on the valves.]~~

(continued)

6

Insert SR 3.6.1.3.10

The combined leakage rate for all MSIV leakage paths is ≤ 46 scfh when tested at ≥ 25 psig. The leakage rate of each main steam isolation valve path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves). If both isolation valves in the penetration are closed the actual leakage rate is the lesser leakage rate of the two valves. This method of quantifying leakage is only to be used for this SR (i.e., Appendix J maximum pathway leakage limits are to be quantified in accordance with the Primary Containment Leakage Rate Testing Program).

BASES

6

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.15 (continued)

~~Verifying each [ ] inch primary 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 analysis of References 2 and 6. [The SR is modified by a Note stating that this SR is only required to be met in MODES 1, 2, and 3.] 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.~~

TSTF-30 changes not shown

2

Technical Requirements Manual 1

REFERENCES

1 2 1. UFSAR, ~~Chapter [15]~~ Section 15.6.5.

1 1 2. ~~FSAR, Table [6.2-5]~~.

3. ~~10 CFR 50, Appendix J.~~ 1

4. ~~FSAR, Section [6.2]~~

5. ~~FSAR, Section [15.1.39]~~. 1

3. UFSAR, Section 15.6.4.
4. UFSAR, Chapter 15.
5. UFSAR, Section 5.2.2.2.3. 1

6. UFSAR, Section 6.2.4.1. 1

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.1.3 - PRIMARY CONTAINMENT ISOLATION VALVES

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. This paragraph in the Applicable Safety Analyses Section of Bases 3.6.1.3 has been modified since it is incorrect; neither the DBA analysis nor the IST Program have a specific assumption for closure time of PCIVs. The analysis assumes the valves will close prior to fuel damage, which is not expected for some time. The closure times of the principle PCIVs are currently specified in the UFSAR, and are based upon such factors as valve size and valve operator capability. In addition, the words in SR 3.6.1.3.5 stating that the isolation times are in the IST Program have also been deleted since these times are also located in the UFSAR.

4. This bracketed requirement/information has been deleted because it is not applicable to Quad Cities 1 and 2.

5. These changes have been made for consistency with similar phrases in other parts of the Bases and/or to be consistent with the Specification.

6. Changes have been made to reflect those changes made to the Specification.

7. Editorial change made for enhanced clarity.

8. Typographical/grammatical error corrected.

9. This change was approved to be made in NUREG-1433, Rev. 1 per change package BWR-15, C.5, but apparently was not made. A similar change was made to NUREG-1433, Rev. 1, Bases 3.6.4.2, Required Actions A.1 and A.2.

10. Some of the Bases changes for TSTF-30, Rev. 2, have not been adopted since the SRs/ information is not applicable to Quad Cities 1 and 2.

11. Changes have been made to be consistent with the Specification. These changes are also consistent with TSTF-207, Rev. 3, and TSTF-30, Rev. 3, except when plant specific differences apply or when typographical/consistency errors were noted.

12. The discussion in the LCO section about closed valves is modified. This editorial preference is based on an incomplete and misleading discussion of the valves. This change does not modify the requirements or the interpretation of the requirements.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Drywell Pressure

BASES

BACKGROUND

The drywell pressure is limited during normal operations to preserve the initial conditions assumed in the accident analysis for a Design Basis Accident (DBA) or loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

Primary containment performance is evaluated for the entire spectrum of break sizes for postulated LOCAs (Ref. 1). Among the inputs to the DBA is the initial primary containment internal pressure (Ref. 1). Analyses assume an initial drywell pressure of ~~[0.75~~ psig~~]~~ 1.5 [1]. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell internal pressure does not exceed the maximum allowable of ~~[~~62~~]~~ psig. [1]

The maximum calculated drywell pressure occurs during the reactor blowdown phase of the DBA, which assumes an instantaneous recirculation line break. The calculated peak drywell pressure for this limiting event is ~~[57.5]~~ psig 47 [11] (Ref. 1).

Drywell pressure satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii). [2]

LCO

1.5 [1]

In the event of a DBA, with an initial drywell pressure ≤ ~~[0.75~~ psig~~]~~, the resultant peak drywell accident pressure will be maintained below the drywell design pressure.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining drywell pressure within limits is not required in MODE 4 or 5.

(continued)

BASES (continued)

ACTIONS

A.1

With drywell pressure not within the limit of the LCO, drywell pressure must be restored within 1 hour. The Required Action is necessary to return operation to within the bounds of the primary containment analysis. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1.1, "Primary Containment," which requires that primary containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

If drywell pressure cannot be restored to within the [3] limit 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 12 hours and to MODE 4 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.4.1

Verifying that drywell pressure is within the [3] limit ensures that unit operation remains within the limit assumed in the primary containment analysis. The 12 hour Frequency of this SR was developed, based on operating experience related to trending of drywell 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 drywell pressure condition.

REFERENCES

[2] 1. UFSAR, Section ~~6.2~~ 1.3.2 [1].

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.1.4 - DRYWELL PRESSURE

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

3. Typographical/grammatical error corrected.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.5 Drywell Air Temperature

BASES

BACKGROUND

The drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell coolers remove heat and maintain a suitable environment. The average airspace temperature affects the calculated response to postulated Design Basis Accidents (DBAs). The limitation on the drywell average air temperature was developed as reasonable, based on operating experience. The limitation on drywell air temperature is used in the Reference 1 safety analyses.

APPLICABLE SAFETY ANALYSES

Primary containment performance is evaluated for a spectrum of break sizes for postulated loss of coolant accidents (LOCAs) (Ref. 1). Among the inputs to the design basis analysis is the initial drywell average air temperature (Ref. 1). Analyses assume an initial average drywell air temperature of ~~[125]~~ 150 °F. [1] This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell temperature does not exceed the maximum allowable temperature of ~~[340]~~ 281 °F (Ref. 2). Exceeding this design temperature may result in the degradation of the primary containment structure under accident loads. Equipment inside primary containment required to mitigate the effects of a DBA is designed to operate and be capable of operating under environmental conditions expected for the accident.

Drywell air temperature satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). [2]

LCO

In the event of a DBA, with an initial drywell average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the drywell design temperature. As a result, the ability of primary containment to perform its design function is ensured.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining drywell average air temperature within the limit is not required in MODE 4 or 5.

ACTIONS

A.1

With drywell average air temperature not within the limit of the LCO, drywell average air temperature must be restored within 8 hours. The Required Action is necessary to return operation to within the bounds of the primary 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

the
3

If the drywell average air temperature cannot be restored to within the limit 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 12 hours and to MODE 4 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.5.1

2
various

Verifying that the drywell average air temperature is within the LCO limit ensures that operation remains within the limits assumed for the primary containment analyses. Drywell air temperature is monitored in 4 quadrants and at various elevations (referenced to mean sea level). Due to the shape of the drywell, a volumetric average is used to determine an accurate representation of the actual average temperature.

2
selected to provide a representative sample of the overall drywell atmosphere

(continued)

BASES

SURVEILLANCE REQUIREMENT

SR 3.6.1.5.1 (continued)

The 24 hour Frequency of the SR was developed based on operating experience related to drywell average air temperature variations and temperature instrument drift during the applicable MODES and the low probability of a DBA occurring between surveillances. 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 drywell air temperature condition.

REFERENCES

2 U .1.3 1

1. FSAR, Section [6.2].
2. FSAR, ~~Section [6.2.1.4.1]~~. Table 6.2-1
3. ~~FSAR, Section [6.2.1.4.5].~~ 3

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.1.5 - DRYWELL AIR TEMPERATURE

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

3. Typographical/grammatical error corrected.

All changes are [1] unless otherwise indicated

Low Set Relief

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.6 Low-Low Set (LLS) Valves

Relief [2]

In addition, one relief valve is designed to open in the safety mode. (However, for the purposes of this LCO, only the low set relief mode of the relief valves is required.)

BASES

BACKGROUND

relief

low set relief

Insert BKGD-1

disc

The safety/relief valves (S/RVs) can actuate in either the safety mode, the Automatic Depressurization System mode, or the LLS mode. In the LLS mode (or power actuated mode of operation), a pneumatic diaphragm and stem assembly overcomes the spring force and opens the pilot valve. As in the safety mode, opening the pilot valve allows a differential pressure to develop across the main valve piston and opens the main valve. The main valve can stay open with valve inlet steam pressure as low as [50] psig. Below this pressure, steam pressure may not be sufficient to hold the main valve open against the spring force of the pilot valve. The pneumatic operator is arranged so that its malfunction will not prevent the valve disk from lifting if steam inlet pressure exceeds the safety mode pressure setpoints.

[3] partially will spring main However, with inlet steam pressure below 150psig fully

Insert BKGD-2

[2] relief valves low set relief setpoints low set relief [2]

Two [3] other

[Four] of the S/RVs are equipped to provide the LLS function. The LLS logic causes the LLS valves to be opened at a lower pressure than the relief or safety mode pressure setpoints and stay open longer, so that reopening more than one S/RV is prevented on subsequent actuations. Therefore, the LLS function prevents excessive short duration S/RV cycles with valve actuation at the relief setpoint.

valves two relief valves relief valve low set relief low set

Each S/RV discharges steam through a discharge line and quencher to a location near the bottom of the suppression pool, which causes a load on the suppression pool wall. Actuation at lower reactor pressure results in a lower load.

relief valve

APPLICABLE SAFETY ANALYSES

[2] low set no more than two relief valves

The LLS relief mode functions to ensure that the containment design basis of one S/RV operating on "subsequent actuations" is met. In other words, multiple simultaneous openings of S/RVs (following the initial opening), and the corresponding higher loads, are avoided. The safety analysis demonstrates that the LLS functions to avoid the induced thrust loads on the S/RV discharge line resulting from "subsequent actuations" of the S/RV during Design Basis Accidents (DBAs). Furthermore, the LLS function justifies the primary containment analysis assumption that

relief valves [2] low set relief relief valve

(continued)

A time delay in the low set relief valve logic permits actuation concurrent with an elevated water level in the discharge line.

Insert BKGD-1

For Unit 1, the low set relief valves are of the Electromatic type. The main valve is operated by a pilot valve assembly which is actuated by a solenoid. This solenoid can be automatically energized by an automatic depressurization logic signal or by pressure switches in the low set relief mode.

Insert BKGD-2

For Unit 2, the low set relief valves are of the Target Rock type. When the solenoid is energized, a magnetic force is developed which moves a plunger upward until it contacts the moveable core. This motion is transmitted through the pilot rod to fully open two pilot discs, allowing the control pressure above the main disc to vent through the second pilot seat to the downstream side of the valve. In addition, the motion of the pilot discs partially reduces the control pressure above the main disc. When the force of the control pressure acting on the top of the main disc falls below the force of the inlet pressure acting on the lower annular area, the main disc will move to the open position. In the open position, with the moveable core positioned close to the fixed core, the magnetic force is well in excess of the closing forces due to control pressure and return spring force. This ensures that the main disc will be held firmly in the open position. The main disc can be opened even with the valve inlet pressure equal to 0 psig.

All changes are [1] unless otherwise indicated

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~simultaneous S/RV openings occur only on the initial actuation for DBAs.~~ Even though ~~four~~ two [3] ~~LLS S/RVs~~ low set relief valves [2] are specified, ~~all~~ only ~~four~~ one [3] ~~LLS/S/RVs~~ ~~do not~~ low set relief valve is required to [2] operate in any DBA analysis.

~~LLS~~ The low set relief [2] valves satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii).

LCO

~~Four~~ Two [3] ~~LLS~~ low set relief valves are required to be OPERABLE to satisfy the assumptions of the safety analyses (Ref. 1). The requirements of this LCO are applicable to the mechanical and electrical ~~/pneumatic~~ capability of the ~~LLS~~ low set relief [2] valves to function for controlling the opening and closing of the ~~S/RVs~~ low set relief valves [2].

APPLICABILITY

In MODES 1, 2, and 3, an event could cause pressurization of the reactor and opening of ~~S/RVs~~ relief valves. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the ~~LLS~~ low set relief [2] valves OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

With one ~~LLS~~ low set relief valve inoperable, the remaining OPERABLE ~~LLS~~ low set relief valve~~s~~ ~~are~~ is adequate to perform the designed function. However, the overall reliability is reduced. The 14 day Completion Time takes into account the redundant capability afforded by the remaining ~~LLS~~ low set relief valve~~s~~ and the low probability of an event occurring during this period [4] in which the remaining ~~LLS~~ low set relief [2] valve capability would be ~~inadequate~~ required.

B.1 and B.2

If two ~~or more~~ ~~LLS~~ low set relief valves are inoperable or if the inoperable ~~LLS~~ low set relief valve 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 12 hours and to MODE 4 within 36 hours. The

(continued)

All changes are 1 unless otherwise indicated

BASES

ACTIONS

B.1 and B.2 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1

low set relief 2

A manual actuation of each LLS valve is performed to verify that the valve and solenoids are functioning properly and no blockage exists in the valve discharge line. This can be demonstrated by the response of the turbine control or bypass valve, by a change in the measured steam flow, or by any other method that is suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Adequate pressure at which this test is to be performed is ≥ [920] psig (the pressure recommended by the valve manufacturer). Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the LLS valves divert steam flow upon opening. Adequate steam flow is represented by [at least 1.25 turbine bypass valves open, or total steam flow ≥ 10^6 lb/hr]. The [18] month Frequency was based on the S/RV tests required by the ASME Boiler and Pressure Vessel Code, Section XI (Ref. 2). The Frequency of 18 months on a STAGGERED TEST BASIS ensures that each solenoid for each S/RV is alternately tested. 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.

300 3 2 low set relief 3 2 Insert SR-1 relief valve 1 4 3 24 2 valve is 2 low set relief 24 3

Since steam pressure is required to perform the Surveillance, however, and steam may not be available during a unit outage, the Surveillance may be performed during the startup following a unit outage. Unit startup is allowed prior to performing the test because valve OPERABILITY and the setpoints for overpressure protection are verified by Reference 2 prior to valve installation. After adequate reactor steam dome pressure and flow are reached, 12 hours is allowed to prepare for and perform the test.

is 4

This SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow is adequate to perform the test.

The 12 hours allowed for manual actuations after the required pressure and flow is reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR.

(continued)

1

Insert SR-1

Sufficient time is therefore allowed, after the required pressure and flow are achieved, to perform this test.

All changes are 1 unless otherwise indicated

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.6.2

The ~~LLS~~ low set relief 2 designated ~~S/RVs~~ relief valves are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to verify that the mechanical portions (i.e., solenoids) of the ~~LLS~~ low set relief 2 function operate as designed when initiated either by an actual or simulated automatic initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in ~~SR 3.3.6.3.7~~ LCO 3.3.6.3, "Low Set Relief Valve Instrumentation," 4 overlaps this SR to provide complete testing of the safety function.

The ~~18~~ 24 2 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the ~~18~~ 24 2 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes valve actuation. This prevents a reactor pressure vessel pressure blowdown.

REFERENCES

1. U 1 FSAR, Section ~~[5.5.17]~~ 6.2.1.3.4.2 3
2. ASME, Boiler and Pressure Vessel Code, Section XI.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.1.6 - LOW SET RELIEF VALVES

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. Changes have been made to reflect those changes made to the Specification.

3. The brackets have been removed and the proper plant specific value/nomenclature has been provided.

4. Changes have been made to be consistent with other places in the Bases.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.7 Reactor Building-to-Suppression Chamber Vacuum Breakers

BASES

BACKGROUND

1

The function of the reactor building-to-suppression chamber vacuum breakers is to relieve vacuum when primary containment depressurizes below reactor building pressure. If the drywell depressurizes below reactor building pressure, the negative differential pressure is mitigated by flow through the reactor building-to-suppression chamber vacuum breakers and through the suppression-chamber-to-drywell vacuum breakers. The design of the external (reactor building-to-suppression chamber) vacuum relief provisions consists of two vacuum breakers (a vacuum breaker and an air operated butterfly valve), located in series in each of two lines from the reactor building to the suppression chamber airspace. The butterfly valve is actuated by differential pressure. The vacuum breaker is self actuating and can be ~~remotely~~ operated for testing purposes. The two vacuum breakers in series must be closed to maintain a leak tight primary containment boundary.

mechanical

parallel 20 inch

Connected to a common 20 inch inlet line

a

switch

(similar to a check valve)

. The two parallel 20 inch vacuum breaker lines connect to a common 20 inch line, which, in turn, connects

locally

mechanical

A negative differential pressure across the drywell wall is caused by rapid depressurization of the drywell. Events that cause this rapid depressurization are cooling cycles, inadvertent primary containment spray actuation, and steam condensation in the event of a primary system rupture. Reactor building-to-suppression chamber vacuum breakers prevent an excessive negative differential pressure across the primary containment boundary. Cooling cycles result in minor pressure transients in the drywell, which occur slowly and are normally controlled by heating and ventilation equipment. Inadvertent spray actuation results in a more significant pressure transient and becomes important in sizing the external (reactor building-to-suppression chamber) vacuum breakers.

The external vacuum breakers are sized on the basis of the air flow from the secondary containment that is required to mitigate the depressurization transient and limit the maximum negative containment (drywell and suppression chamber) pressure to within design limits. The maximum depressurization rate is a function of the primary containment spray flow rate and temperature and the assumed initial conditions of the primary containment atmosphere.

(continued)

All changes are $\boxed{1}$ unless otherwise indicated

BASES

BACKGROUND (continued)

Low spray temperatures and atmospheric conditions that yield the minimum amount of contained noncondensible gases are assumed for conservatism.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving the reactor building-to-suppression chamber vacuum breakers are presented in Reference 1 as part of the accident response of the containment systems. Internal (suppression-chamber-to-drywell) and external (reactor building-to-suppression chamber) vacuum breakers are provided as part of the primary containment to limit the negative differential pressure across the drywell and suppression chamber walls, which form part of the primary containment boundary.

Insert 3.6.1.7 ASA

The safety analyses assume the external vacuum breakers to ~~be closed initially and to be fully open at [0.5] psid (Ref. 1). Additionally, of the two reactor building-to-suppression chamber vacuum breakers, one is assumed to fail in a closed position to satisfy the single active failure criterion.~~ Design Basis Accident (DBA) analyses ~~require~~ assume the vacuum breakers to be closed initially and ~~to~~ remain closed and leak tight with positive primary containment pressure.

that at least one vacuum breaker on each line

5

~~Five cases were considered in the safety analyses to determine the adequacy of the external vacuum breakers:~~

~~a. A small break loss of coolant accident followed by actuation of both primary containment spray loops;~~

~~b. Inadvertent actuation of one primary containment spray loop during normal operation;~~

~~c. Inadvertent actuation of both primary containment spray loops during normal operation;~~

~~d. A postulated DBA assuming Emergency Core Cooling Systems (ECCS) runout flow with a condensation effectiveness of 50%; and~~

~~e. A postulated DBA assuming ECCS runout flow with a condensation effectiveness of 100%.~~

~~The results of these five cases show that the external vacuum breakers, with an opening setpoint of [0.5] psid, are~~

$\boxed{3}$

(continued)

1

<u>Insert 3.6.1.7 ASA</u>

, with the mechanical vacuum breakers counter balanced to open at 0.5 psid and to be fully open in one second. The air operated butterfly valve vacuum breakers are assumed to open concurrent with the mechanical vacuum breakers and be full open in one second (Ref. 1). Since only one of the two parallel 20 inch vacuum breaker lines is required to protect the suppression chamber from excessive negative differential pressure, the single active failure criterion is satisfied.

BASES

APPLICABLE SAFETY ANALYSES (continued)

capable of maintaining the differential pressure within design limits. 3

The reactor building-to-suppression chamber vacuum breakers satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii) 1

LCO

All reactor building-to-suppression chamber vacuum breakers are required to be OPERABLE to satisfy the assumptions used in the safety analyses. The requirement ensures that the two mechanical vacuum breakers (vacuum breaker and air operated butterfly valve) in each of the two lines from the reactor building to the suppression chamber airspace are closed (except during testing or when performing their intended function). Also, the requirement ensures both vacuum breakers in each line will open to relieve a negative pressure in the suppression chamber. 1

APPLICABILITY 4

~~In MODES 1, 2, and 3, a DBA could cause pressurization of primary containment. In MODES 1, 2, and 3, the Suppression Pool Spray System is required to be OPERABLE to mitigate the effects of a DBA.~~ Excessive negative pressure inside primary containment could occur due to inadvertent initiation of ~~this system. Therefore, the vacuum breakers are required to be OPERABLE in MODES 1, 2, and 3, when the Suppression Pool Spray System is required to be OPERABLE, to mitigate the effects of inadvertent actuation of the Suppression Pool Spray System.~~ drywell sprays. 1

4 ~~Also,~~ In MODES 1, 2, and 3, a DBA could result in excessive negative differential pressure across the drywell wall caused by the rapid depressurization of the drywell. The event that results in the limiting rapid depressurization of the drywell is the primary system rupture, which purges the drywell of air and fills the drywell free airspace with steam. Subsequent condensation of the steam would result in depressurization of the drywell, 1 which, after the suppression chamber-to-drywell vacuum breakers open (due to differential pressure between the suppression chamber and drywell), would result in depressurization of the suppression chamber. The limiting pressure and temperature of the primary system prior to a DBA occur in MODES 1, 2, and 3.

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining reactor

(continued)

BASES

APPLICABILITY (continued)

building-to-suppression chamber vacuum breakers OPERABLE is not required in MODE 4 or 5.

ACTIONS

A Note has been added to provide clarification that, for the purpose of this LCO, separate Condition entry is allowed for each penetration flow path.

reactor building-to-suppression chamber vacuum breaker line 5

A.1

lines with one

With one or more vacuum breakers not closed, the leak tight primary containment boundary may be threatened. Therefore, the inoperable vacuum breakers must be restored to OPERABLE status or the open vacuum breaker closed within 72 hours. 8

7 days 5

5 ~~The 72 hour Completion Time is consistent with requirements for inoperable suppression-chamber-to-drywell vacuum breakers in LCO 3.6.1.8, "Suppression-Chamber-to-Drywell Vacuum Breakers."~~ The 72 hour Completion Time takes into account the redundancy capability afforded by the remaining breakers, the fact that the OPERABLE breaker in each of the lines is closed, and the low probability of an event occurring that would require the vacuum breakers to be OPERABLE during this period. 7

B.1

With one or more lines with two vacuum breakers not closed, primary containment integrity is not maintained. Therefore, one open vacuum breaker must be closed within 1 hour. This Completion Time is consistent with the ACTIONS of LCO 3.6.1.1, "Primary Containment," which requires that primary containment be restored to OPERABLE status within 1 hour.

C.1

With one line with one or more vacuum breakers inoperable for opening, the leak tight primary containment boundary is intact. The ability to mitigate an event that causes a containment depressurization is threatened, however, if both vacuum breakers in at least one vacuum breaker penetration are not OPERABLE. Therefore, the inoperable vacuum breaker

(continued)

BASES

ACTIONS

C.1 (continued)

7 days

5

must be restored to OPERABLE status within ~~72 hours~~. This is consistent with the Completion Time for Condition A and the fact that the leak tight primary containment boundary is being maintained.

D.1

5

With two ~~[or more]~~ lines with one or more vacuum breakers inoperable for opening, the primary containment boundary is intact. However, in the event of a containment depressurization, the function of the vacuum breakers is lost. Therefore, all vacuum breakers in ~~[one]~~ line must be restored to OPERABLE status within 1 hour. This Completion Time is consistent with the ACTIONS of LCO 3.6.1.1, which requires that primary containment be restored to OPERABLE status within 1 hour.

2

E.1 and E.2

8

If any Required Action Action and associated Completion Time cannot be met

~~If all the vacuum breakers in [one] line cannot be closed or 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 12 hours and to MODE 4 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.7.1

Each vacuum breaker is verified to be closed to ensure that a potential breach in the primary containment boundary is not present. This Surveillance is performed by observing local or control room indications of vacuum breaker position ~~or by verifying a differential pressure of [0.5] psid is maintained between the reactor building and suppression chamber~~. The 14 day Frequency is based on engineering

6

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.7.1 (continued)

judgment, is considered adequate in view of other indications of vacuum breaker status available to operations personnel, and has been shown to be acceptable through operating experience.

Two Notes are added to this SR. The first Note allows reactor-to-suppression chamber vacuum breakers opened in conjunction with the performance of a Surveillance to not be considered as failing this SR. These periods of opening vacuum breakers are controlled by plant procedures and do not represent inoperable vacuum breakers. The second Note is included to clarify that vacuum breakers open due to an actual differential pressure are not considered as failing this SR.

SR 3.6.1.7.2

Each vacuum breaker must be cycled to ensure that it opens properly to perform its design function and returns to its fully closed position. This ensures that the safety analysis assumptions are valid. The [92] day Frequency of this SR was developed based upon Inservice Testing Program requirements to perform valve testing at least once every [92] days. 2

SR 3.6.1.7.3

Demonstration of vacuum breaker opening setpoint is necessary to ensure that the safety analysis assumption regarding vacuum breaker full open differential pressure of ≤ [0.5] psid is valid. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. For this ~~unit~~, the [18] month Frequency has been shown to be acceptable, based on operating experience, and is further justified because of other surveillances performed at shorter Frequencies that convey the proper functioning status of each vacuum breaker. 2 24 2 24 plant 7 1

REFERENCES

1. a [FSAR], Section ~~[6.2]~~. 1.2.4 2

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.1.7 - REACTOR BUILDING-TO-SUPPRESSION CHAMBER VACUUM BREAKERS

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. These details concerning the five cases which are considered in the safety analyses with respect to reactor building-to-suppression chamber vacuum breakers have been deleted. This level of detail is not necessary to be included in the Bases for understanding of the LCO requirements.

4. Inadvertent actuation of the suppression pool spray system is not the main concern for depressurizing the drywell, a LOCA inside the drywell is the main concern. Therefore, this section has been reworded to place proper emphasis on the proper reason.

5. Changes have been made to reflect those changes made to the Specification.

6. The alternate method has been deleted since it is not valid for Quad Cities 1 and 2.

7. Editorial change made for enhanced clarity.

8. These changes have been made for consistency with similar phrases in other parts of the Bases and/or to be consistent with the Specification.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.8 Suppression Chamber-to-Drywell Vacuum Breakers

BASES

1

BACKGROUND

The function of the suppression chamber-to-drywell vacuum breakers is to relieve vacuum in the drywell. There are 2 [12] internal vacuum breakers located on the vent header of the vent system between the drywell and the suppression chamber, which allow air and steam flow from the suppression chamber to the drywell when the drywell is at a negative pressure with respect to the suppression chamber. Therefore, suppression chamber-to-drywell vacuum breakers prevent an excessive negative differential pressure across the suppression chamber- 3 wetwell drywell boundary. Each vacuum breaker is a self actuating valve, similar to a check valve, which can be remotely operated for testing purposes.

A negative differential pressure across the drywell wall is caused by rapid depressurization of the drywell. Events that cause this rapid depressurization are cooling cycles, inadvertent drywell spray actuation, and steam condensation from sprays or subcooled water reflood of a break in the event of a primary system rupture. Cooling cycles result in minor pressure transients in the drywell that occur slowly and are normally controlled by heating and ventilation equipment. Spray actuation or spill of subcooled water out of a break results in more significant pressure transients and becomes important in sizing the internal vacuum breakers.

In the event of a primary system rupture, steam condensation within the drywell results in the most severe pressure transient. Following a primary system rupture, air in the drywell is purged into the suppression chamber free airspace, leaving the drywell full of steam. Subsequent condensation of the steam can be caused in two possible ways, namely, Emergency Core Cooling Systems flow from a recirculation line break, or drywell spray actuation following a loss of coolant accident (LOCA). These two cases determine the maximum depressurization rate of the drywell.

In addition, the waterleg in the Mark I Vent System downcomer is controlled by the drywell-to-suppression chamber differential pressure. If the drywell pressure is

(continued)

BASES

BACKGROUND (continued)

less than the suppression chamber pressure, there will be an increase in the vent waterleg. This will result in an increase in the water clearing inertia in the event of a postulated LOCA, resulting in an increase in the peak drywell pressure. This in turn will result in an increase in the pool swell dynamic loads. The internal vacuum breakers limit the height of the waterleg in the vent system during normal operation.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving the suppression chamber-to-drywell vacuum breakers are presented in Reference 1 as part of the accident response of the primary containment systems. Internal (suppression chamber-to-drywell) and external (reactor building-to-suppression chamber) vacuum breakers are provided as part of the primary containment to limit the negative differential pressure across the drywell and suppression chamber walls that form part of the primary containment boundary.

2 2 3 5 3 the OPERABILITY of

The safety analyses assume that the internal vacuum breakers are closed initially and are fully open at a differential pressure of [0.5] psid (Ref. 2). Additionally, 3 of the 12 internal vacuum breakers are assumed to fail in a closed position (Ref. 1). The results of the analyses show that the design pressure is not exceeded even under the worst case accident scenario. The vacuum breaker opening differential pressure setpoint and ~~the requirement that~~ [9] of [12] vacuum breakers ~~be OPERABLE~~ are a result of the requirement placed on the vacuum breakers to limit the vent system waterleg height. ~~The total cross sectional area of the main vent system between the drywell and suppression chamber needed to fulfill this requirement has been established as a minimum of [51.5] times the total break area (Ref. 1). In turn, the vacuum relief capacity between the drywell and suppression chamber should be [1/16] of the total main vent cross sectional area,~~ with the valves set to operate at [0.5] psid differential pressure. Design Basis Accident (DBA) analyses ~~require~~ the vacuum breakers to be closed initially and to remain closed and leak tight ~~with~~ the suppression ~~pool~~ at a positive pressure relative to the drywell.

However this requirement is conservative.

2 3 7 7 2 4

Insert ASA-1

until

Chamber

assume

is

(continued)

3 Insert ASA-1

The vacuum breakers are sized on the basis of the Bodega pressure suppression system tests. These tests were conducted by simulating a small break LOCA, which tend to cause vent system waterleg height variations. The vacuum breaker capacity selected is more than adequate to limit the pressure differential between the suppression chamber and drywell post LOCA

BASES

APPLICABLE SAFETY ANALYSES (continued)

The suppression chamber-to-drywell vacuum breakers satisfy Criterion 3 of ~~the NRC Policy Statement~~. 10 CFR 50.36(c)(2)(ii) 3

LCO

Only [9] of the [12] vacuum breakers must be OPERABLE for opening. All suppression chamber-to-drywell vacuum breakers, ~~however,~~ are ~~required to be~~ closed (except during testing or when the vacuum breakers are performing their intended design function). ~~The vacuum breaker OPERABILITY requirement~~ This LCO also ensures that provides assurance that the drywell-to-suppression chamber negative differential pressure remains below the design value. The requirement that the vacuum breakers be closed ensures that there is no excessive bypass leakage should a LOCA occur. vacuum breakers will open so that to 2 5

APPLICABILITY

~~In MODES 1, 2, and 3, the Suppression Pool Spray System is required to be OPERABLE to mitigate the effects of a DBA. Excessive negative pressure inside the drywell could occur due to inadvertent actuation of this system. The vacuum breakers, therefore, are required to be OPERABLE in MODES 1, 2, and 3, when the Suppression Pool Spray System is required to be OPERABLE, to mitigate the effects of inadvertent actuation of the Suppression Pool Spray System.~~ 6

~~Also, i~~In MODES 1, 2, and 3, a DBA could result in excessive negative differential pressure across the drywell wall, caused by the rapid depressurization of the drywell. The event that results in the limiting rapid depressurization of the drywell is the primary system rupture that purges the drywell of air and fills the drywell free airspace with steam. Subsequent condensation of the steam would result in depressurization of the drywell, which, after the suppression chamber-to-drywell vacuum breakers open (due to differential pressure between the suppression chamber and drywell), would result in depressurization of the suppression chamber. The limiting pressure and temperature of the primary system prior to a DBA occur in MODES 1, 2, and 3. drywell sprays 3 6

In MODES 4 and 5, the probability and consequences of these events are reduced by the pressure and temperature limitations in these MODES; therefore, maintaining suppression chamber-to-drywell vacuum breakers OPERABLE is not required in MODE 4 or 5.

(continued)

BASES (continued)

ACTIONS

A.1

(a) 5

With one of the required vacuum breakers inoperable for opening (e.g., the vacuum breaker is not open and may be stuck closed or not within its opening setpoint limit, so that it would not function as designed during an event that depressurized the drywell), the remaining [eight] OPERABLE vacuum breakers are capable of providing the vacuum relief function. However, overall system reliability is reduced because ~~a single~~ failure in ~~one of~~ the remaining vacuum breakers could result in an excessive suppression chamber-to-drywell differential pressure during a DBA. Therefore, with one of the [nine] required vacuum breakers inoperable, 72 hours is allowed to restore at least one of the inoperable vacuum breakers to OPERABLE status so that plant conditions are consistent with ~~those assumed for the design basis analysis~~. The 72 hour Completion Time is considered acceptable due to the low probability of an event in which the remaining vacuum breaker capability would not be adequate.

additional

3

the LCO requirements

2

5

With one vacuum breaker not closed,

exists

B.1

3

primary Containment

~~An open vacuum breaker allows~~ communication between the drywell and suppression chamber airspace, and, as a result, there is the potential for ~~suppression chamber~~ overpressurization due to this bypass leakage if a LOCA were to occur. Therefore, the open vacuum breaker must be closed. A short time is allowed to close the vacuum breaker due to the low probability of an event that would pressurize primary containment. If vacuum breaker position indication is not reliable, an alternate method of verifying that the vacuum breakers are closed is to verify that a differential pressure of [0.5] psid between the suppression chamber and drywell is maintained for 1 hour without makeup. The required 2 hour Completion Time is considered adequate to perform this test. (4) 7

2

C.1 and C.2

If any Required Action and associated Completion Time cannot be met,

8

~~If the inoperable suppression chamber-to-drywell vacuum breaker cannot be closed or 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

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.8.1

Each vacuum breaker is verified closed to ensure that this potential large bypass leakage path is not present. This Surveillance is performed by observing the vacuum breaker position indication or by verifying that a differential pressure of [0.5] psid between the suppression chamber and drywell is maintained for 1 hour without makeup. The 14 day Frequency is based on engineering judgment, is considered adequate in view of other indications of vacuum breaker status available to operations personnel, and has been shown to be acceptable through operating experience. ~~This verification is also required within 2 hours after any discharge of steam to the suppression chamber from the safety/relief valves or any operation that causes the drywell-to-suppression chamber differential pressure to be reduced by ≥ [0.5] psid.~~ 2 7

Two ~~A~~ Note(s) ~~is~~ are added to this SR. The first Note ~~which~~ allows suppression chamber-to-drywell vacuum breakers opened in conjunction with the performance of a Surveillance to not be considered as failing this SR. These periods of opening vacuum breakers are controlled by plant procedures and do not represent inoperable vacuum breakers. The second Note is included to clarify that vacuum breakers open due to an actual differential pressure are not considered as failing this SR. 7 7

SR 3.6.1.8.2

Each required vacuum breaker must be cycled to ensure that it opens adequately to perform its design function and returns to the fully closed position. This ensures that the safety analysis assumptions are valid. The 31 day Frequency of this SR was developed, based on Inservice Testing Program requirements to perform valve testing at least once every 92 days. A 31 day Frequency was chosen to provide

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.8.2 (continued)

7

additional assurance that the vacuum breakers are OPERABLE, since they are located in a harsh environment (the suppression chamber airspace). In addition, this functional test is required within 12 hours after either a discharge of steam to the suppression chamber from the safety/relief valves or after an operation that causes any of the vacuum breakers to open.

3 7

from the closed position 5

SR 3.6.1.8.3

Verification of the vacuum breaker opening setpoint is necessary to ensure that the safety analysis assumption regarding vacuum breaker full open differential pressure of [0.5] psid is valid. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. For this facility, the [18] month Frequency has been shown to be acceptable, based on operating experience, and is further justified because of other surveillances performed at shorter Frequencies that convey the proper functioning status of each vacuum breaker.

≤ 2 24 2 24 8

REFERENCES

3 U

1. FSAR, Section [6.2]. 2 .1.2.4.1

2. UFSAR, Table 6.2-1 3

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.1.8 - SUPPRESSION CHAMBER-TO-DRYWELL VACUUM BREAKERS

1. Typographical error corrected for accuracy.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

4. The statement has been modified since it is incorrect; the pressure could be positive or negative depending upon the situation. Also, the design basis only assumes the pressure is within the limits, not positive. Therefore, the vacuum breakers are required to remain closed only "until" the suppression pool is at a positive pressure relative to the drywell. At this time, they may be open to perform their design function (i.e., relieve pressure).

5. Editorial change made for enhanced clarity.

6. Inadvertent actuation of a spray system is not the main concern for depressurizing the drywell, a LOCA inside the drywell is the main concern. Therefore, this section has been reworded to place proper emphasis on the proper reason. In addition, inadvertent actuation of suppression pool spray is not a concern at all relative to causing an excessive negative pressure event; drywell spray is the system that can cause this event. Therefore, the Bases have been changed from suppression pool spray to drywell spray when discussing this event.

7. Changes have been made to reflect those changes made to the Specification.

8. These changes have been made for consistency with similar phrases in other parts of the Bases and/or to be consistent with the Specification.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.9 Main Steam Isolation Valve (MSIV) Leakage Control System (LCS)

1

BASES

BACKGROUND

The MSIV LCS supplements the isolation function of the MSIVs by processing the fission products that could leak through the closed MSIVs after a Design Basis Accident (DBA) loss of coolant accident (LOCA).

The MSIV LCS consists of two independent subsystems: an inboard subsystem, connected between the inboard and outboard MSIVs, and an outboard subsystem, connected immediately downstream of the outboard MSIVs. Each subsystem is capable of processing leakage from MSIVs following a DBA LOCA. Each subsystem consists of blowers (one blower for the inboard subsystem and two blowers for the outboard subsystem), valves, piping, and heaters (for the inboard subsystem only). Four electric heaters in the inboard subsystem are provided to boil off any condensate prior to the gas mixture passing through the flow limiter.

Each subsystem operates in two process modes: depressurization and bleedoff. The depressurization process reduces the steam line pressure to within the operating capability of equipment used for the bleedoff mode. During bleedoff (long term leakage control), the blowers maintain a negative pressure in the main steam lines (Ref. 1). This ensures the leakage through the closed MSIVs is collected and processed by the MSIV LCS. In both process modes, the effluent is discharged to the secondary containment and ultimately filtered by the Standby Gas Treatment (SGT) System.

The MSIV LCS is manually initiated approximately 20 minutes following a DBA LOCA (Ref. 2).

APPLICABLE SAFETY ANALYSES

The MSIV LCS mitigates the consequences of a DBA LOCA by ensuring that fission products that may leak from the closed MSIVs are diverted to the secondary containment and ultimately filtered by the SGT System. The operation of the MSIV LCS prevents a release of untreated leakage for this type of event.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The MSIV LCS satisfies Criterion 3 of the NRC Policy Statement.
LCO	One MSIV LCS subsystem can provide the required processing of the MSIV leakage. To ensure that this capability is available, assuming worst case single failure, two MSIV LCS subsystems must be OPERABLE.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment. Therefore, MSIV LCS OPERABILITY is required during these MODES. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the MSIV LCS OPERABLE is not required in MODE 4 or 5 to ensure MSIV leakage is processed.

ACTIONS

A.1

With one MSIV LCS subsystem inoperable, the inoperable MSIV LCS subsystem must be restored to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE MSIV LCS subsystem is adequate to perform the required leakage control function. However, the overall reliability is reduced because a single failure in the remaining subsystem could result in a total loss of MSIV leakage control function. The 30 day Completion Time is based on the redundant capability afforded by the remaining OPERABLE MSIV LCS subsystem and the low probability of a DBA LOCA occurring during this period.

B.1

With two MSIV LCS subsystems inoperable, at least one subsystem must be restored to OPERABLE status within 7 days. The 7 day Completion Time is based on the low probability of the occurrence of a DBA LOCA.

(continued)

1

BASES

ACTIONS (continued)

C.1 and C.2

If the MSIV LCS subsystem 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 12 hours and to MODE 4 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.9.1

Each MSIV LCS blower is operated for ≥ [15] minutes to verify OPERABILITY. The 31 day Frequency was developed considering the known reliability of the LCS blower and controls, the two subsystem redundancy, and the low probability of a significant degradation of the MSIV LCS subsystems occurring between surveillances and has been shown to be acceptable through operating experience.

1

SR 3.6.1.9.2

The electrical continuity of each inboard MSIV LCS subsystem heater is verified by a resistance check, by verifying that the rate of temperature increase meets specifications, or by verifying that the current or wattage draw meets specifications. The 31 day Frequency is based on operating experience that has shown that these components usually pass this Surveillance when performed at this Frequency.

SR 3.6.1.9.3

A system functional test is performed to ensure that the MSIV LCS will operate through its operating sequence. This includes verifying that the automatic positioning of the valves and the operation of each interlock and timer are correct, that the blowers start and develop the required flow rate and the necessary vacuum, and that the upstream heaters meet current or wattage draw requirements (if not used to verify electrical continuity in SR 3.6.1.9.2). The

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.9.3 (continued)

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

1

REFERENCES

1. FSAR, Section [6.5].
2. Regulatory Guide 1.96, Revision [1].

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ISTS BASES: 3.6.1.9 - MAIN STEAM ISOLATION VALVE (MSIV)
LEAKAGE CONTROL SYSTEM (LCS)

1. This Bases has been deleted since the associated Specification has been deleted.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.1 Suppression Pool Average Temperature

BASES

BACKGROUND

The suppression chamber is a toroidal shaped, steel pressure vessel containing a volume of water called the suppression pool. The suppression pool is designed to absorb the decay heat and sensible energy released during a reactor blowdown from ~~safety~~/relief valve discharges or from Design Basis Accidents (DBAs). The suppression pool must quench all the steam released through the downcomer lines during a loss of coolant accident (LOCA). This is the essential mitigative feature of a pressure suppression containment that ensures that the peak containment pressure is maintained below the maximum allowable pressure for DBAs (~~[62]~~ psig). The suppression pool must also condense steam from steam exhaust lines in the turbine driven systems (i.e., the High Pressure Coolant Injection System and Reactor Core Isolation Cooling System). Suppression pool average temperature (along with LCO 3.6.2.2, "Suppression Pool Water Level") is a key indication of the capacity of the suppression pool to fulfill these requirements. 3 1

The technical concerns that lead to the development of suppression pool average temperature limits are as follows:

a. Complete steam condensation ~~[—the original limit for the end of a LOCA blowdown was 170°F, based on the Bodega Bay and Humboldt Bay Tests]~~;

b. Primary containment peak pressure and temperature ~~[—design pressure is [62] psig and design temperature is [340]°F (Ref. 1)]~~;

c. Condensation oscillation loads ~~[—maximum allowable initial temperature is [110]°F]~~; and

d. Chugging loads ~~[—these only occur at < [135]°F; therefore, there is no initial temperature limit because of chugging]~~. 2

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The postulated DBA against which the primary containment performance is evaluated is the entire spectrum of postulated pipe breaks within the primary containment. Inputs to the safety analyses include initial suppression pool water volume and suppression pool temperature (Reference 1 for LOCAs and Reference 2 for the pool temperature analyses required by Reference 3). An initial pool temperature of ~~[~~95~~]~~°F is assumed for the Reference 1 ~~and Reference~~ 2 analyses. Reactor shutdown at a pool temperature of ~~[~~110~~]~~°F and vessel depressurization at a pool temperature of ~~[~~120~~]~~°F are assumed for the Reference 2 analyses. The limit of ~~[~~105~~]~~°F, at which testing is terminated, is not used in the safety analyses because DBAs are assumed to not initiate during unit testing.

3 , and 4 3 1

Suppression pool average temperature satisfies Criteria 2 and 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii) 3

LCO

A limitation on the suppression pool average temperature is required to provide assurance that the containment conditions assumed for the safety analyses are met. This limitation subsequently ensures that peak primary containment pressures and temperatures do not exceed maximum allowable values during a postulated DBA or any transient resulting in heatup of the suppression pool. The LCO requirements are:

a. Average temperature ≤ ~~[~~95~~]~~°F ~~[when any OPERABLE intermediate range monitor (IRM) channel is > [25/40] divisions of full scale on Range 7]~~ [with THERMAL POWER > 1% ~~RATED THERMAL POWER~~ ~~[~~RTP~~]~~] and no testing that adds heat to the suppression pool ~~is~~ being performed. This requirement ensures that licensing bases initial conditions are met.

TSTF-206 1 1

b. Average temperature ≤ ~~[~~105~~]~~°F ~~[when any OPERABLE IRM channel is > [25/40] divisions of full scale on Range 7]~~ [with THERMAL POWER > 1% RTP] and testing that adds heat to the suppression pool ~~is~~ being performed. This required value ensures that the unit has testing flexibility, and was selected to provide margin below the ~~[~~110~~]~~°F limit at which reactor shutdown is required. When testing ends, temperature must be restored to ≤ ~~[~~95~~]~~°F within 24 hours according to Required Action A.2. Therefore, the time period that the temperature is > ~~[~~95~~]~~°F is

1 TSTF-206 1

(continued)

BASES

LCO (continued)

short enough not to cause a significant increase in unit risk.

c. Average temperature ≤ ~~[~~110~~]~~°F ~~when all OPERABLE IRM channels are ≤ [25/40] divisions of full scale on Range 7~~. This requirement ensures that the unit will be shut down at > ~~[~~110~~]~~°F. The pool is designed to absorb decay heat and sensible heat but could be heated beyond design limits by the steam generated if the reactor is not shut down. 1

with THERMAL POWER ≤ 1% RTP

TSTF-206

4

~~Note that [25/40] divisions of full scale on IRM Range 7 is a convenient measure of when the reactor is producing power essentially equivalent to 1% RTP.~~ At ~~this power level~~ 1% RTP*, heat input is approximately equal to normal system heat losses. 1

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause significant heatup of the suppression pool. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining suppression pool average temperature within limits is not required in MODE 4 or 5.

ACTIONS

A.1 and A.2

With the suppression pool average temperature above the specified limit when not performing testing that adds heat to the suppression pool and when above the specified power ~~indication~~ limit 4, the initial conditions exceed the conditions assumed for the Reference 1, ~~3,~~ and 4 analyses. However, primary containment cooling capability still exists, and the primary containment pressure suppression function will occur at temperatures well above those assumed for safety analyses. Therefore, continued operation is allowed for a limited time. The 24 hour Completion Time is adequate to allow the suppression pool average temperature to be restored below the limit. Additionally, when suppression pool temperature is > ~~[~~95~~]~~°F, increased monitoring of the suppression pool temperature is required to ensure that it remains ≤ ~~[~~110~~]~~°F. The once per hour Completion Time is adequate based on past experience, which has shown that pool temperature increases relatively slowly except when testing 2 3 1

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

that adds heat to the suppression pool is being performed. Furthermore, the once per hour Completion Time is considered adequate in view of other indications in the control room, ~~including alarms~~, to alert the operator to an abnormal suppression pool average temperature condition.

3

B.1

If the suppression pool average temperature cannot be restored to within limits 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 power must be reduced to ~~≤ [25/40] divisions of full scale on Range 7 for all OPERABLE IRMs~~ within 12 hours. The 12 hour Completion Time is reasonable, based on operating experience, to reduce power from full power conditions in an orderly manner and without challenging plant systems.

TSTF-206

1 [≤ 1% RTP]

C.1

Suppression pool average temperature is allowed to be > ~~[95]~~ °F ~~when any OPERABLE IRM channel is > [25/40] divisions of full scale on Range 7~~, and when testing that adds heat to the suppression pool is being performed. However, if temperature is > ~~[105]~~ °F, all testing must be immediately suspended to preserve the heat absorption capability of the suppression pool. With the testing suspended, Condition A is entered and the Required Actions and associated Completion Times are applicable.

TSTF-206

[with THERMAL POWER > 1% RTP]

1

D.1 and D.2

6

Suppression pool average temperature > ~~[110]~~ °F requires that the reactor be shut down immediately. This is accomplished by placing the reactor mode switch in the shutdown position. Further cooldown to Mode 4 is required at normal cooldown rates (provided pool temperature remains ≤ ~~[120]~~ °F). Additionally, when suppression pool temperature is > ~~[110]~~ °F, increased monitoring of pool temperature is required to ensure that it remains ≤ ~~[120]~~ °F. The once per 30 minute Completion Time is adequate, based on operating

1

5

within 36 hours

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

experience. Given the high suppression pool average temperature in this condition, the monitoring Frequency is increased to twice that of Condition A. Furthermore, the 30 minute Completion Time is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition. 6 3

E.1 and E.2

If suppression pool average temperature cannot be maintained at ≤ [120]°F, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the reactor pressure must be reduced to < [200] psig within 12 hours, and the plant must be brought to at least MODE 4 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. 1 150

Continued addition of heat to the suppression pool with suppression pool temperature > [120]°F could result in exceeding the design basis maximum allowable values for primary containment temperature or pressure. Furthermore, if a blowdown were to occur when the temperature was > [120]°F, the maximum allowable bulk and local temperatures could be exceeded very quickly. 1

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1.1

The suppression pool average temperature is regularly monitored to ensure that the required limits are satisfied. The average temperature is determined by taking an arithmetic average of OPERABLE suppression pool water temperature channels. The 24 hour Frequency has been shown, based on operating experience, to be acceptable. When heat is being added to the suppression pool by testing, however, it is necessary to monitor suppression pool temperature more frequently. The 5 minute Frequency during testing is justified by the rates at which tests will heat up the suppression pool, has been shown to be acceptable based on

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1.1 (continued)

operating experience, and provides assurance that allowable pool temperatures are not exceeded. The Frequencies are further justified in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool average temperature condition.

REFERENCES

3

1. UFSAR, Section ~~[6.2]~~.
2. UFSAR, Section ~~[15.1]~~. 6.2.1.3.4.5 1
3. NUREG-0783.
4. ~~[Mark I Containment Program.]~~

1 Quad Cities Nuclear Power Station Units 1 and 2, Mark I Plant Unique Analysis Report, COM-02-039-1, May 1983.

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. The discussions of the four different concerns that lead to the development of the suppression pool average temperature limits have been deleted. The appropriate analysis is described in the UFSAR (References 1 and 2) and discussion in the Bases is not needed for understanding this Specification.

3. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

4. Changes have been made to reflect those changes made to the Specification.

5. Editorial change made for enhanced clarity.

6. Typographical error corrected.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.2 Suppression Pool Water Level

BASES

BACKGROUND

The suppression chamber is a toroidal shaped, steel pressure vessel containing a volume of water called the suppression pool. The suppression pool is designed to absorb the energy associated with decay heat and sensible heat released during a reactor blowdown from ~~safety/~~relief valve ~~(S/RV)~~ discharges or from a Design Basis Accident (DBA). The suppression pool must quench all the steam released through the downcomer lines during a loss of coolant accident (LOCA). This is the essential mitigative feature of a pressure suppression containment, which ensures that the peak containment pressure is maintained below the maximum allowable pressure for DBAs (~~[62]~~ psig). The suppression pool must also condense steam from the steam exhaust lines in the turbine driven systems (i.e., High Pressure Coolant Injection (HPCI) System and Reactor Core Isolation Cooling (RCIC) System) and provides the main emergency water supply source for the reactor vessel. The suppression pool volume ranges between approximately 111,500 ~~[87,300]~~ ft^3 at the low water level limit of 14 ft 1 inch ~~[12 ft 2 inches]~~ and approximately 115,000 ~~[90,550]~~ ft^3 at the high water level limit of 14 ft 5 inches ~~[12 ft 6 inches]~~.

2 1 1

If the suppression pool water level is too low, an insufficient amount of water would be available to adequately condense the steam from the relief valve ~~S/RV~~ quenchers, downcomer lines ~~main vents~~, or HPCI and RCIC turbine exhaust lines. Low suppression pool water level could also result in an inadequate emergency makeup water source to the Emergency Core Cooling System. The lower volume would also absorb less steam energy before heating up excessively. Therefore, a minimum suppression pool water level is specified.

2 3 3

If the suppression pool water level is too high, it could result in excessive clearing loads from ~~S/RV~~ discharges and excessive pool swell loads during a DBA LOCA. Therefore, a maximum pool water level is specified. This LCO specifies an acceptable range to prevent the suppression pool water level from being either too high or too low.

(continued)

BASES (continued)

relief valve 2

APPLICABLE SAFETY ANALYSES

Initial suppression pool water level affects suppression pool temperature response calculations, calculated drywell pressure during vent clearing for a DBA, calculated pool swell loads for a DBA LOCA, and calculated loads due to S/RV discharges. Suppression pool water level must be maintained within the limits specified so that the safety analysis of Reference 1 remains valid. 2

Suppression pool water level satisfies Criteria 2 and 3 of the NRC Policy Statement.

10 CFR 50.36 (c)(2)(ii) 2

LCO 1

1 14 ft 5 inches

A limit that suppression pool water level be ≥ [12 ft 2 inches] and ≤ [12 ft 6 inches] is required to ensure that the primary containment conditions assumed for the safety analyses are met. Either the high or low water level limits were used in the safety analyses, depending upon which is more conservative for a particular calculation.

14 ft 1 inch

above the bottom of the suppression chamber

APPLICABILITY

In MODES 1, 2, and 3, a DBA would cause significant loads on the primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. The requirements for maintaining suppression pool water level within limits in MODE 4 or 5 is addressed in LCO 3.5.2, "ECCS-Shutdown."

ACTIONS

A.1

2 relief valve

2 the downcomers

With suppression pool water level outside the limits, the conditions assumed for the safety analyses are not met. If water level is below the minimum level, the pressure suppression function still exists as long as main vents are covered, HPCI and RCIC turbine exhausts are covered, and S/RV quenchers are covered. If suppression pool water level is above the maximum level, protection against overpressurization still exists due to the margin in the peak containment pressure analysis and the capability of the Drywell Spray System. Therefore, continued operation for a

2 Containment

(continued)

BASES

ACTIONS

A.1 (continued)

limited time is allowed. The 2 hour Completion Time is sufficient to restore suppression pool water level to within limits. Also, it takes into account the low probability of an event impacting the suppression pool water level occurring during this interval.

B.1 and B.2

If suppression pool water level cannot be restored to within limits 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 12 hours and to MODE 4 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.2.1

Verification of the suppression pool water level is to ensure that the required limits are satisfied. ~~The 24 hour Frequency of this SR was developed considering operating experience related to trending variations in suppression pool water level and water level instrument drift during the applicable MODES and to assessing the proximity to the specified LCO level limits.~~ 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 suppression pool water level condition.

2

The 24 hour Frequency has been shown to be acceptable based on operating experience.

REFERENCES

2

1. UFSAR, Section ~~X~~6.2~~X~~. 1

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.2.2 - SUPPRESSION POOL WATER LEVEL

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

3. Changes have been made to be consistent with other places in the Bases.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

BASES

BACKGROUND

Following a Design Basis Accident (DBA), the RHR Suppression Pool Cooling System removes heat from the suppression pool. The suppression pool is designed to absorb the sudden input of heat from the primary system. In the long term, the pool continues to absorb residual heat generated by fuel in the reactor core. Some means must be provided to remove heat from the suppression pool so that the temperature inside the primary containment remains within design limits. This function is provided by two redundant RHR suppression pool cooling subsystems. The purpose of this LCO is to ensure that both subsystems are OPERABLE in applicable MODES.

Each RHR subsystem contains two pumps and one heat exchanger and is manually initiated and independently controlled. The two subsystems perform the suppression pool cooling function by circulating water from the suppression pool through the RHR heat exchangers and returning it to the suppression pool. RHR service water, circulating through the tube side of the heat exchangers, exchanges heat with the suppression pool water and discharges this heat to the external heat sink.

The heat removal capability of one RHR pump in one subsystem is sufficient to meet the overall DBA pool cooling requirement for loss of coolant accidents (LOCAs) and transient events such as a turbine trip or stuck open safety/relief valve (S/RV). S/RV leakage and High Pressure Core Injection and Reactor Core Isolation Cooling Systems testing increase suppression pool temperature more slowly. The RHR Suppression Pool Cooling System is also used to lower the suppression pool water bulk temperature following such events.

Relief Valve

1

Coolant

2

APPLICABLE SAFETY ANALYSES

Reference 1 contains the results of analyses used to predict primary containment pressure and temperature following large and small break LOCAs. The intent of the analyses is to demonstrate that the heat removal capacity of the RHR Suppression Pool Cooling System is adequate to maintain the primary containment conditions within design limits. The

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

suppression pool temperature is calculated to remain below the design limit.

The RHR Suppression Pool Cooling System satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10CFR 50.36(c)(2)(ii) 1

LCO

During a DBA, a minimum of one RHR suppression pool cooling subsystem is required to maintain the primary containment peak pressure and temperature below design limits (Ref. 1). To ensure that these requirements are met, two RHR suppression pool cooling subsystems must be OPERABLE with power from two safety related independent power supplies. Therefore, in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active failure. An RHR suppression pool cooling subsystem is OPERABLE when one of the pumps, the heat exchanger, and associated piping, valves, instrumentation, and controls are OPERABLE.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause both ~~a~~ release of radioactive material to primary containment and ~~cause~~ a heatup and pressurization of primary containment. 2 In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, the RHR Suppression Pool Cooling System is not required to be OPERABLE in MODE 4 or 5.

ACTIONS

A.1

With one RHR suppression pool cooling subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE RHR suppression pool cooling subsystem is adequate to perform the primary containment cooling function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced primary containment cooling capability. The 7 day Completion Time is acceptable in light of the redundant RHR suppression pool 3

(continued)

BASES

ACTIONS

A.1 (continued)

cooling capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

Insert ACTION B.1

TSTF-230

C

B.1 and B.2

any 4 or B TSTF-230

If the Required Action and associated Completion Time of Condition A cannot be met within the required Completion Time or if two RHR suppression pool cooling subsystems are inoperable, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 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.

4 4

SURVEILLANCE REQUIREMENTS

SR 3.6.2.3.1

Verifying the correct alignment for manual, power operated, and automatic valves in the RHR suppression pool cooling mode flow path provides assurance that the proper flow path exists for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable since the RHR suppression pool cooling mode is manually initiated. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. 4

The Frequency of 31 days is justified because the valves are operated under procedural control, improper valve position would affect only a single subsystem, the probability of an event requiring initiation of the system is low, and the
2 subsystem is a manually initiated system. This Frequency

(continued)

TSTF-230

Insert ACTION B.1

B.1

With two RHR suppression pool cooling subsystems inoperable, one subsystem must be restored to OPERABLE status within 8 hours. In this condition, there is a substantial loss of the primary containment pressure and temperature mitigation function. The 8 hour Completion Time is based on this loss of function and is considered acceptable due to the low probability of a DBA and the potential avoidance of a plant shutdown transient that could result in the need for the RHR suppression pool cooling subsystems to operate.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.3.1 (continued)

has been shown to be acceptable based on operating experience.

SR 3.6.2.3.2 required 4

5 5000 Verifying that each required RHR pump develops a flow rate ≥ ~~[7700]~~ 5000 gpm while operating in the suppression pool cooling mode with flow through the associated heat exchanger ensures that ~~pump performance has not degraded during the cycle.~~ 1 the primary containment peak pressure and temperature can be maintained below the design limits during a DBA (Ref. 1) The Flow is a normal test of centrifugal pump performance required by ASME Code, Section XI (Ref. 2). This test confirms one point on the pump design curve, and the results are indicative of overall performance. Such inservice ~~inspections~~ tests confirm component OPERABILITY, ~~trend performance~~ 7 and detect incipient failures by indicating abnormal performance. The Frequency of this SR is ~~[in~~ 5 in accordance with the Inservice Testing Program ~~or 92 days]~~. 6

REFERENCES

1. 1 UFSAR, Section ~~[~~6.2~~]~~. 5
2. ASME, Boiler and Pressure Vessel Code, Section XI.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.2.3 - RHR SUPPRESSION POOL COOLING

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. Editorial change made for enhanced clarity.

3. Changes have been made for consistency with similar phrases in other parts of the Bases.

4. Changes have been made to reflect those changes made to the Specification.

5. The brackets have been removed and the proper plant specific information/value has been provided.

6. Typographical/grammatical error corrected.

7. The IST Program at Quad Cities 1 and 2 is not required to provide information for trend purposes. Therefore, these words have been deleted.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.4 Residual Heat Removal (RHR) Suppression Pool Spray

BASES

BACKGROUND

Following a Design Basis Accident (DBA), the RHR Suppression Pool Spray System removes heat from the suppression chamber airspace. The suppression pool is designed to absorb the sudden input of heat from the primary system from a DBA or a rapid depressurization of the reactor pressure vessel (RPV) through safety/relief valves. The heat addition to the suppression pool results in increased steam in the suppression chamber, which increases primary containment pressure. Steam blowdown from a DBA can also bypass the suppression pool and end up in the suppression chamber airspace. Some means must be provided to remove heat from the suppression chamber so that the pressure and temperature inside primary containment remain within analyzed design limits. This function is provided by two redundant RHR suppression pool spray subsystems. The purpose of this LCO is to ensure that both subsystems are OPERABLE in applicable MODES. 1

Each of the two RHR suppression pool spray subsystems contains two pumps and one heat exchanger, which are manually initiated and independently controlled. The two subsystems perform the suppression pool spray function by circulating water from the suppression pool through the RHR heat exchangers and returning it to the suppression pool spray sparger~~s~~. The sparger~~s~~ only accommodate s a small portion of the total RHR pump flow; the remainder of the flow returns to the suppression pool through the suppression pool cooling return line or minimum flow line. Thus, both suppression pool cooling and suppression pool spray functions ~~are~~ may be performed when the Suppression Pool Spray System is initiated. RHR service water, circulating through the tube side of the heat exchangers, exchanges heat with the suppression pool water and discharges this heat to the external heat sink. Either RHR suppression pool spray subsystem is sufficient to condense the steam from small bypass leaks from the drywell to the suppression chamber airspace during the postulated DBA. 1 1 1

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Reference 1 contains the results of analyses used to predict primary containment pressure and temperature following large and small break loss of coolant accidents. The intent of the analyses is to demonstrate that the pressure reduction capacity of the RHR Suppression Pool Spray System is adequate to maintain the primary containment conditions within design limits. The time history for primary containment pressure is calculated to demonstrate that the maximum pressure remains below the design limit.

The RHR Suppression Pool Spray System satisfies Criterion 3 of ~~the NRC Policy Statement~~. 10 CFR 50.36(c)(2)(ii) $\boxed{1}$

LCO

In the event of a DBA, a minimum of one RHR suppression pool spray subsystem is required to mitigate potential bypass leakage paths and maintain the primary containment peak pressure below the design limits (Ref. 1). To ensure that these requirements are met, two RHR suppression pool spray subsystems must be OPERABLE with power from two safety related independent power supplies. Therefore, in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active failure. An RHR suppression pool spray subsystem is OPERABLE when one of the pumps, the heat exchanger, and associated piping, valves, instrumentation, and controls are OPERABLE.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause pressurization of primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining RHR suppression pool spray subsystems OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

With one RHR suppression pool spray subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE RHR suppression pool spray subsystem is adequate to perform the primary containment bypass leakage mitigation function. $\boxed{2}$

(continued)

BASES

ACTIONS

A.1 (continued)

However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced primary containment bypass mitigation capability. The 7 day Completion Time was chosen in light of the redundant RHR suppression pool spray capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

B.1

With both RHR suppression pool spray subsystems inoperable, at least one subsystem must be restored to OPERABLE status within 8 hours. In this ~~C~~condition, there is a substantial loss of the primary containment bypass leakage mitigation function. The 8 hour Completion Time is based on this loss of function and is considered acceptable due to the low probability of a DBA and because alternative methods to ~~remove heat from~~ primary containment are available. 2

1 reduce pressure in the

C.1 and C.2

any Required Action and

If ~~the inoperable RHR suppression pool spray subsystem cannot be restored to OPERABLE status within the~~ associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. 2

met

SURVEILLANCE REQUIREMENTS

SR 3.6.2.4.1

Verifying the correct alignment for manual, power operated, and automatic valves in the RHR suppression pool spray mode flow path provides assurance that the proper flow paths will exist for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing, or securing. A 3 2

5

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.4.1 (continued)

valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable since the RHR suppression pool ~~cooling~~ spray mode is manually initiated. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. 2

The Frequency of 31 days is justified because the valves are operated under procedural control, improper valve position would affect only a single subsystem, the probability of an event requiring initiation of the system is low, and the ~~sub~~system is a manually initiated system. This Frequency has been shown to be acceptable based on operating experience. 4

SR 3.6.2.4.2

~~Verifying each RHR pump develops a flow rate ≥ [400] gpm while operating in the suppression pool spray mode with flow through the heat exchanger ensures that pump performance has not degraded during the cycle. Flow is a normal test of centrifugal pump performance required by Section XI of the ASME Code (Ref. 2). This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of this SR is [in accordance with the Inservice Testing Program, but the Frequency must not exceed 92 days].~~ 3

This Surveillance is performed every 5 years to verify that the spray nozzles are not obstructed and that spray flow will be provided when required. The 5 year Frequency is adequate to detect degradation in performance due to the passive nozzle design and has been shown to be acceptable through operating experience.

REFERENCES

1 1. ~~U~~FSAR, Section ~~1~~6.2~~.1~~.2.2 5

~~2. ASME, Boiler and Pressure Vessel Code, Section XI.~~ 3

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.2.4 - RHR SUPPRESSION POOL SPRAY

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. These changes have been made for consistency with similar phrases in other parts of the Bases and/or to be consistent with the Specification.

3. Changes have been made to reflect those changes made to the Specification.

4. Editorial change made for enhanced clarity.

5. The brackets have been removed and the proper plant specific information/value has been provided.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.5 Drywell-to-Suppression Chamber Differential Pressure

BASES

BACKGROUND

The toroidal shaped suppression chamber, which contains the suppression pool, is connected to the drywell (part of the primary containment) by ~~(eight)~~ main vent pipes. The main vent pipes exhaust into a continuous vent header, from which ~~(96)~~ downcomer pipes extend into the suppression pool. The pipe exit is ~~(4)~~ 3.21 ft below the minimum suppression pool water level required by LCO 3.6.2.2, "Suppression Pool Water Level." During a loss of coolant accident (LOCA), the increasing drywell pressure will force the waterleg in the downcomer pipes into the suppression pool at substantial velocities as the "blowdown" phase of the event begins. The length of the waterleg has a significant effect on the resultant primary containment pressures and loads. 1

APPLICABLE SAFETY ANALYSES

The purpose of maintaining the drywell at a slightly higher pressure with respect to the suppression chamber is to minimize the drywell pressure increase necessary to clear the downcomer pipes to commence condensation of steam in the suppression pool and to minimize the mass of the accelerated water leg. This reduces the hydrodynamic loads on the torus during the LOCA blowdown. The required differential pressure results in a downcomer waterleg of ~~(3.06 to 3.58)~~ approximately 1 ft. 1

Initial drywell-to-suppression chamber differential pressure affects both the dynamic pool loads on the suppression chamber and the peak drywell pressure during downcomer pipe clearing during a Design Basis Accident LOCA. Drywell-to-suppression chamber differential pressure must be maintained within the specified limits so that the safety analysis remains valid.

Drywell-to-suppression chamber differential pressure satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). 2

LCO

A drywell-to-suppression chamber differential pressure limit of ~~(1.5)~~ 1.0 1 psid is required to ensure that the containment

(continued)

BASES

LCO (continued)

conditions assumed in the safety analyses are met. A drywell-to-suppression chamber differential pressure of ≤ ~~[0.75]~~ 1.0 psid corresponds to a downcomer water leg of ≥ ~~[3.58]~~ approximately 1 ft. Failure to maintain the required differential pressure could result in excessive forces on the suppression chamber due to higher water clearing loads from downcomer vents and higher pressure buildup in the drywell.

1

APPLICABILITY

Drywell-to-suppression chamber differential pressure must be controlled when the primary containment is inert. The primary containment must be inert in MODE 1, since this is the condition with the highest probability for an event that could produce hydrogen. It is also the condition with the highest probability of an event that could impose large loads on the primary containment.

3

A Note is provided to allow for periods of up to 4 hours when the LCO is not required to be met during the performance of required Surveillances that reduce the differential pressure. The 4 hour time is acceptable since the probability of a DBA LOCA occurring during this time is low.

Inerting primary containment is an operational problem because it prevents primary containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the unit startup and is de-inerted as soon as possible in the unit shutdown. As long as reactor power is < ~~[~~15~~]~~% RTP, the probability of an event that generates hydrogen or excessive loads on primary containment occurring within the first ~~[~~24~~]~~ hours following a startup or within the last ~~[~~24~~]~~ hours prior to a shutdown is low enough that these "windows," with the primary containment not inerted, are also justified. The ~~[~~24~~]~~ hour time period is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting. 1 4

ACTIONS

A.1

If drywell-to-suppression chamber differential pressure is not within the limit, the conditions assumed in the safety analyses are not met and the differential pressure must be restored to within the limit within ~~8~~ 24 hours. The ~~8~~ 24 hour Completion Time provides sufficient time to restore differential pressure to within limit and takes into account the low probability of an event that would create excessive suppression chamber loads occurring during this time period. 3

(continued)

BASES

ACTIONS (continued)

B.1

1

If the differential pressure cannot be restored to within limits within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by reducing power to ≤ ~~[15]%~~ RTP within ~~(12)~~ hours. The ~~(12)~~ hour Completion Time is reasonable, based on operating experience, to reduce reactor power from full power conditions in an orderly manner and without challenging plant systems.

3 8

8 3

SURVEILLANCE REQUIREMENTS

SR 3.6.2.5.1

The drywell-to-suppression chamber differential pressure is regularly monitored to ensure that the required limits are satisfied. The 12 hour Frequency of this SR was developed based on operating experience relative to differential pressure variations and pressure instrument drift during applicable MODES and by assessing the proximity to the specified LCO differential pressure limit. 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 pressure condition.

REFERENCES

None.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.2.5 - DRYWELL-TO-SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

3. Changes have been made to reflect those changes made to the Specification.

4. Typographical/grammatical error corrected.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.1 Primary Containment Hydrogen Recombiners

1

BASES

BACKGROUND

The primary containment hydrogen recombiner eliminates the potential breach of primary containment due to a hydrogen oxygen reaction and is part of combustible gas control required by 10 CFR 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Reactors" (Ref. 1), and GDC 41, "Containment Atmosphere Cleanup" (Ref. 2). The primary containment hydrogen recombiner is required to reduce the hydrogen concentration in the primary containment following a loss of coolant accident (LOCA). The primary containment hydrogen recombiner accomplishes this by recombining hydrogen and oxygen to form water vapor. The vapor remains in the primary containment, thus eliminating any discharge to the environment. The primary containment hydrogen recombiner is manually initiated since flammability limits would not be reached until several days after a Design Basis Accident (DBA).

The primary containment hydrogen recombiner functions to maintain the hydrogen gas concentration within the containment at or below the flammability limit of 4.0 volume percent (v/o) following a postulated LOCA. It is fully redundant and consists of two 100% capacity subsystems. Each primary containment hydrogen recombiner consists of an enclosed blower assembly, heater section, reaction chamber, direct contact water spray gas cooler, water separator, and associated piping, valves, and instruments. The primary containment hydrogen recombiner will be manually initiated from the main control room when the hydrogen gas concentration in the primary containment reaches [3.3] v/o. When the primary containment is inerted (oxygen concentration < 4.0 v/o), the primary containment hydrogen recombiner will only function until the oxygen is used up (2.0 v/o hydrogen combines with 1.0 v/o oxygen). Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safety Feature bus and is provided with separate power panel and control panel.

The process gas circulating through the heater, the reaction chamber, and the cooler is automatically regulated to [150] scfm by the use of an orifice plate installed in the

(continued)

1

BASES

BACKGROUND (continued)

cooler. The process gas is heated to [1200]°F. The hydrogen and oxygen gases are recombined into water vapor, which is then condensed in the water spray gas cooler by the associated residual heat removal subsystem and discharged with some of the effluent process gas to the suppression chamber. The majority of the cooled, effluent process gas is mixed with the incoming process gas to dilute the incoming gas prior to the mixture entering the heater section.

APPLICABLE SAFETY ANALYSES

The primary containment hydrogen recombiner provides the capability of controlling the bulk hydrogen concentration in primary containment to less than the lower flammable concentration of 4.0 v/o following a DBA. This control would prevent a primary containment wide hydrogen burn, thus ensuring that pressure and temperature conditions assumed in the analysis are not exceeded. The limiting DBA relative to hydrogen generation is a LOCA.

Hydrogen may accumulate in primary containment following a LOCA as a result of:

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

b. Radiolytic decomposition of water in the Reactor Coolant System.

To evaluate the potential for hydrogen accumulation in primary containment following a LOCA, the hydrogen generation is calculated as a function of time following the initiation of the accident. Assumptions recommended by Reference 3 are used to maximize the amount of hydrogen calculated.

The calculation confirms that when the mitigating systems are actuated in accordance with emergency procedures, the peak hydrogen concentration in the primary containment is < 4.0 v/o (Ref. 4).

The primary containment hydrogen recombiners satisfy Criterion 3 of the NRC Policy Statement.

(continued)

1

BASES (continued)

LCO

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

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

APPLICABILITY

In MODES 1 and 2, the two primary containment hydrogen recombiners are required to control the hydrogen concentration within primary containment below its flammability limit of 4.0 v/o following a LOCA, assuming a worst case single failure.

In MODE 3, 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 this MODE, the probability of an accident requiring the primary containment hydrogen recombiner is low. Therefore, the primary containment hydrogen recombiner is not required in MODE 3.

In MODES 4 and 5, the probability and consequences of a LOCA are low due to the pressure and temperature limitations in these MODES. Therefore, the primary containment hydrogen recombiner is not required in these MODES.

ACTIONS

A.1

With one primary containment 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 reduced hydrogen control capability. The 30 day Completion Time is based on the low probability of the occurrence of a LOCA that would generate hydrogen in amounts capable of exceeding the flammability limit, the amount of time available after the event for operator action

(continued)

[1]

BASES

ACTIONS

A.1 (continued)

to prevent exceeding this limit, and the low probability of failure of the OPERABLE primary containment hydrogen recombiner.

Required Action A.1 has been modified by a Note indicating that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one recombiner is inoperable. This allowance is provided because of the low probability of the occurrence of a LOCA that would generate hydrogen in amounts capable of exceeding the flammability limit, the low probability of the failure of the OPERABLE subsystem, and the amount of time available after a postulated LOCA for operator action to prevent exceeding the flammability limit.

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.]

With two primary containment 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 [Primary Containment Inerting System or one subsystem of the Containment Atmosphere Dilution 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 once per 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)

1

BASES

ACTIONS

B.1 and B.2 (continued)

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 amounts capable of exceeding the flammability limit.

C.1

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

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1.1

Performance of a system functional test for each primary containment hydrogen recombiner ensures that the recombiners are OPERABLE and can attain and sustain the temperature necessary for hydrogen recombination. In particular, this SR verifies that the minimum heater sheath temperature increases to ≥ [1200]°F in ≤ [1.5] hours and that it is maintained ≥ [1150]°F and < [1300]°F for ≥ [4] hours thereafter to check the ability of the recombiner to function properly (and to make sure that significant heater elements are not burned out). 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 3.6.3.1.2

This SR ensures there are no physical problems that could affect recombiner operation. Since the recombiners are

(continued)

1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1.2 (continued)

mechanically passive, except for the blower assemblies, they are subject to only minimal mechanical failure. The only credible failures involve loss of power or blower function, blockage of the internal flow path, missile impact, etc.

A visual inspection is sufficient to determine abnormal conditions that could cause such failures. 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 3.6.3.1.3

This SR requires performance of a resistance to ground test of each heater phase to make sure 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.

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. 10 CFR 50.44.
2. 10 CFR 50, Appendix A, GDC 41.
3. Regulatory Guide 1.7, Revision [1].
4. FSAR, Section [6.2.5].

1. The Bases has been deleted since the associated Specification has been deleted.

1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.2 [Drywell Cooling System Fans]

BASES

BACKGROUND

The [Drywell Cooling System fans] ensure a uniformly mixed post accident primary containment atmosphere, thereby minimizing the potential for local hydrogen burns due to a pocket of hydrogen above the flammable concentration.

The [Drywell Cooling System fans] are an Engineered Safety Feature and are designed to withstand a loss of coolant accident (LOCA) in post accident environments without loss of function. The system has two independent subsystems consisting of fans, fan coil units, motors, controls, and ducting. Each subsystem is sized to circulate [500] scfm. The [Drywell Cooling System fans] employ both forced circulation and natural circulation to ensure the proper mixing of hydrogen in primary containment. The recirculation fans provide the forced circulation to mix hydrogen while the fan coils provide the natural circulation by increasing the density through the cooling of the hot gases at the top of the drywell causing the cooled gases to gravitate to the bottom of the drywell. The two subsystems are initiated manually since flammability limits would not be reached until several days after a LOCA. Each subsystem is powered from a separate emergency power supply. Since each subsystem can provide 100% of the mixing requirements, the system will provide its design function with a worst case single active failure.

The [Drywell Cooling System fans] use the Drywell Cooling System recirculating fans to mix the drywell atmosphere. The fan coil units and recirculation fans are automatically disengaged during a LOCA but may be restored to service manually by the operator. In the event of a loss of offsite power, all fan coil units, recirculating fans, and primary containment water chillers are transferred to the emergency diesels. The fan coil units and recirculating fans are started automatically from diesel power upon loss of offsite power.

(continued)

1

BASES (continued)

APPLICABLE SAFETY ANALYSES

The [Drywell Cooling System fans] provide the capability for reducing the local hydrogen concentration to approximately the bulk average concentration following a Design Basis Accident (DBA). The limiting DBA relative to hydrogen generation is a LOCA.

Hydrogen may accumulate in primary containment following a LOCA as a result of:

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

b. Radiolytic decomposition of water in the Reactor Coolant System.

To evaluate the potential for hydrogen accumulation in primary containment following a LOCA, the hydrogen generation as a function of time following the initiation of the accident is calculated. Conservative assumptions recommended by Reference 1 are used to maximize the amount of hydrogen calculated.

The Reference 2 calculations show that hydrogen assumed to be released to the drywell within 2 minutes following a DBA LOCA raises drywell hydrogen concentration to over 2.5 volume percent (v/o). Natural circulation phenomena result in a gradient concentration difference of less then 0.5 v/o in the drywell and less than 0.1 v/o in the suppression chamber. Even though this gradient is acceptably small and no credit for mechanical mixing was assumed in the analysis, two [Drywell Cooling System fans] are [required] to be OPERABLE (typically four to six fans are required to keep the drywell cool during operation in MODE 1 or 2) by this LCO.

The [Drywell Cooling System fans] satisfy Criterion 3 of the NRC Policy Statement.

LCO

Two [Drywell Cooling System fans] must be OPERABLE to ensure operation of at least one fan in the event of a worst case single active failure. Each of these fans must be powered from an independent safety related bus.

(continued)

1

BASES

LCO (continued)

Operation with at least one fan provides the capability of controlling the bulk hydrogen concentration in primary containment without exceeding the flammability limit.

APPLICABILITY

In MODES 1 and 2, the two [Drywell Cooling System fans] ensure the capability to prevent localized hydrogen concentrations above the flammability limit of 4.0 v/o in drywell, assuming a worst case single active failure.

In MODE 3, 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 this MODE, the probability of an accident requiring the [Drywell Cooling System fans] is low. Therefore, the [Drywell Cooling System fans] are not required in MODE 3.

In MODES 4 and 5, the probability and consequences of a LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, the [Drywell Cooling System fans] are not required in these MODES.

ACTIONS

A.1

With one [required] [Drywell Cooling System fan] inoperable, the inoperable fan must be restored to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE fan is adequate to perform the hydrogen mixing function. However, the overall reliability is reduced because a single failure in the OPERABLE fan could result in reduced hydrogen mixing capability. The 30 day Completion Time is based on the availability of the second fan, the low probability of the occurrence of a LOCA that would generate hydrogen in amounts capable of exceeding the flammability limit, the amount of time available after the event for operator action to prevent exceeding this limit, and the availability of the Primary Containment Hydrogen Recombiner System and the Containment Atmosphere Dilution System.

Required Action A.1 has been modified by a Note indicating that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one [Drywell Cooling System fan] is inoperable. This allowance is provided

(continued)

BASES

ACTIONS

A.1 (continued)

because of the low probability of the occurrence of a LOCA that would generate hydrogen in amounts capable of exceeding the flammability limit, the low probability of the failure of the OPERABLE fan, and the amount of time available after a postulated LOCA for operator action to prevent exceeding the flammability limit.

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.]

With two [Drywell Cooling System fans] 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 [Primary Containment Inerting System or one subsystem of the Containment Atmosphere Dilution 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 once per 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 [Drywell Cooling System fans] inoperable for up to 7 days. Seven days is a reasonable time to allow two [Drywell Cooling System fans] 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 amounts capable of exceeding the flammability limit.

(continued)

1

1

BASES

ACTIONS (continued)

C.1

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

SURVEILLANCE REQUIREMENTS

SR 3.6.3.2.1

Operating each [required] [Drywell Cooling System fan] for ≥ 15 minutes ensures that each subsystem is 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 92 day Frequency is consistent with the Inservice Testing Program Frequencies, operating experience, the known reliability of the fan motors and controls, and the two redundant fans available.

[SR 3.6.3.2.2

Verifying that each [required] [Drywell Cooling System fan] flow rate is ≥ [500] scfm ensures that each fan is capable of maintaining localized hydrogen concentrations below the flammability limit. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown 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. Regulatory Guide 1.7, Revision [1].
2. FSAR, Section [6.2.5].

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ISTS BASES: 3.6.3.2 - DRYWELL COOLING SYSTEM FANS

1. This Bases has been deleted since the associated Specification has been deleted.

1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.3 1 Primary Containment Oxygen Concentration

BASES

2

BACKGROUND

The primary containment is ~~All nuclear reactors must be~~ designed to withstand events that generate hydrogen either due to the zirconium metal water reaction in the core or due to radiolysis. The primary method to control hydrogen is to inert the primary containment. With the primary containment inert ed ~~that is,~~ oxygen concentration < 4.0 volume percent (v/o), a combustible mixture cannot be present in the primary containment for any hydrogen concentration. The capability to inert the primary containment and maintain oxygen < 4.0 v/o ~~works together with the Hydrogen Recombiner System (LCO 3.6.3.1, "Primary Containment Hydrogen Recombiners") and the [Drywell Cooling System fans] (LCO 3.6.3.2, "[Drywell Cooling System Fans]") to~~ provide a ~~redundant and diverse~~ method~~s~~ to mitigate events that produce hydrogen and oxygen. For example, an event that rapidly generates hydrogen from zirconium metal water reaction will result in excessive hydrogen in primary containment, but oxygen concentration will remain < 4.0 v/o and no combustion can occur. Long term generation of both hydrogen and oxygen from radiolytic decomposition of water may eventually result in a combustible mixture in primary containment~~, except that the hydrogen recombiners remove hydrogen and oxygen gases faster than they can be produced from radiolysis and again no combustion can occur~~. Insert BKGD This LCO ensures that oxygen concentration does not exceed 4.0 v/o during operation in the applicable conditions.

3 3

APPLICABLE SAFETY ANALYSES

The Reference ~~1~~ 2 calculations assume that the primary containment is inerted when a Design Basis Accident loss of coolant accident occurs. Thus, the hydrogen assumed to be released to the primary containment as a result of metal water reaction in the reactor core will not produce combustible gas mixtures in the primary containment. Oxygen, which is subsequently generated by radiolytic decomposition of water, ~~is recombined by the hydrogen recombiners (LCO 3.6.3.1) more rapidly than it is produced~~ will not result in the primary containment becoming deinerted within the first 30 days following an accident.

3 3

Primary containment oxygen concentration satisfies ~~Criterion 2 of the NRC Policy Statement~~ 10CFR50.36(c)(2)(ii).

(continued)

3

Insert BKGD

Radiolysis is the only significant reaction mechanism whereby oxygen, the limiting combustion reactant, is produced within the containment. The Technical Specification requirement to inert the primary containment and maintain oxygen < 4.0 v/o, in conjunction with the elimination of potential sources of air and oxygen (other than by radiolysis) from entering the primary containment provide assurance that the amount of oxygen that could be introduced into the containment will not cause the containment to become de-inerted within the first 30 days after an accident. This is consistent with the requirements of Generic Letter 84-09 (Ref. 1) for plants without recombiners.

BASES (continued)

LCO

The primary containment oxygen concentration is maintained < 4.0 v/o to ensure that an event that produces any amount of hydrogen and oxygen 3 does not result in a combustible mixture inside primary containment.

APPLICABILITY

The primary containment oxygen concentration must be within the specified limit when primary containment is inerted, except as allowed by the relaxations during startup and shutdown addressed below. The primary containment must be inert in MODE 1, since this is the condition with the highest probability of an event that could produce hydrogen and oxygen 3.

Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and de-inerted as soon as possible in the plant shutdown. As long as reactor power is < 15% RTP, the potential for an event that generates significant hydrogen and oxygen is low and the primary containment need not be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first [24] hours of a startup, or within the last [24] hours before a shutdown, is low enough that these "windows," when the primary containment is not inerted, are also justified. The [24] hour time period is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting. 4

ACTIONS

A.1

If oxygen concentration is ≥ 4.0 v/o at any time while operating in MODE 1, with the exception of the relaxations allowed during startup and shutdown, oxygen concentration must be restored to < 4.0 v/o within 24 hours. The 24 hour Completion Time is allowed when oxygen concentration is ≥ 4.0 v/o because of the availability of other hydrogen and oxygen 3 mitigating systems (e.g., ~~hydrogen recombiners~~ post-accident nitrogen purge) and the low probability and long duration of an event that would generate significant amounts of hydrogen and oxygen 3 occurring during this period.

(continued)

BASES

ACTIONS (continued)

B.1

If oxygen concentration cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, power must be reduced to ≤ 15% RTP within 8 hours. The 8 hour Completion Time is reasonable, based on operating experience, to reduce reactor power from full power conditions in an orderly manner and without challenging plant systems. 4

SURVEILLANCE REQUIREMENTS

SR 3.6.3.3.1 1 1

The primary containment must be determined to be inert ed by verifying that oxygen concentration is < 4.0 v/o. The 7 day Frequency is based on the slow rate at which oxygen concentration can change and on other indications of abnormal conditions (which ~~would~~ could lead to more frequent checking by operators in accordance with plant procedures). Also, this Frequency has been shown to be acceptable through operating experience. 5

REFERENCES

1. Generic Letter 84-09, May 1984.
2. 1. UFSAR, Section 6.2.5. 4
3

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.3.1 - PRIMARY CONTAINMENT OXYGEN CONCENTRATION

1. The Bases has been renumbered due to the deletion of ISTS Bases 3.6.3.1 and 3.6.3.2.

2. Editorial change made for enhanced clarity.

3. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

4. The brackets have been removed and the proper plant specific information/value has been provided.

5. Typographical/grammatical error corrected.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.4 Containment Atmosphere Dilution (CAD) System

1

BASES

BACKGROUND

The CAD System functions to maintain combustible gas concentrations within the primary containment at or below the flammability limits following a postulated loss of coolant accident (LOCA) by diluting hydrogen and oxygen with nitrogen. To ensure that a combustible gas mixture does not occur, oxygen concentration is kept < [5.0] volume percent (v/o), or hydrogen concentration is kept < 4.0 v/o.

The CAD System is manually initiated and consists of two independent, 100% capacity subsystems. Each subsystem includes a liquid nitrogen supply tank, ambient vaporizer, electric heater, and connected piping to supply the drywell and suppression chamber volumes. The nitrogen storage tanks each contain ≥ [4350] gal, which is adequate for [7] days of CAD subsystem operation.

The CAD System operates in conjunction with emergency operating procedures that are used to reduce primary containment pressure periodically during CAD System operation. This combination results in a feed and bleed approach to maintaining hydrogen and oxygen concentrations below combustible levels.

APPLICABLE SAFETY ANALYSES

To evaluate the potential for hydrogen and oxygen accumulation in primary containment following a LOCA, hydrogen and oxygen generation is calculated (as a function of time following the initiation of the accident). The assumptions stated in Reference 1 are used to maximize the amount of hydrogen and oxygen generated. The calculation confirms that when the mitigating systems are actuated in accordance with emergency operating procedures, the peak oxygen concentration in primary containment is < [5.0] v/o (Ref. 2).

Hydrogen and oxygen may accumulate within primary containment following a LOCA as a result of:

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

(continued)

1

BASES

APPLICABLE SAFETY ANALYSES (continued)	b. Radiolytic decomposition of water in the Reactor Coolant System. The CAD System satisfies Criterion 3 of the NRC Policy Statement.
LCO	Two CAD subsystems must be OPERABLE. This ensures operation of at least one CAD subsystem in the event of a worst case single active failure. Operation of at least one CAD subsystem is designed to maintain primary containment post-LOCA oxygen concentration < 5.0 v/o for 7 days.
APPLICABILITY	In MODES 1 and 2, the CAD System is required to maintain the oxygen concentration within primary containment below the flammability limit of 5.0 v/o following a LOCA. This ensures that the relative leak tightness of primary containment is adequate and prevents damage to safety related equipment and instruments located within primary containment. In MODE 3, both the hydrogen and oxygen production rates and the total amounts produced after a LOCA would be less than those calculated for the Design Basis Accident LOCA. Thus, if the analysis were to be performed starting with a LOCA in MODE 3, the time to reach a flammable concentration would be extended beyond the time conservatively calculated for MODES 1 and 2. The extended time would allow hydrogen removal from the primary containment atmosphere by other means and also allow repair of an inoperable CAD subsystem, if CAD were not available. Therefore, the CAD System is not required to be OPERABLE in MODE 3. In MODES 4 and 5, the probability and consequences of a LOCA are reduced due to the pressure and temperature limitations of these MODES. Therefore, the CAD System is not required to be OPERABLE in MODES 4 and 5.

(continued)

1

BASES (continued)

ACTIONS

A.1

If one CAD subsystem is inoperable, it must be restored to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE CAD subsystem is adequate to perform the oxygen control function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced oxygen control capability. The 30 day Completion Time is based on the low probability of the occurrence of a LOCA that would generate hydrogen and oxygen in amounts capable of exceeding the flammability limit, the amount of time available after the event for operator action to prevent exceeding this limit, and the availability of the OPERABLE CAD subsystem and other hydrogen mitigating systems.

Required Action A.1 has been modified by a Note that indicates that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one CAD subsystem is inoperable. This allowance is provided because of the low probability of the occurrence of a LOCA that would generate hydrogen and oxygen in amounts capable of exceeding the flammability limit, the low probability of the failure of the OPERABLE subsystem, the amount of time available after a postulated LOCA for operator action to prevent exceeding the flammability limit, and the availability of other hydrogen mitigating systems.

B.1 and B.2

[Reviewer's Note: This Condition is only allowed for plants with an alternate hydrogen control system acceptable to the technical staff.]

With two CAD subsystems 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 [Primary Containment Inerting System or one hydrogen recombiner and one Drywell Cooling System fan]. 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,

(continued)

1

BASES

ACTIONS

B.1 and B.2 (continued)

the alternate hydrogen control system capability must be verified once per 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 CAD subsystems inoperable for up to 7 days. Seven days is a reasonable time to allow two CAD subsystems 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 amounts capable of exceeding the flammability limit.

With two CAD subsystems inoperable, one CAD subsystem must be restored to OPERABLE status within 7 days. The 7 day Completion Time is based on the low probability of the occurrence of a LOCA that would generate hydrogen in the amounts capable of exceeding the flammability limit, the amount of time available after the event for operator action to prevent exceeding this limit, and the availability of other hydrogen mitigating systems.

C.1

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

(continued)

1

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.3.4.1

Verifying that there is ≥ [4350] gal of liquid nitrogen supply in the CAD System will ensure at least [7] days of post-LOCA CAD operation. This minimum volume of liquid nitrogen allows sufficient time after an accident to replenish the nitrogen supply for long term inerting. This is verified every 31 days to ensure that the system is capable of performing its intended function when required. The 31 day Frequency is based on operating experience, which has shown 31 days to be an acceptable period to verify the liquid nitrogen supply and on the availability of other hydrogen mitigating systems.

SR 3.6.3.4.2

Verifying the correct alignment for manual, power operated, and automatic valves in each of the CAD subsystem flow paths provides assurance that the proper flow paths exist for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing.

A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable because the CAD System is manually initiated. This SR 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 that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is appropriate because the valves are operated under procedural control, improper valve position would only affect a single subsystem, the probability of an event requiring initiation of the system is low, and the system is a manually initiated system.

REFERENCES

1. Regulatory Guide 1.7, Revision [2].
2. FSAR, Section [ ].

1. This Bases has been deleted since the associated Specification has been deleted.

1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 [Secondary] Containment

BASES

BACKGROUND

The function of the [secondary] containment is to contain, dilute, and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA). In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines penetrate the [secondary] containment, the [secondary] containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside primary containment.

both

s

The [secondary] containment is ~~a~~ structure that completely encloses the primary containment, including the MSIV rooms, and those components that may be postulated to contain primary system fluid. This structure forms a control volume that serves to hold up and dilute the fission products. It is possible for the pressure in the control volume to rise relative to the environmental pressure (e.g., due to pump and motor heat load additions). To prevent ground level exfiltration while allowing the [secondary] containment to be designed as a conventional structure, the [secondary] containment requires support systems to maintain the control volume pressure at less than the external pressure. Requirements for these systems are specified separately in LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System." 2 1

APPLICABLE SAFETY ANALYSES

There are two principal accidents for which credit is taken for [secondary] containment OPERABILITY. These are a loss of coolant accident (LOCA) (Ref. 1) and a fuel handling accident ~~inside [secondary] containment~~ (Ref. 2). The [secondary] containment performs no active function in response to each of these limiting events; however, its leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis and that fission 2 1

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

products entrapped within the [secondary] containment structure will be treated by the SGT System prior to discharge to the environment.

[Secondary] containment satisfies Criterion 3 of ~~the NRC Policy Statement~~. 10CFR 50.36(c)(2)(ii) 2

LCO

An OPERABLE [secondary] containment provides a control volume into which fission products that bypass or leak from primary containment, or are released from the reactor coolant pressure boundary components located in [secondary] containment, can be diluted and processed prior to release to the environment. For the [secondary] containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained. Insert LCO 3

APPLICABILITY

In MODES 1, 2, and 3, a LOCA could lead to a fission product release to primary containment that leaks to [secondary] containment. Therefore, [secondary] containment OPERABILITY is required during the same operating conditions that require primary containment OPERABILITY.

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining [secondary] containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the [secondary] containment.

ACTIONS

A.1

If [secondary] containment is inoperable, it must be restored to OPERABLE status within 4 hours. The 4 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of

(continued)

3

Insert LCO

, the hatches and blowout panels must be closed and sealed, the sealing mechanisms (e.g., welds, bellows, or O-rings) associated with each secondary containment penetration must be OPERABLE (such that secondary containment leak tightness can be maintained), and all inner or all outer doors in each secondary containment access opening must be closed.

BASES

ACTIONS

A.1 (continued)

maintaining [secondary] containment during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring [secondary] containment OPERABILITY) occurring during periods where [secondary] containment is inoperable is minimal.

B.1 and B.2

If [secondary] 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 12 hours and to MODE 4 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.

1

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

Movement of irradiated fuel assemblies in the [secondary] containment, CORE ALTERATIONS, and OPDRVs can be postulated to cause fission product release to the [secondary] containment. In such cases, the [secondary] containment is the only barrier to release of fission products to the environment. CORE ALTERATIONS and movement of irradiated fuel assemblies must be immediately suspended if the [secondary] containment is inoperable.

Suspension of these activities shall not preclude completing an action that involves moving a component to a safe position. Also, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

Required Action C.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. ~~Therefore, in either case, inability to suspend~~

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

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3,

4

(continued)

4 Insert C.1, C.2, and C.3

Entering LCO 3.0.3 while in MODE 1, 2, or 3 would require the unit to be shutdown, but would not require immediate suspension of movement of irradiated fuel assemblies. The Note to the ACTIONS, "LCO 3.0.3 is not applicable," ensures that the actions for immediate suspension of irradiated fuel assembly movement are not postponed due to entry into LCO 3.0.3.

BASES

ACTIONS

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

~~movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.~~ 4

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1

This SR ensures that the {secondary} containment boundary is sufficiently leak tight to preclude exfiltration under expected wind conditions. The 24 hour Frequency of this SR was developed based on operating experience related to {secondary} containment vacuum variations during the applicable MODES and the low probability of a DBA occurring between surveillances. 1

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 {secondary} containment vacuum condition.

TSTF-18 3 ~~SR 3.6.4.1.2 and~~ SR 3.6.4.1.3 2 3 3 one 1 is 3

Verifying that one {secondary} containment ~~equipment hatches and~~ access door ~~s~~ ~~are~~ in each access opening is closed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the {secondary} containment will not occur. ~~In this application, the term "sealed" has no connotation of leak tightness.~~ 3 Maintaining {secondary} containment OPERABILITY requires verifying ~~each~~ one door in the access opening is closed~~, except when the access opening is being used for normal transient entry and exit (then at least one door must remain closed)~~. The 31 day Frequency for these SRs has been shown to be adequate, based on operating experience, and is considered adequate in view of the other indications of door ~~and hatch~~ status that are available to the operator. 3

Insert B 3.6.4.1.2

TSTF-18

(continued)

TSTF-18

Insert B 3.6.4.1.2

An access opening contains one inner and one outer door. In some cases ~~[secondary] containment access openings are shared such that~~ a [secondary] containment barrier ~~may have~~ contains multiple inner or multiple outer doors. For these cases, the access openings share the inner door or the outer door, i.e., the access openings have a common inner door or outer door. The intent is to not breach the [secondary] containment at any time when [secondary] containment is required. This is achieved by maintaining the inner or outer portion of the barrier closed at all times; i.e., all inner doors closed or all outer doors closed. Thus, each access opening has one door closed. However, all [secondary] containment access doors are normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access opening.

1

6

6

6

6

All changes are [1] unless otherwise indicated

Each SGT subsystem is designed to maintain the secondary containment at ≥ 0.25 inches of vacuum water gauge for 1 hour at a flow rate of ≤ 4000 cfm.

BASES

released to secondary containment

can be maintained

SURVEILLANCE REQUIREMENTS (continued)

[1] [3]

[SR 3.6.4.1.4 and SR 3.6.4.1.5]

The SGT System exhausts the [secondary] containment atmosphere to the environment through appropriate treatment equipment. To ensure that all fission products are treated, SR 3.6.4.1.4 verifies that the SGT System will rapidly establish and maintain a pressure in the [secondary] containment that is less than the lowest postulated pressure external to the [secondary] containment boundary. This is confirmed by demonstrating that one SGT subsystem will draw down the [secondary] containment to ≥ [0.25] inches of vacuum water gauge in ≤ [120] seconds. This cannot be accomplished if the [secondary] containment boundary is not intact. SR 3.6.4.1.5 demonstrates that one SGT subsystem can maintain ≥ [0.25] inches of vacuum water gauge for 1 hour at a flow rate ≤ [4000] cfm. The 1 hour test period allows [secondary] containment to be in thermal equilibrium at steady state conditions. Therefore, these two tests are used to ensure [secondary] containment boundary integrity. Since these SRs are [secondary] containment tests, they need not be performed with each SGT subsystem. The SGT subsystems are tested on a STAGGERED TEST BASIS, however, to ensure that in addition to the requirements of LCO 3.6.4.3, either SGT subsystem will perform this test. Operating experience has shown 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.

③ [3]

When the SGT System is operating as designed, the maintenance of secondary containment pressure

[5]

the pressure in the secondary containment can be maintained

using one SGT subsystem

[4]

secondary containment boundary

24

es

REFERENCES

1. UFSAR, Section [15.1.39]. 15.6.5

2. UFSAR, Section [15.1.41]. 15.7.2

[1]

[2]

The inoperability of the SGT System does not constitute a failure of this Surveillance relative to secondary containment OPERABILITY.

[5]

used for this Surveillance is

[5]

The primary purpose of the SR is to ensure secondary containment boundary integrity. The secondary purpose of the SR is to ensure that the SGT subsystem being tested functions as designed. There is a separate LCO with Surveillance Requirements that serves the primary purpose of ensure OPERABILITY of the SGT System. This SR

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

3. Changes have been made to reflect those changes made to the Specification.

4. These changes have been made for consistency with similar phrases in other parts of the Bases and/or to be consistent with the Specification.

5. ISTS SR 3.6.4.1.5 is a test that ensures the Secondary Containment is OPERABLE; the leak tightness of the Secondary Containment boundary is within the assumptions of the accident analyses. However, it is written in such a manner that it implies that if a SGT subsystem is inoperable, the SR is failed ("Verify each standby gas treatment (SGT) subsystem can..."). As stated above, this is not the intent of the SR. Therefore, to ensure this misinterpretation cannot occur, the SR and this Bases description have been rephrased to more clearly convey the original intent of the SR, to verify the Secondary Containment is OPERABLE. With the new wording, if a SGT subsystem is inoperable, ITS SR 3.6.4.1.3 will still be met and only the SGT System Specification, LCO 3.6.4.3, will be required to be entered. This is clearly identified in the Bases.

6. The Bases have been modified to provide additional clarity when describing the design of each access opening.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

BASES

BACKGROUND

The function of the SCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Ref~~s~~. 1 ~~and 2~~). Secondary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that fission products that leak from primary containment following a DBA, or that are released during certain operations when primary containment is not required to be OPERABLE or take place outside primary containment, are maintained within the secondary containment boundary. [1]

The OPERABILITY requirements for SCIVs help ensure that an adequate ~~[secondary]~~ containment boundary is maintained during and after an accident by minimizing potential paths to the environment. 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), and blind flanges are considered passive devices. [2]

Automatic SCIVs (i.e., dampers) close on a ~~[secondary]~~ containment isolation signal to establish a boundary for untreated radioactive material within ~~[secondary]~~ containment following a DBA or other accidents. [1] [2]

Other penetrations required to be closed during accident conditions are isolated by the use of valves in the closed position or blind flanges. [1]

APPLICABLE SAFETY ANALYSES

The SCIVs must be OPERABLE to ensure the ~~[secondary]~~ containment barrier to fission product releases is established. The principal accidents for which the ~~[secondary]~~ containment boundary is required are a loss of coolant accident (Ref. 1) and a fuel handling accident ~~inside [secondary] containment~~ (Ref. 2). The ~~[secondary]~~ containment performs no active function in response to either of these limiting events, but the boundary [2] [1]

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

established by SCIVs is required to ensure that leakage from the primary containment is processed by the Standby Gas Treatment (SGT) System before being released to the environment.

Maintaining SCIVs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment so that they can be treated by the SGT System prior to discharge to the environment. 2

SCIVs satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10CFR 50.36(c)(2)(ii). 1

LCO

SCIVs form a part of the secondary containment boundary. 2 The SCIV safety function is related to control of offsite radiation releases resulting from DBAs.

The power operated, automatic TSTF-46 isolation valves are considered OPERABLE when their isolation times are within limits and the valves actuate on an automatic isolation signal. The valves covered by this LCO, along with their associated stroke times, are listed in ~~Reference 3~~ the Technical Requirements Manual (Ref. 3).

The normally closed 8 manual SCIVs ~~isolation valves or blind flanges~~ and blind flanges in place 8 are considered OPERABLE when ~~manual~~ 8 the valves are closed or open ~~in accordance with appropriate~~ under administrative controls, ~~automatic SCIVs are de-activated and secured in their closed position~~ 7, ~~and blind flanges are in place~~ B. These passive isolation valves or devices are listed in Reference 3.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the secondary containment. Therefore, the OPERABILITY of SCIVs is required. 2

In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIVs OPERABLE is not required in MODE 4 or 5, except for other situations under which significant radioactive releases can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), during CORE

(continued)

BASES

APPLICABILITY (continued)

ALTERATIONS, or during movement of irradiated fuel assemblies in the [secondary] containment. Moving irradiated fuel assemblies in the [secondary] containment may also occur in MODES 1, 2, and 3. 2 1

ACTIONS

The ACTIONS are modified by three Notes. The first Note allows penetration flow paths to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for [secondary] containment isolation is indicated. 2

3

The second Note provides clarification that, for the purpose of 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 SCIV. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SCIVs are governed by subsequent Condition entry and application of associated Required Actions.

The third Note ensures appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by an inoperable SCIV.

A.1 and A.2

In the event that there are one or more penetration flow paths with one SCIV inoperable, the affected penetration flow path(s) 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 SCIV, a closed manual valve, and a blind flange. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available device to [secondary] containment. The Required Action must be completed within the 8 hour Completion Time. The specified time period is reasonable considering the time required to 2

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

isolate the penetration, and the probability of a DBA, which requires the SCIVs to close, occurring during this short time is very low.

For affected penetrations that have been isolated in accordance with Required Action A.1, the affected penetration must be verified to be isolated on a periodic basis. This is necessary to ensure that secondary containment penetrations required to be isolated following an accident, but no longer capable of being automatically isolated, will be in the isolation position should an event occur. The Completion Time of once per 31 days is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low. This Required Action does not require any testing or device manipulation. Rather, it involves verification that the affected penetration remains isolated.

2

isolation devices

4

4

isolation

two 3 Note 1

Required Action A.2 is modified by a Note that applies to devices located in high radiation areas and allows them to be verified closed by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment, once they have been verified to be in the proper position, is low.

TSTF-269

Insert A1 and A.2

B.1

With two SCIVs in one or more penetration flow paths inoperable, the affected penetration flow path must be isolated within 4 hours. 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 4 hour Completion Time is reasonable considering the time required to isolate the penetration and the probability of a DBA, which requires the SCIVs to close, occurring during this short time, is very low.

The Condition has been modified by a Note stating that Condition B is only applicable to penetration flow paths

(continued)

TSTF-269

Insert A.1 and A.2

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.

BASES

ACTIONS

B.1 (continued)

with two isolation valves. This clarifies that only Condition A is entered if one SCIV is inoperable in each of two penetrations.

C.1 and C.2

If any Required Action and associated Completion Time cannot be met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 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.

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

If any Required Action and associated Completion Time are not met, the plant must be placed in a condition in which the LCO does not apply. If applicable, CORE ALTERATIONS and the movement of irradiated fuel assemblies in the secondary containment must be immediately suspended. 2 Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

Required Action D.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving fuel while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. ~~Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.~~

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

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, 4

(continued)

4

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

Entering LCO 3.0.3 while in MODE 1, 2, or 3 would require the unit to be shutdown, but would not require immediate suspension of movement of irradiated fuel assemblies. The Note to the ACTIONS, "LCO 3.0.3 is not applicable," ensures that the actions for immediate suspension of irradiated fuel assembly movement are not postponed due to entry into LCO 3.0.3.

TSTF-45 Rev. 2

not locked sealed, or otherwise secured and is

7

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

This SR verifies that each secondary containment manual isolation valve and blind flange that is required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the [secondary] containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those SCIVs in [secondary] containment that are capable of being mispositioned are in the correct position. 2

TSTF-45

This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position since these valves were verified to be in the correct position upon locking, sealing, or securing.

Since these SCIVs are readily accessible to personnel during normal operation and verification of their position is relatively easy, the 31 day Frequency was chosen to provide added assurance that the SCIVs are in the correct positions.

Two Notes have been added to this SR. The first Note applies to valves and blind flanges located in high radiation areas and allows them to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these SCIVs, once they have been verified to be in the proper position, is low.

A second Note has been included to clarify that SCIVs that are open under administrative controls are not required to meet the SR during the time the SCIVs are open.

Insert SR 3.6.4.2.1 5

SR 3.6.4.2.2

Verifying that the isolation time of each power operated ~~and each~~ automatic SCIV is within limits is required to demonstrate OPERABILITY. The isolation time test ensures that the SCIV 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~~ is [in accordance with the Inservice Testing Program or 92 days]. TSTF-46 2

6

(continued)

5 Insert SR 3.6.4.2.1

These controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

BASES

While this surveillance can be 1

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.2.3

Verifying that each automatic SCIV closes on a secondary containment isolation signal is required to prevent leakage of radioactive material from ~~[secondary]~~ containment following a DBA or other accidents. This SR ensures that each automatic SCIV will actuate to the isolation position on a ~~[secondary]~~ containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in ~~SR 3.3.6.2.6~~ LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation" 4 overlaps this SR to provide complete testing of the safety function. ~~The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were~~ performed with the reactor at power. 1 ~~Operating~~ operating experience has shown these components usually pass the Surveillance when performed at the ~~[18]~~ 24 2 month Frequency, which is based on the refueling cycle 1. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. 1 UFSAR, Section ~~[15.1.39]~~ 15.6.5.
2. 2 UFSAR, Section ~~[15.1.41]~~ 15.7.2.
3. ~~FSAR, Section [ ]~~ Technical Requirements Manual. 1

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.4.2 - SECONDARY CONTAINMENT ISOLATION VALVES (SCIVs)

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. Typographical/grammatical error corrected.

4. These changes have been made for consistency with similar phrases in other parts of the Bases and/or to be consistent with the Specification.

5. Editorial change made for enhanced clarity.

6. The words in SR 3.6.4.2.2, stating that the isolation times are in the IST Program have been deleted. The IST Program does not include the times for the SCIVs. They are located in the Technical Requirements Manual.

7. This statement has been deleted since it is incorrect. Automatic SCIVs that are deactivated and secured in the closed position are not OPERABLE; they are inoperable.

8. The discussion in the LCO section about closed valves is modified. This editorial preference is based on an incomplete and misleading discussion of the valves. This change does not modify the requirements or the interpretation of the requirements.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.3 Standby Gas Treatment (SGT) System

UFSAR, Section 3.1.9.1 1

BASES

BACKGROUND

The SGT System is required by 10 CFR 50, Appendix A, GDC 41, "Containment Atmosphere Cleanup" (Ref. 1). The function of the SGT System is to ensure that radioactive materials that leak from the primary containment into the [secondary] containment following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment. 2

The SGT System consists of two fully redundant subsystems, that are shared between Unit 1 and 2 each with its own set of ductwork, dampers, charcoal filter train, and controls. 1

Each charcoal filter train consists of (components listed in order of the direction of the air flow):

a. A demister ~~or moisture separator~~; 1

b. An electric heater;

c. A rough prefilter; 1

d. A high efficiency particulate air (HEPA) filter;

e. A charcoal adsorber; 1

f. A second HEPA after filter; and 1

g. A centrifugal fan.

The sizing of the SGT System equipment and components is based on the results of an infiltration analysis, as well as an exfiltration analysis of the [secondary] containment. Each SGT subsystem is capable of processing the secondary containment volume, which includes both Unit 1 and Unit 2. 1 The internal pressure of the ~~SGT System boundary region~~ secondary containment is maintained at a negative pressure of ≥ [0.25] inches water gauge when the SGT system is in operation, which represents the internal pressure required to ensure zero exfiltration of air from the building when exposed to a [10] 35 mph wind ~~blowing at an angle of [45] to the building~~. 2 1 2 4

The demister is provided to remove entrained water in the air, while the electric heater reduces the relative humidity

(continued)

BASES

BACKGROUND (continued)

of the airstream to less than ~~70%~~ (Ref. 2). The prefilter removes large particulate matter, while the HEPA filter removes fine particulate matter and protects the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides, and the final HEPA filter collects any carbon fines exhausted from the charcoal adsorber. 2

The SGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. Following initiation, ~~both charcoal filter train fans start. Upon verification that both subsystems are operating, the redundant subsystem is normally shut down.~~ Insert BKGD 1

APPLICABLE SAFETY ANALYSES

The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident and fuel handling accidents (Ref~~s~~. ~~2~~ 3, 4, and 5). For all events analyzed, the SGT System is shown to be automatically initiated to reduce, via filtration and adsorption, the radioactive material released to the environment.

The SGT System satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10CFR 50.36(c)(2)(ii). 1

LCO

Following a DBA, ~~a minimum~~ of one SGT subsystem is required to maintain the ~~[~~secondary~~]~~ containment at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for two OPERABLE subsystems ensures operation of at least one SGT subsystem in the event of a single active failure. OPERABILITY of a subsystem also requires the associated cooling air damper remain OPERABLE. 1 2

APPLICABILITY

In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, SGT System OPERABILITY is required during these MODES.

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SGT

(continued)

1 Insert BKGD

the pre-selected subsystem train inlet and outlet dampers will automatically open, the associated train's cooling air damper closes, and the associated fan starts and operates at a flow rate of 4000 cfm $\pm$ 10%. The Reactor Building suction damper for the subsystem on the unaffected reactor unit closes and the subsystem's associated cooling air damper remains open to provide decay heat removal. After secondary containment isolation, the SGT subsystem, under calm wind conditions, holds the building at an average negative pressure of 0.25 inches water gauge. A failure of the primary SGT subsystem to start within 25 seconds will initiate the automatic start and alignment of the standby SGT subsystem.

BASES

APPLICABILITY (continued)

System in OPERABLE status is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the [secondary] containment. 2

ACTIONS

A.1

With one SGT subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status in 7 days. In this condition, the remaining OPERABLE SGT subsystem is adequate 3 to perform the required radioactivity release control function. However, the overall system reliability is reduced because a single failure in the OPERABLE subsystem could result in the radioactivity release control function not being adequately performed. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant SGT System and the low probability of a DBA occurring during this period.

B.1 and B.2

If the SGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1, C.2.1, C.2.2, and C.2.3

During movement of irradiated fuel assemblies, in the [secondary] containment, during CORE ALTERATIONS, or during 2 OPDRVs, when Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE SGT subsystem should immediately be placed in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that could prevent automatic actuation have will 3

(continued)

BASES

ACTIONS

C.1, C.2.1, C.2.2, and C.2.3 (continued)

3

occurred, and that any other failure would be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that represent a potential for releasing radioactive material to the ~~secondary~~ containment, thus placing the plant in a condition that minimizes risk. If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies must immediately be suspended. Suspension of these activities must not preclude completion of movement of a component to a safe position. Also, if applicable, actions must immediately be initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended. 2

4

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3,

~~The Required Actions of Condition C have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations.~~ Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

Insert ACTION C 3

D.1

If both SGTS subsystems are inoperable in MODE 1, 2, or 3, the SGT system may not be capable of supporting the required radioactivity release control function. ~~Therefore, actions are required to enter LCO 3.0.3 immediately.~~

Insert ACTION D

Insert ACTION E

5

F

~~E~~.1, ~~E~~.2, and ~~E~~.3

When two SGT subsystems are inoperable, if applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in ~~secondary~~ containment must immediately be suspended. 2
Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action~~s~~ must immediately be initiated to suspend OPDRVs in order to minimize the probability of a vessel 3

(continued)

3 Insert ACTION C

Entering LCO 3.0.3 while in MODE 1, 2, or 3 would require the unit to be shutdown, but would not require immediate suspension of movement of irradiated fuel assemblies. The Note to the ACTIONS, "LCO 3.0.3 is not applicable," ensures that the actions for immediate suspension of irradiated fuel assembly movement are not postponed due to entry into LCO 3.0.3.

3 Insert ACTION D

Therefore, one SGT subsystem must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of supporting the required radioactivity release control function in MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring the SGT System) occurring during periods where the required radioactivity release control function may not be maintained is minimal.

5 Insert ACTION E

E.1 and E.2

If one SGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 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.

BASES

ACTIONS 5 F

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

draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

3

5 F

LCO 3.0.3 is not applicable while in MODE 4 or 5; However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3,

Required Action F.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. ~~Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.~~

Insert F.1, F.2 and F.3 — 3

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

(from the control room using the manual initiation switch) — 4

2 — Operating each SGT subsystem for ≥ [10] continuous hours ensures that [both] subsystems 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. Operation [with the heaters on (automatic heater cycling to maintain temperature)] for ≥ [10] continuous hours every 31 days eliminates moisture on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

SR 3.6.4.3.2

This SR verifies that the required SGT filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The SGT System filter tests are in accordance with Regulatory Guide 1.52 (Ref. 3). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP. — 6 — 1

(continued)

3 **Insert F.1, F.2, and F.3**

Entering LCO 3.0.3 while in MODE 1, 2, or 3 would require the unit to be shutdown, but would not require immediate suspension of movement of irradiated fuel assemblies. The Note to the ACTIONS, "LCO 3.0.3 is not applicable," ensures that the actions for immediate suspension of irradiated fuel assembly movement are not postponed due to entry into LCO 3.0.3.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.3.3

This SR verifies that each SGT subsystem starts on receipt of an actual or simulated initiation signal. While this Surveillance can be performed with the reactor at power, operating experience has shown that these components usually pass the Surveillance when performed at the ~~[18]~~ 24 month Frequency. The LOGIC SYSTEM FUNCTIONAL TEST in SR LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," ~~.6~~ overlaps this SR to provide complete testing of the safety function. Therefore, the Frequency was found to be acceptable from a reliability standpoint.

2

3

~~SR 3.6.4.3.4~~

~~This SR verifies that the filter cooler bypass damper can be opened and the fan started. This ensures that the ventilation mode of SGT System operation is available. While this Surveillance can be performed with the reactor at power, operating experience has shown that these components usually pass the Surveillance when performed at the [18] month Frequency, which is based on the refueling cycle. Therefore, the Frequency was found to be acceptable from a reliability standpoint.~~

5

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 41~~. UFSAR, Section 3.1.9.1
2. UFSAR, Section ~~[6.2.3]~~ 5.1.1
6. ~~3.~~ Regulatory Guide 1.52, Rev. ~~[2]~~.
3. UFSAR Section 15.6.2
4. UFSAR Section 15.6.5
5. UFSAR Section 15.7.2

1

2

1

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.6.4.3 - STANDBY GAS TREATMENT (SGT) SYSTEM

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. These changes have been made for consistency with similar phrases in other parts of the Bases and/or to be consistent with the Specification.

4. Editorial change made for enhanced clarity.

5. Changes have been made to reflect those changes made to the Specification.

ADMINISTRATIVE CHANGES
("A.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change involves reformatting, renumbering, and rewording the existing Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. 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 involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old 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 reduce a margin of safety because it has no impact on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

RELOCATED SPECIFICATIONS
("R.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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 that do not meet the criteria for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the Dresden 2 and 3 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 document which will be maintained pursuant to 10 CFR 50.59. In addition, the affected structures, systems, components or variables are addressed in existing surveillance procedures which are also controlled by 10 CFR 50.59 and subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. 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 involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements and adequate control of existing requirements 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 relocated requirements and surveillances for the affected structure, system, component or variable remain the same as the existing Technical Specifications. Since any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59, no reduction in a margin of safety will be permitted.

RELOCATED SPECIFICATIONS
("R.x" Labeled Comments/Discussions

3. (continued)

The existing requirement for NRC review and approval of revisions, in accordance with 10 CFR 50.92, to these details proposed for relocation does not have a specific margin of safety upon which to evaluate. However, since the proposed change is consistent with the BWR ISTS, NUREG-1433, Rev. 1, approved by the NRC Staff, revising the Technical Specifications to reflect the approved level of detail ensures no significant reduction in the margin of safety.

TECHNICAL CHANGES - MORE RESTRICTIVE
("M.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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 operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue 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 involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. 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 imposition of more restrictive requirements either has no impact on or increases the margin of plant safety. As provided in the discussion of the change, each change in this category is by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATING DETAILS TO TECHNICAL SPECIFICATION BASES, UFSAR, TRM, OR OTHER PLANT CONTROLLED DOCUMENTS
("LA.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change relocates certain details from the Technical Specifications to the Bases, UFSAR, TRM, or other plant controlled documents. The Bases, UFSAR, TRM, and other plant controlled documents containing the relocated information will be maintained in accordance with 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specification Bases are subject to the change control provisions in the Administrative Controls Chapter of the ITS. The UFSAR is subject to the change control provisions of 10 CFR 50.71(e), and the plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Since any changes to the Bases, UFSAR, TRM, or other plant controlled documents will be evaluated per the requirements of the Bases Control Program in Chapter 5.0 of the ITS or 10 CFR 50.59, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed. 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 involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any 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 details to be transposed from the Technical Specifications to the Bases, UFSAR, TRM, or other plant controlled

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATING DETAILS TO TECHNICAL SPECIFICATION BASES, UFSAR, TRM, OR OTHER PLANT CONTROLLED DOCUMENTS
("LA.x" Labeled Comments/Discussions)

3. (continued)

documents are the same as the existing Technical Specifications. Since any future changes to these details in the Bases, UFSAR, TRM, or other plant controlled documents will be evaluated per the requirements of 10 CFR 50.59, no reduction (significant or insignificant) in a margin of safety will be allowed. Based on 10 CFR 50.92, the existing requirement for NRC review and approval of revisions, to these details proposed for relocation, does not have a specific margin of safety upon which to evaluate. However, since the proposed change is consistent with the BWR ISTS, NUREG-1433, Rev. 1, approved by the NRC Staff, revising the Technical Specifications to reflect the approved level of detail ensures no significant reduction in the margin of safety.

"GENERIC" LESS RESTRICTIVE CHANGES:
EXTENDING SURVEILLANCE FREQUENCIES FROM 18 MONTHS TO 24 MONTHS FOR SURVEILLANCES OTHER THAN CHANNEL CALIBRATIONS
("LD.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change involves a change in the surveillance testing intervals from 18 months to 24 months. The proposed change does not physically impact the plant nor does it impact any design or functional requirements of the associated systems. That is, the proposed change does not degrade the performance or increase the challenges of any safety systems assumed to function in the accident analysis. The proposed change does not impact the Surveillance Requirements themselves nor the way in which the Surveillances are performed. Additionally, the proposed change does not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to the frequency of surveillance testing. The proposed change does not affect the availability of equipment or systems required to mitigate the consequences of an accident because of the availability of redundant systems or equipment and because other tests performed more frequently will identify potential equipment problems. Furthermore, an historical review of surveillance test results indicated that all failures identified were unique, non-repetitive, and not related to any time-based failure modes, and indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not increase 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 involves a change in the surveillance testing intervals from 18 months to 24 months. The proposed change does not introduce any failure mechanisms of a different type than those previously evaluated since there are no physical changes being made to the facility. In addition, the Surveillance Requirements themselves and the way Surveillances are performed will remain unchanged. Furthermore, an historical review of surveillance test results indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

"GENERIC" LESS RESTRICTIVE CHANGES:
EXTENDING SURVEILLANCE FREQUENCIES FROM 18 MONTHS TO 24 MONTHS FOR SURVEILLANCES OTHER THAN CHANNEL CALIBRATIONS
("LD.x" Labeled Comments/Discussions) (continued)

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

   Although the proposed change will result in an increase in the interval between surveillance tests, the impact on system availability is minimal based on other, more frequent testing or redundant systems or equipment, and there is no evidence of any failures that would impact the availability of the systems. Therefore, the assumptions in the licensing basis are not impacted, and the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.6.1.1 - PRIMARY CONTAINMENT

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change provides an allowed outage time to restore drywell-to-suppression chamber bypass leakage during operation in MODE 1, 2, or 3. With drywell-to-suppression chamber bypass leakage outside of limits in MODE 1, 2, or 3, the current Technical Specifications do not provide any actions. The proposed change provides 1 hour for restoration of this condition prior to commencing a required shutdown. Drywell-to-suppression chamber bypass leakage is an attribute of maintaining Primary Containment Integrity (in ITS terminology, primary containment OPERABILITY) and is not considered as an initiator of any previously analyzed accident. Therefore, this change does not significantly increase the probability of such accidents. The proposed change allows temporary operation when the drywell-to-suppression chamber bypass leakage requirement is not met. However, the consequences of an event that may occur during the proposed allowed outage time are not any different than during the current allowed outage time for other loss of primary containment integrity (OPERABILITY) situations. Therefore, this change 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?

   This change does not result in any changes to the equipment design or capabilities or to the operation of the plant. Further, the change impacts only the Required Action Completion Time for restoring drywell-to-suppression chamber bypass leakage and does not result in any change in the response of the equipment to an accident. Therefore, the 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 impacts only the Required Action Completion Time for restoration of drywell-to-suppression chamber bypass leakage during operation in MODE 1, 2, or 3. The methodology and limits of the accident analysis are not affected, nor is the primary containment response. This change results in an allowed outage time consistent with other ITS ACTIONS for similar primary containment degradations. Therefore, the change does not involve a significant reduction in the margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change deletes the requirement associated with CTS 4.7.K.5 to obtain an NRC review of the test schedule for subsequent tests if any leak rate test result is not within required limits. The subsequent test schedule has already been approved by the NRC. If two consecutive tests have failed, then the test must be performed every 9 months until two consecutive tests pass. The requirement to obtain NRC concurrence with the test schedule is not assumed to be an initiator of any analyzed event and does not impact assumptions of any design basis accident. Additionally, the concurrence is not required or assumed for the mitigation of any accident. Therefore, this change does not significantly increase 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 introduce a new mode of plant operation and does not involve physical modification to the plant. This change deletes a requirement to obtain NRC concurrence for a leak rate test schedule that is already approved by the NRC. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   This change does not involve a significant reduction in a margin of safety since the increased test schedule is already approved by the NRC and since experience has shown that the Surveillance normally meets its acceptance criterion when performed at the normal Frequency.

L.3 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change allows the drywell-to-suppression chamber bypass leakage to exceed to the current limit as long as leakage is less than or equal to the acceptable design $A/\sqrt{k}$ limit assumed in the safety analysis at times other than during the first unit startup following performance of bypass leakage testing. The change also deletes the detail of the initial differential pressure to perform the bypass leakage test from the Technical Specifications. Drywell-to-suppression chamber bypass leakage rate is an attribute of maintaining Primary Containment Integrity, and consequently, of Primary Containment OPERABILITY. Drywell-to-suppression chamber bypass leakage and testing methods are not considered as initiators of any previously analyzed accident, and therefore, the proposed change does not significantly increase the probability of such accidents. The proposed change allows continued operation with drywell-to-suppression chamber leakage that is greater than 2% of the acceptable design value, but less than or equal to the design leakage limit. The change also deletes the detail of the initial differential pressure to perform the bypass leakage test from the Technical Specifications. Therefore, this change 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?

   This change does not result in any changes to the equipment design or capabilities, or to the operation of the plant. Drywell-to-suppression chamber bypass leakage is assumed to be less than or equal to the design $A/\sqrt{k}$ limit under accident conditions. The change will not result in drywell-to-suppression chamber leakage in excess of this design limit, or result in any change in the response of the equipment to an accident. Therefore, the change does not increase the possibility of a new or different kind of accident from any previously analyzed.

L.3 CHANGE (continued)

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

This change impacts the acceptance criteria for drywell-to-suppression chamber bypass leakage rate at times other than during the first unit startup following performance of bypass leakage testing performed in accordance with proposed ITS SR 3.6.1.1.2. The change also deletes the detail of the initial differential pressure to perform the bypass leakage test from the Technical Specifications. The methodology and limits of the accident analyses are not affected, nor is the primary containment response. The change will result in an allowable drywell-to-suppression chamber bypass leakage that is less than or equal to the design $A/\sqrt{k}$ limit at all times. Therefore, the change does not involve a significant reduction in the margin of safety.

L.4 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change eliminates the requirement to perform testing of drywell-to-suppression chamber bypass leakage at an increased frequency following two consecutive leak test failures. If two consecutive tests result in a leakage that is greater than the specified limit, the current Technical Specifications require testing at an increased frequency until testing results in two consecutive, successful tests. The proposed change would dispense with this provision. Drywell-to-suppression chamber bypass leakage rate is an attribute of maintaining Primary Containment Integrity, and consequently of Primary Containment OPERABILITY. Drywell-to-suppression chamber bypass leakage is not considered as an initiator of any previously analyzed accident, and therefore, the proposed change does not significantly increase the probability of such accidents. The proposed change will not result in operation with leakage in excess of the acceptable design value. Therefore, this change 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?

   This change does not result in any changes to the equipment design or capabilities, or to the operation of the plant. Drywell-to-suppression chamber bypass leakage is assumed to be less than or equal to the design $A/\sqrt{k}$ limit under the accident conditions. The change will not result in drywell-to-suppression chamber leakage in excess of this design limit, or result in any change in the response of the equipment to an accident. Therefore, the change does not increase the possibility of a new or different kind of accident from any previously analyzed.

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

   This change only impacts the frequency of drywell-to-suppression chamber leakage testing in the event that the results of two consecutive tests are not within the specified limit. The effect of the change is considered minimal considering the history of consistently successful test results since plant startup, and provisions of the maintenance rule that would invoke remedial actions, such as increased test frequency, in the event

L.4 CHANGE

3. (continued)

of an adverse trend in bypass leakage rate. Additionally, the methodology and limits of the accident analyses are not affected by the change, nor is the primary containment response. Further, the change will not result in an allowable drywell-to-suppression chamber bypass leakage that is greater than the design $A/\sqrt{k}$ limit at any time. Therefore, the change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.6.1.2 - PRIMARY CONTAINMENT AIR LOCK

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change would allow the temporary opening of the remaining OPERABLE door for the purpose of making repairs to a primary containment air lock. This change does not affect the air lock design or function, and failure of an air lock is not identified as the initiator of any event. Therefore, this proposed change does not involve an increase in the probability of an accident previously evaluated. The change to allow the temporary opening of the one OPERABLE door for the purpose of making repairs results in a potential increase in consequences should an accident occur while it is open, but this increase is minimized through administrative controls and offset by the avoided potential consequences of an unnecessary transient during shutdown. The potential consequences resulting from the combination of: 1) the frequency of experiencing an inoperable air lock door such that temporarily opening the OPERABLE door is required for access to repair; 2) the brief period the OPERABLE door would be opened for access (typically on the order of one minute per entry/exit); and 3) the occurrence of an event of sufficient magnitude to cause an immediate containment pressure increase such that an air lock door could not be closed; are not considered to be significant. Additionally, providing the ability to eliminate the potential consequences of extended operation with only one OPERABLE door closed (not allowing repairs to be made to restore the second door to OPERABLE status); further minimizes the consequences. The allowance is proposed to have strict administrative control, which will provide assurance that any associated potential consequences are minimized. Finally, the allowed time for both doors to be open is not expected to exceed the currently allowed time for required action when containment integrity is determined to not be met. Therefore, these proposed changes do not involve 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 involve any design changes, plant modifications, or changes in plant operation. The primary containment air lock is designed and assumed to be used for entry and exit. Its operation does not interface with the reactor coolant or any controls which could impact the reactor coolant pressure boundary or its support systems. Further, brief periods of loss of containment integrity are acknowledged in

L.1 CHANGE

2. (continued)

the existing license; Specification 3.6.1.1 allows 1 hour to restore losses in containment integrity prior to requiring a plant shutdown.

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

The design, function, and OPERABILITY requirements for the primary containment air lock remains unchanged with this proposed revision. Containment leak rate limits are unaffected. The proposed change to allow the temporary opening of the one OPERABLE door for the purpose of repairing an inoperable door, is not considered to be a significant reduction in the margin of safety. The combination of: 1) the frequency of experiencing an inoperable air lock door such that containment entry is required for access to repair; 2) the brief period the OPERABLE door would be opened for access (typically on the order of one minute per entry/exit); and 3) the occurrence of an event of sufficient magnitude to cause an immediate containment pressure increase such that the air lock door could not be closed; are not representative of a significant reduction in the margin of safety. Additionally, providing the ability to eliminate any reduction in safety resulting from the extended operation with only one OPERABLE door closed (not allowing repairs to be made to restore the second door to OPERABLE status); minimizes any reduction in the margin of safety. The allowance is proposed to have strict administrative control, which will provide assurance that any associated safety reduction is further minimized. Therefore, these proposed changes do not involve a significant reduction in the margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change allows time to verify an OPERABLE air lock door is closed when a primary containment air lock is inoperable. This change does not affect the air lock design or function and one primary containment air lock door per airlock is sufficient to maintain primary containment integrity during a DBA. Additionally, the air lock doors are normally closed except for entry and exit and ITS 3.6.1.2 ACTIONS continue to provide adequate assurance that the primary containment function is maintained by requiring one OPERABLE air lock door to be closed within 1 hour which results in the same consequences as the primary containment being inoperable for 1 hour. Therefore, this change does not involve a significant increase in the probability or consequences of a previously evaluated accident.

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

   The 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. 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 allows time to verify an OPERABLE air lock door is closed when a primary containment air lock is inoperable. This change does not affect the air lock design or function and one primary containment air lock door per airlock is sufficient to maintain primary containment integrity during a DBA. Additionally, the air lock doors are normally closed except for entry and exit and ITS 3.6.1.2 ACTIONS require one air lock door to be closed within 1 hour. The proposed changes provides a time period for closing an OPERABLE air lock door that is consistent with respect to the time period provided for the condition of primary containment inoperable. In addition, the proposed change provides the benefit of potentially avoiding an unnecessary plant shutdown by providing time to close an OPERABLE air lock door. As such, no significant reduction in a margin of safety is involved with this change.

L.3 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change would allow verification that primary containment air locks are locked closed to be done by administrative means if the barrier is in a high radiation area or the access to them is limited due to inerting. Neither an open nor an inoperable airlock is considered as an initiator of any previously analyzed accident. Therefore, this change does not significantly increase the probability of such accidents. The proposed change provides actions with appropriate compensatory measures to maintain a level of safety equivalent to compliance with this and similar LCOs, such as containment OPERABILITY. These actions do not result in isolation barrier function different than assumed in any accident. Therefore, this change 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?

   This change does not result in any changes to the equipment design or capabilities, but does allow a different method of verification. However, since the change includes compensatory measures which maintain a level of safety equivalent to the capabilities of the equipment, the change does not create the possibility of a new or different kind of accident from any previously analyzed accident.

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

   This change allows the use of administrative means to provide compensatory actions in place of actual visual verification. The high radiation area and primary containment inerted access control and these additional administrative controls continue to provide adequate containment boundary should an accident occur. Therefore, the change does not involve a significant reduction in the margin of safety.

L.4 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change replaces the cumulative time limitation on a yearly basis for removal of personnel with an inoperable air lock door to a time period of 7 days for any single entry into the Condition. This change does not affect the air lock design or function, and failure of an air lock is not identified as the initiator of any event. Therefore, this proposed change does not involve an increase in the probability of an accident previously evaluated. The change to allow the temporary opening of the one OPERABLE door for purposes other than making repairs in excess of current limitations (1 hour per year) will not increase the consequences should an accident occur while it is open since the allowance is currently permitted. Since additional administrative controls are required, the actual time the air lock will be opened will be minimized thereby reducing the potential of operating outside the design basis. Additionally, providing the ability to eliminate the potential consequences of the transient of plant shutdown to follow (due to inability to perform preventive or corrective maintenance) further minimizes the consequences. Finally, the allowed time for both doors to be open is not expected to exceed the currently allowed time limit. Therefore, these proposed changes do not involve 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 involve any design changes, plant modifications, or changes in plant operation. The primary containment air lock is designed and assumed to be used for entry and exit. Its operation does not interface with the reactor coolant or any controls which could impact the reactor coolant pressure boundary or its support systems. Further, brief periods of loss of containment integrity are acknowledged in the current Technical Specifications prior to requiring a plant shutdown. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

L.4 CHANGE (continued)

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

The design, function, and OPERABILITY requirements for the primary containment air lock is unchanged with this proposed revision. Containment leak rate limits are unaffected. The proposed change replaces the cumulative time limitation on a yearly basis for removal of personnel with an inoperable air lock door to a time period of 7 days for any single entry into the Condition. This is not considered to be a significant reduction in the margin of safety. The combination of: 1) the frequency of experiencing an inoperable air lock door such that containment entry is required; 2) the brief period the OPERABLE door would be opened for access (typically on the order of one minute per entry/exit); and 3) the occurrence of an event of sufficient magnitude to cause an immediate containment pressure increase such that the air lock door could not be closed; are not representative of a significant reduction in the margin of safety. Additionally, providing the ability to eliminate any reduction in safety resulting from the transient of plant shutdown to follow (due to inability to perform preventive or corrective maintenance) minimizes any reduction in the margin of safety. The allowance is proposed to have strict administrative control which will provide assurance that any associated safety reduction is further minimized. Therefore, these proposed changes do not involve a significant reduction in the margin of safety.

L.5 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The primary containment air lock interlock is not assumed to be an initiator of any analyzed event. The role of the interlock is to ensure the primary containment boundary is maintained, thereby limiting consequences. Failure of the interlock during testing could result in a loss of primary containment OPERABILITY. Since the proposed change reduces the frequency of challenge to the interlock, the probability of a loss of primary containment OPERABILITY during the MODES when primary containment is required (LCO 3.6.1.1) is reduced. The OPERABILITY of the interlock has no effect on the consequences of an accident previously evaluated because no credit is taken for it in the mitigation of an accident. Therefore, this change does not involve a significant increase in the probability or consequences of a previously evaluated accident.

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

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure the interlocks remain OPERABLE when required. 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 reduces the challenges to primary containment OPERABILITY during MODES when primary containment is required to be OPERABLE. Further, proving the OPERABILITY of the air lock interlock at more frequent intervals serves no useful purpose since no enhancement to safety is gained by simply testing the interlock. From the standpoint of primary containment OPERABILITY and a reduction of unnecessary testing, the proposed change represents an enhancement to safety. As such, no significant reduction in a margin of safety is involved with this change.

L.6 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change deletes the requirement to have one air lock door "locked" closed at all times when an air lock is being used for entry and exit when the air lock mechanism is found to be inoperable. This change does not affect the air lock design or function, and failure of an air lock is not identified as the initiator of any event. Therefore, this proposed change does not involve an increase in the probability of an accident previously evaluated. In the proposed Specifications, with an air lock mechanism inoperable entry into and exit from primary containment is permissible only under the control of a dedicated individual. The duties of this individual are to perform the function of the interlock; to ensure both air lock doors are not opened simultaneously. That is, one door will be closed at all times. The requirement to have one door "locked" closed is not necessary. As long as one door is closed the containment integrity function will be maintained, and therefore, the requirement is not necessary during entry and exit into the containment. Locking an air lock door does not allow normal operation of the air lock. More time is required for locking therefore personnel will spend more time in the air lock instead of performing safety related activities. When entry and exit is no longer required, the proposed Specifications will continue to require at least one door to be "locked" closed. With the door locked, the dedicated individual is no longer required, and entry into the containment is prevented. The proposed requirements are considered adequate for ensuring primary containment integrity and at the same time control entry into the primary containment when the air lock mechanism is found to be inoperable. Therefore, the proposed change does not involve 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 involve any design changes, plant modifications, or change in plant operation. The primary containment air lock is designed and assumed to function when it is closed vice "locked" closed. Its operation does not interface with the reactor coolant or any controls which could impact the reactor coolant pressure boundary or its support systems. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

L.6 CHANGE (continued)

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

The design, function, and OPERABILITY requirements for the primary containment air lock remains unchanged with this proposed revision. Containment leak rate limits are unaffected. In the proposed Specifications, with an air lock interlock mechanism inoperable entry into and exit from primary containment is permissible only under the control of a dedicated individual. The duties of this individual are to perform the function of the interlock; to ensure both air lock doors are not opened simultaneously. That is, one door will be closed at all times. The requirement to have one door "locked" closed is not necessary. As long as one door is closed the containment integrity function will be maintained during entry and exit into the containment. Locking an air lock door does not allow normal operation of the air lock. More time is required for locking, therefore, personnel will spend more time in the air lock instead of performing safety related activities. When entry and exit is no longer required, the proposed Specifications will continue to require at least one door to be "locked" closed. With the door locked, the dedicated individual is no longer required and entry into the containment is prevented. The proposed requirements are considered adequate for ensuring primary containment integrity, and at the same time, control entry into the primary containment when the air lock interlock mechanism is found to be inoperable. The proposed requirements will ensure the function of the interlock is met. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.6.1.3 - PRIMARY CONTAINMENT ISOLATION VALVES

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change relaxes the allowed restoration times to isolate the affected penetration(s) if one valve is inoperable from 4 hours to 72 hours for PCIVs in penetrations with a closed system and only one PCIV. The proposed change does not increase the probability of an accident. The time allowed to isolate the penetration by use of de-activated automatic valve, blind flange, etc. is not assumed to be an initiator of any analyzed event. The PCIVs isolate to control leakage from the primary containment during accidents. Allowing the additional time to isolate the PCIVs will not significantly increase the consequences of an accident. The consequences will be the same for the proposed times as for the current time. The additional time, however, will allow more time to repair the inoperable PCIV and possibly avoid a shutdown. Shutting down the plant is a transient which puts thermal stress on components which could increase the chances of challenging safety systems. In addition, the closed system piping or water seal will ensure primary containment integrity is maintained. This change will not alter assumptions relative to the mitigation of an accident or transient event. Therefore, this change will 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?

   This change will not result in any changes to equipment design or capabilities or the operation of the plant. The proposed change will still require the PCIVs to be restored to OPERABLE status. Therefore, this change will 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 relaxes the allowed restoration time for isolating the affected penetration(s) if one valve is inoperable from 4 hours to 72 hours for PCIVs in penetrations with a closed system and only one PCIV. The margin of safety is not significantly reduced because the closed system piping or the water seal acts as a primary containment isolation barrier. Also, the time allowed to isolate penetrations is not assumed in any safety analysis and current safety analysis assumptions will be maintained. The added time also allows more time to isolate the PCIVs.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   Check valves that serve as containment isolation valves are not assumed to be initiators of any analyzed event. The role of these valves is to isolate containment during analyzed events, thereby limiting consequences. The change establishes compensatory measures using a check valve as an isolation barrier which are equivalent to those already included in Technical Specifications. The proposed actions will not allow continuous operation such that a single failure could allow a containment release through an unisolated path. Therefore, this proposed change will 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?

   This change does not result in any changes to equipment design or capabilities or the operation of the plant. The proposed change will still ensure the containment boundary is 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 check valves which would be used for this proposed compensatory measure are containment isolation valves and leak tested per 10 CFR 50, Appendix J. In addition, the proposed action established the check valve as an isolation barrier that cannot be adversely affected by a single active failure. As a result, any reduction in a margin of safety will be insignificant and offset by the benefit gained by reducing unnecessary plant shutdown transients when equivalent compensatory measures exist to ensure the containment boundary is maintained.

L.3 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change would allow additional time to isolate a primary containment penetration if two or more isolation devices are inoperable. Primary containment isolation is not considered as an initiator of any previously analyzed accident. Therefore, this change does not significantly increase the probability of such accidents. The proposed change allows additional temporary operation with less than the required isolation capability. However, the consequences of an event that may occur during the extended outage time would not be any different than during the currently allowed outage time for other loss of containment integrity situations. Therefore, this change 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?

   This change does not result in any changes to the equipment design or capabilities or to the operation of the plant. Further, since the change impacts only the Required Action Completion Time for the system and does not result in any change in the response of the equipment to an accident, the change does not create the possibility of a new or different kind of accident from any previously analyzed accident.

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

   This change impacts only the Required Action Completion Time for inoperable valves that provide primary containment isolation. The methodology and limits of the accident analysis are not affected, nor is the primary containment response. Therefore, the change does not involve a significant reduction in the margin of safety.

L.4 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change would allow an isolated primary containment penetration to be opened under administrative controls. Primary containment isolation is not considered as an initiator of any previously analyzed accident. Therefore, this change does not significantly increase the probability of such accidents. The proposed administrative controls provide an acceptable compensatory action to assure the penetration is isolated in the event of an accident. Therefore, the consequences of a previously analyzed event that may occur during the opening of the isolated line would not be significantly increased.

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

   This change provides an additional acceptable compensatory action following failure of other equipment. The current requirements are based on providing a single active failure proof boundary to compensate for the loss of one of the two active boundaries. The proposed change provides an alternative which essentially returns the system to its original configuration (i.e., configuration which can provide a single active failure proof boundary.) Therefore, this change does not create the possibility of a new or different kind of accident from any previously analyzed accident.

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

   The margin of safety considered in determining the required compensatory action is also based on providing the single active failure proof boundary. Since the proposed compensatory boundary essentially meets the original criteria and provides leakage characteristics essentially similar to currently approved compensatory boundaries, the change does not involve a significant reduction in the margin of safety.

L.5 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The requirement to verify primary containment isolation valve isolation times are within limits to verify the restoration of a primary containment isolation valve is not assumed in the initiation of any analyzed event. This requirement was specified in the Technical Specifications to ensure the OPERABILITY of a primary containment isolation valve was positively verified following repair, maintenance, or replacement. The proposed deletion of this explicit requirement is considered acceptable since SR 3.0.1 requires the appropriate SRs to be performed to demonstrate OPERABILITY after restoration of a component that cause the SR to be failed. In this case, SR 3.0.1 would require SR 3.6.1.3.5 (for PCIVs other than MSIVs) and SR 3.6.1.3.6 (for MSIVs), as applicable, to be performed, which require verification that isolation times of the affected primary containment isolation valves are within limits. As a result, the accident consequences are unaffected by this change. Therefore, this change will 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 possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant.

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

   The proposed deletion of the explicit requirement to verify primary containment isolation valve isolation times are within limits following repair, maintenance, or replacement is considered acceptable since SR 3.0.1 requires the appropriate SRs to be performed to demonstrate OPERABILITY after restoration of a component that cause the SR to be failed. In this case, SR 3.0.1 would require SR 3.6.1.3.5 (for PCIVs other than MSIVs) and SR 3.6.1.3.6 (for MSIVs), as applicable, to be performed, which require verification that isolation times of the affected primary containment isolation valves are within limits. As a result, the existing requirement to verify primary containment isolation valve isolation times are within limits following repair, maintenance, or replacement is maintained. Therefore, this change does not involve a significant reduction in a margin of safety.

L.6 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The phrase "actual or," in reference to the isolation test signal, has been added to the system functional test surveillance test description. This does not impose a requirement to create an "actual" signal, and does not eliminate any restriction on producing an "actual" signal. While creating an "actual" signal could increase the probability of an event, existing procedures and 10 CFR 50.59 control of revisions to them, dictate the acceptability of generating this signal. The proposed change does not affect the procedures governing plant operations and the acceptability of creating these signals; it simply would allow such a signal to be utilized in evaluating the acceptance criteria for the system functional test requirements. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated. Since the method of initiation will not affect the acceptance criteria of the system functional test, the change does not involve 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 introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Use of an actual signal instead of the existing requirement, which limits use to a test signal, will not affect the performance or acceptance criteria of the Surveillance. OPERABILITY is adequately demonstrated in either case since the system itself can not discriminate between "actual" or "test" signals. Therefore, the change does not involve a significant reduction in a margin of safety.

L.7 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change removes the requirement that the EFCVs must check flow and replaces it with a requirement to isolate to their isolation position. The EFCVs are designed to automatically go to the isolation position in the event of an instrument line break during normal reactor operation, or under accident conditions. The EFCVs are not credited to isolate in the instrument line break accident and are not the initiators of any accidents. Therefore, the change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

   The proposed change will not impact the method of testing the EFCVs. Accident analysis for the instrument line break assumes the line breaks at containment and that neither the EFCV nor the manual block valve are available to isolate the instrument line. (The accident is terminated by cooling down the plant and closing the manual valve after the plant is shutdown and depressurized.) Since the testing method is not being changed and no credit is taken for the EFCV to isolate on an instrument line break, the change does not create the possibility of a new or different kind of accident from any accident evaluated previously.

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

   EFCVs are installed in instrument lines to automatically act to check flow within the first few seconds of the instrument line break. The proposed surveillance will not change the method by which the valves are tested, since the requirement to verify the EFCVs isolate to their isolation position remains. Neither GDC 55, GDC 56, Regulatory Guide 1.11, nor the Quad Cities 1 and 2 design basis analysis require leakage measurements be performed for the EFCVs. None of the EFCVs are required to be leak checked to meet the 10 CFR 50 Appendix J requirements. The instrument lines are designed such that in the event of an instrument line break between containment and the EFCV, the leakage is reduced to the maximum extent practical consistent with other safety requirements. Accident analysis does not credit the EFCVs or the manual block valve for the instrument line break. Therefore, this change does not involve a significant reduction in a margin of safety.

L.8 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change would allow additional time to isolate an excess flow check valve penetration. Excess flow check valve isolation is not considered an initiator of any previously analyzed accident. Therefore, this change does not significantly increase the probability of such accidents. The proposed change allows additional temporary operation with less than the required isolation capability. However, the consequences of an event that may occur during the extended outage time would not be any different than during the currently allowed outage time. Therefore, this change 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?

   This change does not result in any changes to equipment design or capabilities, but does allow an extended period of operation with equipment not capable of performing its safety function. However, the leakage that may occur in the event of an additional single failure would be less than the previously analyzed leakage, thus, the additional time provided for isolation of the penetration does not impact the reactor coolant pressure boundary or its support systems. Therefore, this change does not create the possibility of a new or different kind of accident from any previously analyzed accident.

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

   The margin of safety considered in determining the allowed outage time is based on engineering judgement, and the probability of occurrence of an event requiring the unavailable capabilities. The proposed extension is based on the minimal impact of an excess flow valve being out of service, and the need to avoid an unnecessary plant transient caused by the forced shutdown. Therefore, the change does not involve a significant reduction in the margin of safety.

L.9 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change excludes the position verification of manual valves and blind flanges when the manual valves and blind flanges are locked, sealed or secured in the correct position. Primary containment isolation is not considered an initiator of any previously analyzed accident. Therefore, this change will not involve an increase in the probability of an accident previously evaluated. This change only alters the method of verifying the position of manual valves and blind flanges that are locked, sealed, or otherwise secured in the correct position. This allowance is acceptable since the probability of misalignment of a locked, sealed, or secured manual valve or blind flange, once it has been verified to be in the proper position, is small. The position verification of these manual valves and blind flanges is still maintained (the verification is performed upon locking, sealing, or securing the manual isolation device in position). As a result, the accident consequences are unaffected by this change. Therefore, this change will not involve an 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?

   This change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The position verification of these manual valves and blind flanges is still maintained (the verification is performed upon locking, sealing, or securing the manual isolation device in position). Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The proposed change excludes the position verification of manual valves and blind flanges when the manual valves and blind flanges are locked, sealed or secured in the correct position. This change only alters the method of verifying the position of manual valves and blind flanges that are locked, sealed, or otherwise secured in the correct position. This allowance is acceptable since the probability of misalignment of a locked, sealed, or secured manual valve or blind flange, once it has been verified to be in the proper position, is small. The position verification of these manual valves

L.9 CHANGE

3. (continued)

and blind flanges is still maintained (the verification is performed upon locking, sealing, or securing the manual isolation device in position). Eliminating the position verification of these manual valves and blind flanges in radiation areas increases safety to plant personnel and reduces exposure to plant personnel which is consistent with the As-Low-As-Reasonably-Achievable (ALARA) concept. Since the position verification of these manual valves and blind flanges is still maintained and the probability of misalignment of these manual valves and blind flanges is small due to the affected manual valves and blind flanges being locked, sealed, or secured in the correct position, this change does not involve a significant reduction in a margin of safety.

L.10 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change will allow the verification of closure of isolation devices such as valves and blind flanges located in high radiation areas (whether or not the isolation device is located inside the containment) or that are locked, sealed, or otherwise secured, to be performed by the use of administrative means. The entry into high radiation areas is restricted by plant procedures, therefore, any inadvertent opening of these devices is very low. If a procedure or maintenance is performed and these valves are opened, their closure would be required upon completion of the associated procedure or maintenance. Therefore, adequate measures are in place to ensure these valves remain closed. The Required Action or Surveillance may be verified by reviewing that no work was performed in the radiation area since it was closed or if work was performed in the area that closure was verified upon completion of the work if the valve was opened. Plant procedures control the operation of locked, sealed, or otherwise secured isolation devices; thus the potential for inadvertent misalignment of these devices after locking, sealing, or otherwise securing is low. In addition, the isolation devices were verified to be in the correct position prior to locking, sealing, or otherwise securing. This change does not cause a significant increase in the probability or consequences of any previously analyzed accident since administrative methods are in place to ensure the penetration is closed when required.

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

   This change does not result in any changes to the equipment design or capabilities or to the operation of the plant. Further, since the change impacts only the method of verification and does not result in any change in the response of the equipment to an accident, the change does not create the possibility of a new or different kind of accident from any previously analyzed accident.

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

   This change allows the use of administrative means to provide compensatory actions in place of actual visual verification. The high radiation area access control, locked valve controls, and these additional administrative controls continue to provide adequate containment should an accident occur. Therefore, this change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.6.1.4 - DRYWELL PRESSURE

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.6.1.5 - DRYWELL AIR TEMPERATURE

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.6.1.6 - LOW SET RELIEF VALVES

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The requirement to place the reactor mode switch in Shutdown in the event of an open relief valve is not assumed in the initiation of any analyzed event. The requirement of Action 1 of CTS 3.6.F was provided to ensure that, in the event of an open relief valve which could not be closed in a timely manner, the reactor mode switch would be placed in the Shutdown position in anticipation of exceeding a suppression pool average temperature of 110°F. However, Required Action D.1 of ITS 3.6.2.1 will still require that the reactor mode switch be immediately placed in Shutdown if the suppression pool average temperature is ≥110°F. As such, the Required Actions of ITS 3.6.2.1 are adequate to ensure that the reactor mode switch will immediately be placed in the Shutdown position if the suppression pool average temperature exceeds 110°F. As a result, accident consequences are unaffected by the deletion of the requirement to place the reactor mode switch in the Shutdown position if an open relief valve is unable to be closed. Therefore, this change will 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 possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant.

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

   This change deletes the requirement to place the reactor mode switch in the Shutdown position if an open relief valve is unable to be closed. This requirement of Action 1 of CTS 3.6.F was provided to ensure that, in the event of an open relief valve which could not be closed in a timely manner, the reactor mode switch would be placed in the Shutdown position in anticipation of exceeding a suppression pool average temperature of 110°F. However, Required Action D.1 of ITS 3.6.2.1 will still require that the reactor mode switch be immediately placed in Shutdown if the suppression pool average temperature is ≥110°F. As such, the Required Action of ITS 3.6.2.1 are adequate to ensure that the reactor mode switch will immediately be placed in the Shutdown position if the suppression pool average temperature exceeds 110°F. In

L.1 CHANGE

3. (continued)

addition, Emergency Operating Procedures and Special Operating Procedures address the appropriate actions to take in response to an open relief valve. As a result, continued assurance is provided that plant operation will be maintained with safety analysis assumptions. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.6.1.7 - REACTOR BUILDING-TO-SUPPRESSION CHAMBER VACUUM BREAKER

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change would allow 1 hour of operation with one or both vacuum breakers in both lines inoperable for opening. The vacuum breakers are not initiators of any previously analyzed accident. Therefore, the change does not significantly increase the frequency of such accidents. The change will not increase the consequences of an accident previously analyzed since continued operation is not allowed with both lines inoperable, thus the consequences are the same during the additional 1 hour as it is during the current shutdown times.

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

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

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

   The change is acceptable based on the small probability of an event requiring the vacuum breakers and the desire to minimize plant transients. This 1 hour Completion Time is also consistent with the allowed time for other containment inoperabilities (i.e., leakage). As such, any reduction in a margin of safety will be insignificant and offset by the benefit gained from providing some time to restore the vacuum breakers.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The reactor building-to-suppression chamber vacuum breaker position indication instrumentation is not assumed in the initiation of any analyzed event. The requirements for the vacuum breaker position indication instrumentation do not need to be explicitly stated in the Technical Specifications. To perform the verifications and tests required for the Surveillance Requirements of ITS 3.6.1.7, the capability to determine vacuum breaker position must be available. If the capability to determine vacuum breaker position is not available, these verifications and tests cannot be satisfied and the appropriate actions must be taken for inoperable vacuum breakers in accordance with the ACTIONS of ITS 3.6.1.7. As a result, accident consequences are unaffected by this change. Therefore, this change will 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 possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant.

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

   The proposed deletion of the vacuum breaker position indication instrumentation requirements from Technical Specifications does not impact any margin of safety. The requirements for the vacuum breaker position indication instrumentation do not need to be explicitly stated in the Technical Specifications. To perform the verifications and tests required for the Surveillance Requirements of ITS 3.6.1.7, the capability to determine vacuum breaker position must be available. If the capability to determine vacuum breaker position is not available, these verifications and tests cannot be satisfied and the appropriate actions must be taken for inoperable vacuum breakers in accordance with the ACTIONS of ITS 3.6.1.7. As a result, the capability to determine vacuum breaker position will be maintained to satisfy the associated SRs of ITS 3.6.1.7 without the need for explicit instrumentation requirements in the Technical Specifications. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.6.1.7 - REACTOR BUILDING-TO-SUPPRESSION CHAMBER VACUUM BREAKER

L.3 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change extends the frequency of vacuum breaker position verification from 7 days to every 14 days. The vacuum breakers are not assumed to be an initiator of any previously analyzed accident. Therefore, this change does not involve a significant increase in the probability of an accident previously evaluated. Since the vacuum breakers are normally closed and indication is provided in the control room of position, extending the Surveillance Frequency does not involve 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 introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, 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 change will not result in a reduction in a margin of safety since the vacuum breakers are still required to be closed. The change extends the frequency to verify the vacuum breakers are closed. Operational history has shown these vacuum breakers are normally closed. In addition, the vacuum breakers are single failure proof, in that, two vacuum breakers are available to ensure the penetration is closed, but only one vacuum breaker is needed to effect isolation.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.6.1.8 - SUPPRESSION CHAMBER-TO-DRYWELL VACUUM BREAKERS

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The suppression chamber-to-drywell vacuum breaker position indication instrumentation is not assumed in the initiation of any analyzed event. The requirements for the vacuum breaker position indication instrumentation do not need to be explicitly stated in the Technical Specifications. To perform the verifications and tests required for the Surveillance Requirements of ITS 3.6.1.8, the capability to determine vacuum breaker position must be available. If the capability to determine vacuum breaker position is not available, these verifications and tests cannot be satisfied and the appropriate actions must be taken for inoperable vacuum breakers in accordance with the ACTIONS of ITS 3.6.1.8. As a result, accident consequences are unaffected by this change. Therefore, this change will 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 possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant.

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

   The proposed deletion of the vacuum breaker position indication instrumentation requirements from Technical Specifications does not impact any margin of safety. The requirements for the vacuum breaker position indication instrumentation do not need to explicitly stated in the Technical Specifications. To perform the verifications and tests required for the Surveillance Requirements of ITS 3.6.1.8, the capability to determine vacuum breaker position must be available. If the capability to determine vacuum breaker position is not available, these verifications and tests cannot be satisfied and the appropriate actions must be taken for inoperable vacuum breakers in accordance with the ACTIONS of ITS 3.6.1.8. As a result, the capability to determine vacuum breaker position will be maintained to satisfy the associated SRs of ITS 3.6.1.8 without the need for explicit instrumentation requirements in the Technical Specifications. Therefore, this change does not involve a significant reduction in a margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change extends the frequency of vacuum breaker position verification from 7 days to every 14 days. The vacuum breakers are not assumed to be an initiator of any previously analyzed accident. Therefore, this change does not involve a significant increase in the probability of an accident previously evaluated. Since the vacuum breakers are normally closed and indication is provided in the control room of position, extending the Surveillance Frequency does not involve 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 introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, 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 change will not result in a reduction in a margin of safety since the vacuum breakers are still required to be closed. The change extends the frequency to verify the vacuum breakers are closed. Operational history has shown these vacuum breakers are normally closed. In addition, local position indication and redundant control room alarms are provided for each vacuum breaker to ensure that the vacuum breakers are maintained closed.

L.3 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change provides an exception allowing the vacuum breakers to be open when performing required Surveillances (the exception is to the Surveillance that would otherwise require the vacuum breakers to be closed at all times). The vacuum breakers are not assumed to be an initiator of any previously analyzed accident. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated. The Surveillance exception is made only for circumstances where the vacuum breaker is under the immediate control of an operator (manually opening to confirm Operability). As such, the vacuum breaker is expected to continue to perform its intended and assumed safety function, and therefore this change does not significantly increase 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 introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the 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 vacuum breakers are still required to be Operable. The exception is made only for circumstances where the vacuum breaker is under the immediate control of an operator (manually opening to confirm Operability). As such, the vacuum breaker is expected to continue to perform its intended and assumed safety function, and therefore this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.6.2.1 - SUPPRESSION POOL AVERAGE TEMPERATURE

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The change proposes to remove the explicit details of methods for reducing suppression pool temperature to within limits. The method used to reduce suppression pool temperature to within limits is not assumed in the initiation of any analyzed event. The proposed change does not affect the probability of an accident. Also, the consequences of an accident are not affected by this change since the Required Actions of Condition D of ITS 3.6.2.1 ensure the unit is placed in a non-applicable MODE if the suppression pool temperature is not reduced to within limits. With the unit in a non-applicable MODE, the requirements of ITS LCO 3.0.4 ensure that suppression pool temperature is reduced to within limits prior to entering an applicable MODE. In addition, methods for reducing suppression pool temperature to within limits are part of a coordinated response to an unplanned event governed by plant procedures. Since restoration of suppression pool temperature will still be required as part of the coordinated response to the event, consequences of previously analyzed accidents are not impacted by the removal of the explicit method for reducing suppression pool temperature to within limits. Therefore, this change 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 will not create the possibility of an accident. This change will not physically alter the plant (no new or different type of equipment will be installed). The change does not affect methods governing normal plant operation or the planned response to off-normal conditions. Therefore, this change will 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 proposes to remove the explicit details of methods for reducing suppression pool temperature to within limits. If the suppression pool temperature is not reduced to within limits, the Required Actions of Condition D of ITS 3.6.2.1 ensure the unit is placed in a non-applicable MODE. With the unit in a non-applicable MODE, the requirements of ITS LCO 3.0.4 ensure that suppression pool temperature is reduced to within limits prior to entering an applicable MODE. In addition, methods

L.1 CHANGE

3. (continued)

for reducing suppression pool temperature to within limits are part of a coordinated response to an unplanned event governed by plant procedures. The requirements of ITS 3.6.2.1 are considered to be adequate to ensure the suppression pool temperature is reduced to within required limits. Since restoration of suppression pool temperature will still be required by both Technical Specifications and as part of the coordinated response to the event, the margin of safety is not impacted by this change. Therefore, this change does not involve a significant reduction in a margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change would delete a surveillance frequency increase based on suppression pool temperature that is within the LCO limits. The suppression pool is not considered an initiator of any previously analyzed accident. Therefore, this change does not significantly increase the frequency of such accidents. The proposed change in surveillance frequency does not impact the ability of systems to reduce the temperature of the suppression pool or the suppression pool capabilities to respond to an accident. Therefore, this change 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?

   This change does not result in any changes to the equipment design or capabilities, but simply maintains the acceptable surveillance frequency as long as the LCO is being met. Therefore, the change does not create the possibility of a new or different kind of accident from any previously analyzed accident.

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

   The change removes an unnecessary surveillance frequency increase when conditions do not warrant such an increase. The frequency continues to increase when the LCO is not being met. Therefore, the change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.6.2.2 - SUPPRESSION POOL WATER LEVEL

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change would allow an additional hour to restore suppression pool level when it is found outside the limits. The suppression pool is not considered an initiator of any previously analyzed accident. Therefore, this change does not significantly increase the probability of such accidents. The proposed change would allow additional temporary operation with the required suppression pool level not met. However, since the change is in the allowed outage time, the consequences of an event that may occur during the extended outage time would not be any different than during the currently allowed outage time. Therefore, this change 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?

   This change does not result in any changes to the equipment design or capabilities, but does allow operation of the plant with equipment not capable of performing its safety function. However, loss of the pressure suppression function does not impact the reactor coolant pressure boundary or its support systems, and therefore, does not create the possibility of a new or different kind of accident from any previously analyzed accident.

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

   The change increases the allowed outage time by one hour. The margin of safety considered in determining the allowed outage time is based on engineering judgement and probability of occurrence of an event requiring the unavailable capabilities. An extension of one hour is based on the minimal impact to the margin of safety and allows appropriate actions to be taken without undo haste and potentially prevents a shutdown. Therefore, the change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.6.2.3 - RHR SUPPRESSION POOL COOLING

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change would allow an additional 8 hours to restore one loop when both are found to be inoperable. Suppression pool cooling is not considered an initiator of any previously analyzed accident. Therefore, this change does not significantly increase the frequency of such accidents. The proposed change would allow additional temporary operation with less than the required suppression pool cooling capability. However, since the only change is in the allowed outage time, the consequences of an event that may occur during the extended outage time would not be any different than during the currently allowed outage time. Therefore, this change 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?

   This change does not result in any changes to the equipment design or capabilities, but does allow operation of the plant with equipment not capable of performing its safety function. However, loss of the suppression pool cooling function does not impact the reactor coolant pressure boundary or its support systems, and therefore, does not create the possibility of a new or different kind of accident from any previously analyzed accident.

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

   The change increases the allowed outage time. The margin of safety considered in determining the allowed outage time is based on engineering judgement and probability of occurrence of an event requiring the unavailable capabilities. The proposed 8 hour extension is based on similar allowed outage times for the drywell spray system and the suppression pool spray system. Therefore, the change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.6.2.4 - RHR SUPPRESSION POOL SPRAY

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.6.2.5 - DRYWELL-TO-SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The drywell-suppression chamber differential pressure instrumentation is not assumed in the initiation of any analyzed event. The requirements for the drywell-suppression chamber differential pressure instrumentation do not need to be explicitly stated in the Technical Specifications. To perform the verifications required for the Surveillance Requirements of ITS 3.6.2.5, the drywell-suppression chamber differential pressure instrumentation must be OPERABLE. If the drywell-suppression chamber pressure instrumentation is inoperable, these verifications cannot be satisfied and the appropriate actions must be taken for drywell-suppression chamber differential pressure not within limits in accordance with the ACTIONS of ITS 3.6.2.5. As a result, accident consequences are unaffected by this change. Therefore, this change will 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 possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant.

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

   The proposed deletion of the drywell-suppression chamber differential pressure instrumentation requirements from Technical Specifications does not impact any margin of safety. The requirements for drywell-suppression chamber differential pressure instrumentation do not need to be explicitly stated in the Technical Specifications. To perform the verifications required for the Surveillance Requirement of ITS 3.6.2.5, the drywell-suppression chamber differential pressure instrumentation must be OPERABLE. If the drywell-suppression chamber differential pressure instrumentation is inoperable, these verifications cannot be satisfied and the appropriate actions must be taken for drywell-suppression chamber differential pressure not within limits in accordance with the ACTIONS of ITS 3.6.2.5. As a result, the OPERABILITY of the

L.1 CHANGE

3. (continued)

drywell-suppression chamber differential pressure instrumentation will be maintained to satisfy the associated SR of ITS 3.6.2.5 without the need for explicit instrumentation requirements in the Technical Specifications. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.6.3.1 - PRIMARY CONTAINMENT OXYGEN CONCENTRATION

There were no plant specific less restrictive changes identified for this Specification.

There were no plant specific less restrictive changes identified for this Specification.

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change would allow an isolated secondary containment penetration to be opened under administrative controls similar to most other primary containment penetrations. Secondary containment isolation is not considered as an initiator of any previously analyzed accident. Therefore, this change does not significantly increase the probability of such accidents. The proposed administrative controls provide an acceptable compensatory action to assure the penetration is isolated in the event of an accident. Therefore, the consequences of a previously analyzed event that may occur during the opening of the isolated line would not be significantly increased.

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

   This change provides an additional acceptable compensatory action following failure of other equipment. The current requirements are based on providing a single active failure proof boundary to compensate for the loss of one of the two active boundaries. The proposed change provides an alternative which essentially returns the system to its original configuration (i.e., configuration which can provide a single active failure proof boundary.) Therefore, this change does not create the possibility of a new or different kind of accident from any previously analyzed accident.

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

   The margin of safety considered in determining the required compensatory action is also based on providing the single active failure proof boundary. Since the proposed compensatory boundary essentially meets the original criteria and provides leakage characteristics essentially similar to currently approved compensatory boundaries, the change does not involve a significant reduction in the margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   This change would allow additional time to isolate a secondary containment penetration if both isolation devices are inoperable. Secondary containment isolation is not considered as an initiator of any previously analyzed accident. Therefore, this change does not significantly increase the probability of such accidents. The proposed change allows additional temporary operation with less than the required isolation capability. However, the consequences of an event that may occur during the extended outage time would not be any different than during the currently allowed outage time for other loss of secondary containment integrity situations. Therefore, this change 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?

   This change does not result in any changes to the equipment design or capabilities or to the operation of the plant. Further, since the change impacts only the Required Action Completion Time for the system and does not result in any change in the response of the equipment to an accident, the change does not create the possibility of a new or different kind of accident from any previously analyzed accident.

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

   This change impacts only the Required Action Completion Time for inoperable valves that provide secondary containment isolation. The methodology and limits of the accident analysis are not affected, and the secondary containment response in unaffected. Therefore, the change does not involve a significant reduction in the margin of safety.

L.3 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The requirement to verify secondary containment isolation valve isolation times are within limits to verify the restoration of a secondary containment isolation valve is not assumed in the initiation of any analyzed event. This requirement was specified in the Technical Specifications to ensure the OPERABILITY of a secondary containment isolation valve was positively verified following repair, maintenance, or replacement. The proposed deletion of this explicit requirement is considered acceptable since SR 3.0.1 requires the appropriate SRs to be performed to demonstrate OPERABILITY after restoration of a component that caused the SR to be failed. In this case, SR 3.0.1 would require SR 3.6.4.2.2 to be performed, which requires verification that isolation times of the affected secondary containment isolation valves are within limits. As a result, the accident consequences are unaffected by this change. Therefore, this change will 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 possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant.

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

   The proposed deletion of the explicit requirement to verify secondary containment isolation valve isolation times are within limits following repair, maintenance, or replacement is considered acceptable since SR 3.0.1 requires the appropriate SRs to be performed to demonstrate OPERABILITY after restoration of a component that cause the SR to be failed. In this case, SR 3.0.1 would require SR 3.6.4.2.2 to be performed, which requires verification that isolation times of the affected secondary containment isolation valves are within limits. As a result, the existing requirement to verify secondary containment isolation valve isolation times are within limits following repair, maintenance, or replacement is maintained. Therefore, this change does not involve a significant reduction in a margin of safety.

L.4 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The phrase "actual or," in reference to the isolation test signal, has been added to the system functional test surveillance test description. This does not impose a requirement to create an "actual" signal, nor does it eliminate any restriction on producing an "actual" signal. While creating an "actual" signal could increase the probability of an event, existing procedures and 10 CFR 50.59 control of revisions to them, dictate the acceptability of generating this signal. The proposed change does not affect the procedures governing plant operations and the acceptability of creating these signals; it simply would allow such a signal to be utilized in evaluating the acceptance criteria for the system functional test requirements. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated. Since the method of initiation will not affect the acceptance criteria of the system functional test, the change does not involve 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 introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Use of an actual signal instead of the existing requirement, which limits use to a test signal, will not affect the performance or acceptance criteria of the Surveillance. OPERABILITY is adequately demonstrated in either case since the system itself can not discriminate between "actual" or "test" signals. Therefore, the change does not involve a significant reduction in a margin of safety.

L.5 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change excludes the position verification of manual valves and blind flanges when the manual valves and blind flanges are locked, sealed or secured in the correct position. Secondary containment isolation is not considered an initiator of any previously analyzed accident. Therefore, this change will not involve an increase in the probability of an accident previously evaluated. This change only alters the method of verifying the position of the manual valves and blind flanges that are locked, sealed or otherwise secured in the correct position. This allowance is acceptable since the probability of misalignment of a locked, sealed or secured manual valve or blind flange, once it has been verified to be in the proper position, is small. The position verification of these manual valves and blind flanges is still maintained (the verification is performed upon locking, sealing, or securing the manual isolation device in position). As a result, the accident consequences are unaffected by this change. Therefore, this change will not involve an increase in the consequence 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 change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The position verification of these manual valves and blind flanges is still maintained (the verification is performed upon locking, sealing, or securing the manual isolation device in position). Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The proposed change excludes the position verification of manual valves and blind flanges when the manual valves and blind flanges are locked, sealed or secured in the correct position. This change only alters the method of verifying the position of manual valves and blind flanges that are locked, sealed, or otherwise secured in the correct position. This allowance is acceptable since the probability of misalignment of a locked, sealed, or secured manual valve or blind flange, once it has been verified to be in the proper position, is small. The position verification of these manual valves and blind flanges is still maintained (the verification is performed upon locking, sealing

L.5 CHANGE

3. (continued)

or securing the manual isolation device in position). Eliminating the position verification of these manual valves and blind flanges in radiation areas increases safety to plant personnel and reduces exposure to plant personnel which is consistent with the As-Low-As-Reasonably-Achievable (ALARA) concept. Since the position verification of these valves and blind flanges is still maintained and the probability of misalignment of these manual valve and blind flanges is small due to the affected manual valves and blind flanges being locked, sealed, or secured in the correct position, this change does not involve a significant reduction in a margin of safety.

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not involve a significant hazards consideration. The following is provided in support of this conclusion.

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

   An alternative is proposed to suspending operations if a standby gas treatment subsystem cannot be returned to OPERABLE status that would allow continued movement of irradiated fuel assemblies, core alterations, or operations with the potential for draining the reactor vessel. The alternative is to place the OPERABLE Standby Gas Treatment (SGT) subsystem in operation and continue to conduct operations (e.g., OPDRVs). Operation of the SGT System is not considered as an initiator of a previously analyzed accident. Therefore, the operation does not significantly increase the probability of an accident previously identified. Since one subsystem is sufficient to mitigate the consequences of previously evaluated accidents, the consequences of any previously evaluated accidents are not significantly increased.

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

   This change provides for continued performance of previously evaluated operations. Since these operations have been previously considered, their continued performance does not create the possibility of a new or different kind of accident from any previously analyzed accident.

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

   The margin of safety considered in performance of these operations is maintained by starting and running the system that would be required to initiate should an accident occur. Operation of the system significantly reduces the risk that the system may not perform its intended function initiate when required. Therefore, the change does not involve a significant reduction in the margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not involve a significant hazards consideration. The following is provided in support of this conclusion.

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

   The phrase "actual or," in reference to the initiation test signal, has been added to the system functional test surveillance test description. This does not impose a requirement to create an "actual" signal, nor does it eliminate any restriction on producing an "actual" signal. Creating an "actual" signal could increase the probability of an event, existing procedures and 10 CFR 50.59 control of revisions to them, dictate the acceptability of generating this signal. The proposed change does not affect the procedures governing plant operations and the acceptability of creating these signals; it simply would allow such a signal to be utilized in evaluating the acceptance criteria for the system functional test requirements. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated. Since the method of initiation will not affect the acceptance criteria of the system functional test, the change does not involve 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 possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant.

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

   Use of an actual signal instead of the existing requirement, which limits use to a test signal, will not affect the performance or acceptance criteria of the Surveillance test. OPERABILITY is adequately demonstrated in either case since the system itself can not discriminate between "actual" or "test" signals. Therefore, the change does not involve a significant reduction in a margin of safety.

ENVIRONMENTAL ASSESSMENT
ITS: SECTION 3.6 - CONTAINMENT SYSTEMS

In accordance with the criteria set forth in 10 CFR 50.21, ComEd has evaluated this proposed Technical Specification change for identification of licensing and regulatory actions requiring environmental assessment, determined it meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or which changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

1. The amendment involves no significant hazards consideration.

   As demonstrated in the No Significant Hazards Consideration, this proposed amendment does not involve any significant hazards consideration.

2. There is no significant change in the type or significant increase in the amounts of any effluents that may be released offsite.

   The proposed change will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no change in the types or significant increase in the amounts of any effluents released offsite resulting from this change.

3. There is no significant increase in individual or cumulative occupational radiation exposure.

   The proposed change will not result in changes in the operation or configuration of the facility which impact radiation exposure. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.

Therefore, based upon the above evaluation, ComEd has concluded that no irreversible consequences exist with the proposed change.