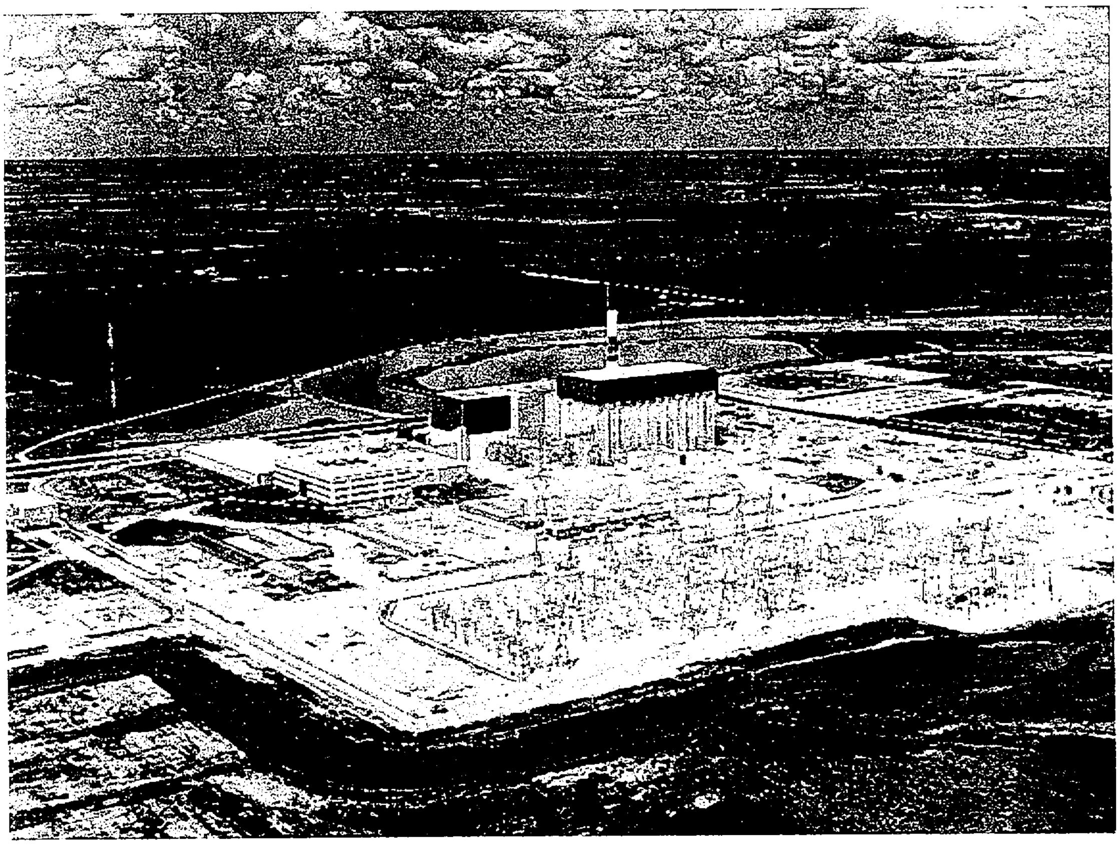
Improved Technical Specifications



LaSalle County Station

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

ComEd

<CTS>

3.6 CONTAINMENT SYSTEMS

3.6.1.1 Primary Containment

<LCO 3.6.1.1>
<LCO 3.6.2.1.b>
LCO 3.6.1.1 Primary containment shall be OPERABLE.

<Appl 3.6.1.1>
<Appl 3.6.2.1>
APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.1.1 Act> <3.6.2.1 Act>	A. Primary containment inoperable.	A.1 Restore primary containment to OPERABLE status.	1 hour
<3.6.1.1 Act> <Doc L.1> <4.6.2.1.d>	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

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
<4.6.1.1.d>	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~~	1
<4.6.1.1.e>	SR 3.6.1.1.2 Verify primary containment structural integrity in accordance with the ~~Primary Containment Tendon Surveillance Program.~~ Inservice Inspection Program for Post Tensioning Tendons	In accordance with the ~~Primary Containment Tendon Surveillance Program~~	2
	SR 3.6.1.1.3 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 is ≤10% of the drywell-to-suppression chamber bypass leakage limit.	24 months	3

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, 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 proposed TSTF-52.

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

3. The drywell-to-suppression chamber bypass leakage Surveillance Requirement (ITS SR 3.6.1.1.3) has been based on CTS 4.6.2.1.d and BWR/6 ISTS (NUREG-1434) SR 3.6.5.1.1. In addition, the Frequency of ISTS SR 3.6.5.1.1, (ITS SR 3.6.1.1.3) has been changed to reflect the change in the LaSalle 1 and 2 refuel cycle.

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3.6 CONTAINMENT SYSTEMS

3.6.1.2 Primary Containment Air Locks 1

The 2

<LCO 3.6.1.3> LCO 3.6.1.2 ~~[Two]~~ primary containment air locks shall be OPERABLE.

<Appl 3.6.1.3> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

1

---NOTES---

<DOC L.1> 1. Entry and exit is permissible to perform repairs of the ~~affected~~ air lock components.

~~2. Separate Condition entry is allowed for each air lock.~~ 1

<DOC A.3> 2 1 ~~3~~. 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.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.1.3 Act a 3.6.1.3 Act a.1 3.6.1.3 Act a.2> 1 A. ~~One or more primary containment air locks with~~ 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 ~~same~~ air lock are inoperable and Condition C is entered. 1 2. Entry and exit is permissible for 7 days under administrative controls ~~[if both air locks are inoperable]~~. 2	
		(continued)

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.1.3 Act a 3.6.1.3 Act a.1 3.6.1.3 Act a.2> A. (continued)	A.1 Verify the OPERABLE door is closed ~~in the affected air lock~~. [1] AND A.2 Lock the OPERABLE door closed ~~in the affected air lock~~. AND A.3 --------NOTE--------- Air lock doors in high radiation areas [3] or areas with limited access due to inerting may be verified locked closed by administrative means. ------------------------ Verify the OPERABLE door is locked closed ~~in the affected air lock~~. [1]	1 hour 24 hours Once per 31 days
<DOC L.5> B. [1] ~~One or more primary containment air locks with~~ ~~p~~rimary containment air lock interlock mechanism inoperable.	------------NOTES------------ 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the ~~same~~ air lock are inoperable and Condition C is entered. [1] 2. Entry into and exit from [4] primary containment is permissible under the control of a dedicated individual. ------------------------------	
		(continued)

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ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
<DOC L.5> B. (continued)	B.1	Verify an OPERABLE door is closed ~~in the affected air lock~~. 1	1 hour
	AND		
	B.2	Lock an OPERABLE door closed ~~in the affected air lock~~. 1	24 hours
	AND		
	B.3	--------NOTE--------- Air lock doors in high radiation areas 3 or areas with limited access due to inerting may be verified locked closed by administrative means. ---------------------	
		Verify an OPERABLE door is locked closed ~~in the affected air lock~~. 1	Once per 31 days
<3.6.1.3 Act b> C. 1 ~~One or more~~ primary containment air lock~~s~~ 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
	AND		
	C.2	Verify a door is closed ~~in the affected air lock~~. 1	1 hour
	AND		
			(continued)

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ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.1.3 Act b>	C. (continued)	C.3 Restore air lock to OPERABLE status.	24 hours
<3.6.1.3 Act a.3 3.6.1.3 Act b>	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

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
<4.6.1.3.a>	SR 3.6.1.2.1	-----NOTES----- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria ~~of~~ applicable to SR 3.6.1.1.1 ~~in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.~~ 5 ----- Perform required primary containment air lock leakage rate testing in accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions.~~ the Primary Containment Leakage Rate Testing Program. 5 ~~The acceptance criteria for air lock testing are: a. Overall air lock leakage rate is ≤ [2 scfh] when tested at ≥ P_a. b. For each door, leakage rate is ≤ [2 scfh] when the gap between the door seals is pressurized to ≥ [1.0 P_a].~~	~~-----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 5
6	~~[SR 3.6.1.2.2~~	~~Verify primary containment air lock seal air flask pressure is ≥ [90] psig.~~	~~7 days]~~

(continued)

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SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
<4.6.1.3.b> SR 3.6.1.2.~~4~~ 2 6	~~NOTE~~ ~~Only required to be performed upon entry or exit through the primary containment air lock.~~ TSTF-17 Verify only one door in the primary containment air lock can be opened at a time.	~~184 days~~ 24 months
~~SR 3.6.1.2.4~~	~~Verify, from an initial pressure of [90] psig, the primary containment air lock seal pneumatic system pressure does not decay at a rate equivalent to > [2] psig for a period of [48] hours.~~	~~[18] months~~ 6

1. The LaSalle 1 and 2 design only includes one primary containment air lock, similar to the BWR/4 design. Therefore, numerous changes have been made to reflect the one air lock design. The changes are consistent with the BWR/4 ISTS, (NUREG-1433, Rev. 1), except where discussed separately.

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

3. Changes have been made to allow the verification of closure of the air lock doors by administrative means when the primary containment is inerted to reflect the BWR-5 design. This change is consistent with plants with inerted containments as reflected in the BWR/4 ISTS, NUREG-1433, Rev 1.

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

5. The Primary Containment Leakage Rate Testing Program is included in CTS 6.2.F.7 and in ITS 5.5.13. 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 consistent with the Current Licensing Basis and with proposed TSTF-52.

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

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3.6 CONTAINMENT SYSTEMS

3.6.1.3 Primary Containment Isolation Valves (PCIVs)

<LCO 3.6.3>
<LCO 3.4.7>
<LCO 3.6.1.8>

LCO 3.6.1.3 Each PCIV shall be OPERABLE.

<Appl 3.6.3>
<Appl 3.4.7>
<Appl 3.6.1.8>

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-----

<3.6.3 Act fnote *>

1. Penetration flow paths ~~[except for [ ] inch purge valve penetration flow paths]~~ may be unisolated intermittently under administrative controls. 1

<Doc A.2>

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 PCIVs.

<3.6.3 Act b.1.b>
<Doc A.3>

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

<3.6.3 Act a>
<3.6.3 Act a.1>
<3.4.7 Act 1>
<4.6.1.1.c>
<3.6.1.8 Act>

CONDITION	REQUIRED ACTION	COMPLETION TIME
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 or secondary containment bypass]~~ due to 4 leakage not within limit].	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. AND	4 hours except for main steam line AND 8 hours for main steam line

(continued)

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ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.3 Acta 3.6.3Act a.1> <3.4.7 Act 1> <4.6.1.1.a> <4.6.1.1.a fnote**> <3.6.1.8 Act> A. (continued) TSTF-269	A.2 -----NOTE(S)----- 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, ~~drywell, and steam tunnel~~ 5
		AND
	if primary containment was de-inerted while in MODE 4 5	Prior to entering MODE 2 or 3 from ~~MODE 4, if not performed~~ within the previous 92 days, for isolation devices inside primary containment, ~~drywell, or steam tunnel~~ 5

(continued)

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ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<DOC L.3>	B. ---------NOTE--------- Only applicable to penetration flow paths with two or more PCIVs. [3] ------------------------- One or more penetration flow paths with two PCIVs inoperable ~~except for purge valve~~ due to [4] leakage not within limit.	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
<3.6.3 Act a, 3.6.3 Act a.1, 3.6.3 Act b, 3.6.3 Act b.1> <4.6.1.1.a> <3.6.1.8 Act>	C. ---------NOTE--------- Only applicable to penetration flow paths with only one PCIV. ------------------------- One or more penetration flow paths with one PCIV inoperable except due to leakage not within limit [4].	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 TSTF-269 (5) ---------NOTE--------- 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 administrative means. ------------------------- Verify the affected penetration flow path is isolated.	~~4~~ hours except for excess flow check valves (EFCVs) and penetrations with a closed system [1] AND [6] 72 hours for EFCVs and penetrations with a closed system Once per 31 days
<DOC L.13>	D. ~~Secondary containment bypass~~ [1] leakage rate not within limit. One or more flow paths with MSIV leakage rate or hydrostatically tested line [4]	D.1 Restore leakage rate to within limit.	~~(8)~~ 8 hours for MSIV leakage [8] AND 72 hours for hydrostatically tested line leakage on a closed system 4 hours for hydrostatically tested line leakage not on a closed system AND [8]

(continued)

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	
	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	
	E.3 Perform SR 3.6.1.3.6 for the resilient seal purge valves closed to comply with Required Action E.1.	Once per [92] days

TSTF-269 changes Not shown

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(continued)

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.3 Act a.2> <3.6.3 Act b.2> <3.4.7 Act 2> <3.6.1.8 Act> 1 ~~F.~~ E. Required Action and associated Completion Time of Condition A, B, C, ~~D,~~ or ~~E~~ D not met in MODE 1, 2, or 3.	~~F~~E.1 Be in MODE 3.	12 hours
	AND 1	
	~~F~~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 the [primary or secondary containment]~~	~~G.1 --------NOTE-------- LCO 3.0.3 is not applicable. ------------------ Suspend movement of irradiated fuel assemblies in [primary and 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~~
<Doc M.1> 1 ~~I.~~ F. Required Action and associated Completion Time of Condition A, B, C, ~~D,~~ or ~~E~~ D not met for PCIV(s) required to be OPERABLE during 9 MODE 4 or 5 ~~or during operations with a potential for draining the reactor vessel (OPDRVs)~~. 10	~~I~~F.1 Initiate action to suspend ~~OPDRVs~~ operations with a potential for draining the reactor vessel.	Immediately
	OR 1	
	~~I~~F.2 Initiate action to restore valve(s) to OPERABLE status. 11	Immediately

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
~~SR 3.6.1.3.1~~	~~-----NOTE----- Only required to be met in MODES 1, 2, and 3. ----- Verify each [ ] 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~~ 1
<DOC M.3> <DOC L.12> SR 3.6.1.3.2 1 1	-----NOTES----- 2 ~~1. Only required to be met in MODES 1, 2, and 3.~~ 2 2. 2 Not required to be met when the [20] 8 inch and 26 9 inch primary containment purge valves are open for inerting, de-inerting, 5 pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open, provided the drywell [purge supply and exhaust] lines are isolated purge valves and suppression chamber purge valves are not open simultaneously 9. ----- Verify each [20] 8 inch and 26 9 inch primary containment purge valve is closed. 9	31 days

(continued)

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SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
<3.6.3 fnote **> <4.6.1.1.a>	SR 3.6.1.3.~~3~~ 2 $\boxed{1}$ -----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, ~~drywell, and steam tunnel~~ $\boxed{5}$ and not locked, sealed, or otherwise secured $\boxed{\text{TSTF-45}}$ and is required to be closed during accident conditions is closed.	31 days
<3.6.3 fnote **> <4.6.1.1.a> <4.6.1.1.a fnote **>	SR 3.6.1.3.~~4~~ 3 $\boxed{1}$ -----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 inside primary containment, ~~drywell, or steam tunnel~~ $\boxed{5}$ and is required to be closed during accident conditions is closed. $\boxed{\text{TSTF-45}}$ and not locked, sealed, or otherwise secured	Prior to entering MODE 2 or 3 from MODE 4, if primary containment was de-inerted while in MODE 4 $\boxed{5}$ if not performed within the previous 92 days

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<4.6.3.5.a>	SR 3.6.1.3.4 Verify continuity of the traversing incore probe (TIP) shear isolation valve explosive charge.	31 days	$\boxed{6}$

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SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
<4.6.3.3>	SR 3.6.1.3.5	TSTF-46 Verify the isolation time of each power operated ~~and each~~ automatic PCIV~~s~~, except MSIVs, is within limits. 9	In accordance with the Inservice Testing Program ~~or 92 days~~ 9
	~~SR 3.6.1.3.6~~	~~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.~~ 1	~~184 days AND Once within 92 days after opening the valve~~
<LCO 3.4.7> <4.4.7>	SR 3.6.1.3.~~7~~ 6 1	Verify the isolation time of each MSIV is ≥ [3] seconds and ≤ [5] seconds. 9	In accordance with the Inservice Testing Program ~~or 18 months~~ 9
<4.6.3.2>	SR 3.6.1.3.~~8~~ 7 1	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	~~[18]~~ 24 months 9

Insert SR 3.6.1.3.8 and SR 3.6.1.3.9 6

(continued)

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Insert SR 3.6.1.3.8

SR 3.6.1.3.8 <4.6.3.4>	Verify each EFCV actuates to the isolation position on an actual or simulated instrument line break signal.	24 months

Insert SR 3.6.1.3.9

SR 3.6.1.3.9 <4.6.3.5.b>	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.9	-----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
<4.6.3.6.a> SR 3.6.1.3.10	Verify leakage rate through all four main steam lines is ≤ [100] 400 scfh when tested at ≥ [11.5] 25.0 psig. any one main steam line is ≤ 100 scfh and through	~~-----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

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
<4.6.3.6.6> SR 3.6.1.3.11	~~NOTES~~ ~~1. Only required to be met in MODES 1, 2, and 3.~~ 2 ~~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.~~ 14 15 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 ≥ 1.1 P_a.~~ within limits.	~~NOTE~~ ~~SR 3.0.2 is not applicable~~ 13 In accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~ the Primary Containment Leakage Rate Testing Program
~~SR 3.6.1.3.12~~	~~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]%.~~ 1	~~[18] months~~

1. This bracketed requirement has been deleted because it is not applicable to LaSalle 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 LaSalle 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 and ISTS SR 3.6.1.3.11 Note 1 have been deleted for the same reason. The following Notes have been renumbered, if applicable, due to these Notes deletion.

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 LaSalle 1 and 2 have more than two PCIVs. This was required by the NRC for some penetrations whose outside PCIV was not close enough to the primary containment. 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. (It is noted that the BWR/6 ISTS markup provided in TSTF-207, Rev. 3, inadvertently left out the words "or more" in Condition B. The BWR/4 ISTS markup included these words in Condition B.)

4. The words inside the brackets have been modified to reflect the different types of leakage categories. Since there is more than one, the generic word "leakage" has been used in ISTS 3.6.1.3 Conditions A, B, and C. 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 certain PCIVs 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 LaSalle 1 and 2 ITS, ACTION A is taken if the PCIV is inoperable for reasons other than leakage; ACTION D is required if the SRs for individual valve leakage limits are not met. Currently (in the ISTS), Condition A would only exempt purge valve leakage and secondary containment bypass leakage requirements and Condition C does not exempt any leakage requirements. If a MSIV or a hydrostatically tested valve was not meeting the leakage limits, Condition A or C, as applicable, would be entered and Required Action A.1 or

4. (continued)

C.1 would be required. These Required Actions allow 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 a MSIV or hydrostatically tested valve leakage is not within limits, Condition D should be entered. The Required Action for this Condition would require the leakage to be restored within limit in 4 hours, 8 hours, or 72 hours, as applicable, consistent with the time provided in Required Actions A.1 and C.1 to isolate the 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 Actions A.1 and C.1, which allows isolation using the leaking valve. Condition B has also been modified to exclude leakage. This Condition is appropriate if the valve is in the incorrect position or will not close. As discussed above, the Required Action for Condition B would also allow the penetration to be isolated using the leaking valve if the bracketed phrase were not deleted. This change is also consistent with TSTF-207, Rev. 3, except when plant specific differences apply or consistency errors were noted.

5. The LaSalle 1 and 2 design includes the drywell as part of the primary containment and the primary containment is inerted while operating, similar to the BWR/4 design. Therefore, changes have been made to the requirements which check proper position of isolation devices, similar to the BWR/4 ISTS (NUREG-1433).

6. The LaSalle 1 and 2 design also includes EFCVs and TIPs, similar to the BWR/4 design. Therefore, ITS 3.6.1.3 Required Action C.1 Completion Times have been modified to be consistent with the BWR/4 ISTS (NUREG-1433) and approved TSTF-30, Rev. 3. The change also provides a 72 hour Completion Time for EFCVs consistent with TSTF-323. ITS SR 3.6.1.3.4, SR 3.6.1.3.8, and SR 3.6.1.3.9 have also been added, consistent with the BWR/4 ISTS. The following requirements have been renumbered, where applicable, to reflect the additions.

7. Not used.

8. The time provided in ISTS ACTION D to restore MSIV leakage and hydrostatically tested line leakage on a closed system to within limits has been changed. The Required Action for this condition would require the leakage to be restored within limit in 4 hours for hydrostatically tested line leakage not on a closed system (no change), 8 hours for MSIV leakage, and 72 hours for hydrostatically tested line leakage on a closed system. The new 8 hour Completion Time for MSIV leakage is consistent with the time provided in Required Action A.1 to isolate the main steam line penetrations. The 72 hour Completion Time for hydrostatically tested line leakage on a closed system

8. (continued)

is deemed appropriate based in part on the approved generic change TSTF-30, Rev. 1, which provides a 72 hour Completion Time for single valve penetrations in a closed system. Some of the hydrostatically tested lines are on a closed system, while the others are water sealed and remain that way after the accident. This water sealed design was reviewed and approved by the NRC, as documented in the original LaSalle 1 and 2 SER and its supplements. This change is also consistent with TSTF-207, Rev. 3, except where plant specific differences apply.

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

10. The words in ISTS 3.6.1.3 Condition I (ITS 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 LaSalle 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.

11. 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.

12. The current leakage rate limit for the MSIVs is on a per line basis as well as on a total leakage rate limit through all four main steam lines. ITS SR 3.6.1.3.10 reflects the current licensing basis.

13. The Primary Containment Leakage Rate Testing Program has been added to 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.

14. The Appendix J testing requirements and associated acceptance criteria, or exemptions to applying leakage to that acceptance criteria, is adequately addressed in proposed SR 3.6.1.1.1, The deleted Note (ISTS SR 3.6.1.3.11 Note 2) serves no purpose. Additionally, the ITS 3.6.1.3 ACTIONS Note 4 ("Enter applicable Conditions...results in exceeding overall containment leakage rate acceptance criteria") provides appropriate and sufficient control to direct the proper ACTIONS should excessive leakage be discovered. In addition, these Notes were approved to be deleted from NUREG-1434, Rev. 1 per change package BWR-14, C.3, but apparently were not deleted. The BWR/4 ISTS (NUREG-1433) did delete the Note (NUREG-1433, SR 3.6.1.3.14).

15. The leakage limit and test pressure for ISTS SR 3.6.1.3.11 (ITS SR 3.6.1.3.11) have been deleted from the Technical Specifications consistent with the current licensing basis. This is also consistent with TSTF-52, Rev. 2.

<CTS>

Drywell and Suppression Chamber 1

3.6 CONTAINMENT SYSTEMS

3.6.1.4 Primary Containment Pressure

<LCO 3.6.1.6> LCO 3.6.1.4 Primary containment [to secondary containment differential] pressure shall be ≥ -0.1 psid and ≤ 1.0 psid.

-0.5 3 +0.75 9 2

<Appl 3.6.1.6> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.1.6 Act>	A. Primary containment [to secondary containment differential] pressure not within limits. Drywell or suppression chamber 1	A.1 Restore primary containment [to secondary containment differential] pressure to within limits. drywell and suppression chamber	1 hour
<3.6.1.6 Act>	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
<4.6.1.6>	SR 3.6.1.4.1 Verify primary containment [to secondary containment differential] pressure is within limits. drywell and suppression chamber 1	12 hours

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

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

<CTS> $\boxed{1}$ Drywell

3.6 CONTAINMENT SYSTEMS

3.6.1.5 Primary Containment Air Temperature

<LCO 3.6.1.7> LCO 3.6.1.5 Primary containment average air temperature shall be ≤ [95] °F.

135 $\boxed{2}$

<Appl 3.6.1.7> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

	CONDITION	REQUIRED ACTION		COMPLETION TIME
<3.6.1.7 Act>	A. Drywell Primary containment average air temperature not within limit.	A.1	$\boxed{1}$ drywell Restore primary containment average air temperature to within limit.	8 hours
<3.6.1.7 Act>	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
<4.6.1.7>	SR 3.6.1.5.1	Verify drywell $\boxed{1}$ primary containment average air temperature is within limit.	24 hours

<Insert ITS 3.6.1.6 (BWR/4 ISTS 3.6.1.8)> $\boxed{3}$

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.6.1.5 - DRYWELL AIR TEMPERATURE

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

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

3. A new Specification has been added, ITS 3.6.1.6. This Specification is from the BWR/4 ISTS (NUREG-1433 ISTS 3.6.1.8), since the LaSalle 1 and 2 design is similar to the BWR/4 design with regard to the vacuum breakers. Therefore, the BWR/4 Specification is used and any deviations from the BWR/4 ISTS are discussed in the Justification for Deviations for ITS: 3.6.1.6.

Insert BWR/4 ISTS 3.6.1.8 * 1

<CTS>

3.6 CONTAINMENT SYSTEMS

3.6.1.8 6 1 Suppression Chamber-to-Drywell Vacuum Breakers

<LCO 3.6.4> LCO 3.6.1.8 ~~[Nine]~~ Each suppression chamber-to-drywell vacuum breaker~~s~~ shall be OPERABLE ~~for opening~~. 2

~~AND~~

~~[Twelve] suppression chamber-to-drywell vacuum breakers shall be closed, except when performing their intended function.~~ 3

<Appl 3.6.4> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

	CONDITION	REQUIRED ACTION		COMPLETION TIME
<3.6.4 Act a>	A. One ~~required~~ 3 suppression chamber-to-drywell vacuum breaker inoperable for opening.	A.1	Restore ~~one~~ the 3 vacuum breaker to OPERABLE status.	72 hours
<3.6.4 Act b>	B. One suppression chamber-to-drywell vacuum breaker not closed.	B.1	Close ~~the open vacuum breaker~~ both manual isolation valves in the affected line.	~~2~~ 4 hours 4
		AND		
		B.2	Restore the vacuum breaker to OPERABLE status.	72 hours 4
<3.6.4 Act a> <3.6.4 Act b>	C. Required Action and associated Completion Time of Condition A or B 4 not met.	C.1	Be in MODE 3.	12 hours
		AND		
		C.2	Be in MODE 4.	36 hours
<Doc M.1>	D. Two or more suppression chamber-to-drywell vacuum breakers inoperable.	D.1	Enter LCO 3.0.3	Immediately 4

* This BWR/4 Specification Insert was used because it best represented the LaSalle Unit 2 design

Insert Page 3.6-20a

Insert BWR/4 ISTS 3.6.1.8 [1]
(continued)

<CTS>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<4.6.4.1.a> <4.6.4.1.a fnote*>	SR 3.6.1.8.1 [1] 6 ---NOTE--- 5 [3] 1. Not required to be met for vacuum breakers that are open during Surveillances. [3] 2. Not required to be met for vacuum breakers open when performing their intended function. --- 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~~ [5]

(continued)

Insert Page 3.6-20b

Insert BWR/4 ISTS 3.6.1.8 [1]
(continued)

[1] 6

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
<4.6.4.1.b.1>	SR 3.6.1.6.2 [1] Perform a functional test of each required [3] vacuum breaker.	31 days 92 [6] AND Within 12 hours after any discharge of steam to the suppression chamber from the S/RVs safety/relief valves [7] ~~AND Within 12 hours following an operation that causes any of the vacuum breakers to open~~ [8]
<4.6.4.1.b.3.a)>	SR 3.6.1.6.3 [1] Verify the opening setpoint of each required [3] vacuum breaker is ≤ 0.5 psid. [2]	18 24 months

Insert Page 3.6-20c

1. A new Specification has been added, ITS 3.6.1.6. This Specification is from the BWR/4 ITS (NUREG-1433 ISTS 3.6.1.8), since the LaSalle 1 and 2 design is similar to the BWR/4 design with regard to the vacuum breakers. Therefore, the BWR/4 LCO is used and any deviations from the BWR/4 ISTS are discussed below.

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

3. The design to which the BWR/4 ISTS 3.6.1.8 was written required all the suppression chamber-to-drywell vacuum breakers to be closed, but did not require all the suppression chamber-to-drywell vacuum breakers to be Operable. Therefore, two separate LCO statements were provided. The LaSalle 1 and 2 current licensing basis requires all the suppression chamber-to-drywell vacuum breakers to be Operable and closed. To more closely match the LaSalle 1 and 2 design, only a single LCO statement is needed. This LCO statement requires each suppression chamber-to-drywell vacuum breaker to be Operable, with the requirement to be closed as part of the Operability requirement. This is consistent with the BWR/4 ISTS 3.6.1.7 LCO statement, which requires each reactor building-to-suppression chamber vacuum breaker to be Operable (in this LCO statement, closed is part of Operable). In addition, since the second part of the deleted LCO statement ("except when performing their intended function") is still needed to be included in the Specification, a second Note has been included in SR 3.6.1.6.1 providing this allowance. The location of the Note is also consistent with the BWR/4 ISTS 3.6.1.7.1. Also, ISTS Condition A and SRs 3.6.1.8.2 and 3.6.1.8.3 have been modified to delete the word "required" and Required Action A.1 has been changed from "one" to "the."

4. The LaSalle 1 and 2 design for the suppression chamber-to-drywell vacuum breakers has one vacuum breaker per line, with manual isolation valves on both sides of the vacuum breaker. With a vacuum breaker open, the isolation capability of the line can be maintained by closing both manual isolation valves. Therefore, the ISTS 3.6.1.8 ACTIONS have been modified to reflect this design and the current licensing basis. The changes are as follows:

    a. Required Action B.1 has been modified to require closing both manual isolation valves in the affected line in lieu of closing the open vacuum breaker. This action essentially maintains the isolation capability of the vacuum breaker line. The time to perform this action has also been changed to 4 hours.

    b. New Required Action B.2 has been added to require restoration of the inoperable vacuum breaker. This is needed since the modified Required Action B.1 does not restore the vacuum breaker to OPERABLE status.

4. (continued)

   c. New ACTION D has been added to ensure LCO 3.0.3 is entered if more than one vacuum breaker is inoperable. The current analysis can only support one inoperable vacuum breaker.

5. The second Frequency to NUREG-1433 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 3.6.1.6.1. 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 position indication in the control room, and are continuously 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 current LaSalle 1 and 2 Technical Specifications.

6. The Frequency for ISTS SR 3.6.1.8.2, the vacuum breaker functional test, has been extended from 31 days to 92 days in ITS SR 3.6.1.6.2. These vacuum breakers are not located in a harsh environment; they are located in the secondary containment, similar to many other PCIVs that are tested on a 92 day Frequency (per the IST Program). This ISTS Frequency was based on the fact that these types of vacuum breakers are in a harsh environment (as stated in the Bases for ISTS SR 3.6.1.8.2). For vacuum breakers that are not in a harsh environment, a 92 day functional test Frequency is used. This is shown in ISTS SR 3.6.1.7.2, which has a 92 day Frequency for the functional test. The Bases of ISTS 3.6.1.7 describes that these vacuum breakers are not located in a harsh environment. Therefore, since the LaSalle 1 and 2 design locates the suppression chamber-to-drywell vacuum breakers outside the primary containment (in the secondary containment), a 92 day Frequency is justified.

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

8. The third Frequency to NUREG-1433 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."

8. (continued)

Since the vacuum breakers are designed to operate and 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. Therefore, this Frequency, which is not in the LaSalle 1 and 2 CTS, has not been added to the LaSalle 1 and 2 ITS.

[1]

3.6 CONTAINMENT SYSTEMS

3.6.1.6 Low-Low Set (LLS) Valves

LCO 3.6.1.6 The LLS function of [six] safety/relief valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

1

SURVEILLANCE REQUIREMENTS

	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.	[18] months [on a STAGGERED TEST BASIS for each valve solenoid]
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.	18 months

JUSTIFICATION FOR DEVIATION FROM NUREG-1434, REVISION 1
ISTS: 3.6.1.6 - LOW-LOW SET (LLS) VALVES

1. The LaSalle 1 and 2 design basis analyses do not assume the Low-Low Set function of the Safety/Relief Valves, nor are they required to be operable in the CTS. As stated in UFSAR section 7.3.1.2.2.10, the Low-Low Set Function was added as a product improvement to improve the primary containment design margins, but is not required to accommodate the primary containment loads as defined in NUREG-0487. Therefore, this Specification has not been adopted in the LaSalle 1 and 2 ITS.

3.6 CONTAINMENT SYSTEMS

3.6.1.7 Residual Heat Removal (RHR) Containment Spray System

LCO 3.6.1.7 Two RHR containment spray subsystems shall be OPERABLE.

1

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.1.7.1	-------------------NOTE------------------- RHR containment spray subsystems may be considered OPERABLE during alignment and operation for decay heat removal when below [the RHR cut in permissive pressure in MODE 3] if capable of being manually realigned and not otherwise inoperable. -- Verify each RHR containment 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.	31 days
SR 3.6.1.7.2	Verify each RHR pump develops a flow rate of ≥ [5650] gpm on recirculation flow through the associated heat exchanger to the suppression pool.	[In accordance with the Inservice Testing Program or 92 days]
SR 3.6.1.7.3	Verify each RHR containment spray subsystem automatic valve in the flow path actuates to its correct position on an actual or simulated automatic initiation signal.	[18] months
SR 3.6.1.7.4	Verify each spray nozzle is unobstructed.	[At first refueling <u>AND</u>] 10 years

1

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ISTS: 3.6.1.7 - RHR CONTAINMENT SPRAY SYSTEM

1. The LaSalle 1 and 2 CTS does not include requirements for drywell spray, since it is not credited for mitigating any design basis accidents. Therefore, it has not been included in the ITS.

1

3.6 CONTAINMENT SYSTEMS

3.6.1.8 Penetration Valve Leakage Control System (PVLCS)

LCO 3.6.1.8 [Two] PVLCS subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PVLCS subsystem inoperable.	A.1 Restore PVLCS subsystems to OPERABLE status.	30 days
B. [Two] PVLCS subsystems inoperable.	B.1 Restore one PVLCS subsystem to OPERABLE status.	7 days
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 Verify air pressure in each subsystem is ≥ [101] psig.	24 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.8.2	Perform a system functional test of each PVLCS subsystem.	[18] months

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ISTS: 3.6.1.8 - PENETRATION VALVE LEAKAGE CONTROL SYSTEM (PVLCS)

1. The LaSalle 1 and 2 design does not include a Penetration Valve Leakage Control System. Therefore, this Specification has been deleted.

1

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
	<u>AND</u>		
	C.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.9.1	Operate each MSIV LCS blower ≥ [15] minutes.	31 days

(continued)

1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
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. The LaSalle 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

<LCO 3.6.2.1 LCO 3.6.2.1.a.2 LCO 3.6.2.1.a.2.a> LCO 3.6.2.1 Suppression pool average temperature shall be: TSTF-206 Not shown

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;~~ 1

1 a. ~~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~~ [with THERMAL POWER > 1% RTP] TSTF-206 2 1; and

1 6 b. ~~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 2 TSTF-206

<Appl 3.6.2.1> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.2.1 Act b> <4.6.2.1.b.2> A. Suppression pool average temperature > [95] 105 2 °F but ≤ [110]°F. AND 2 [THERMAL POWER > 1% RTP] TSTF-206 ~~[Any OPERABLE IRM channel > [25/40] divisions of full scale on Range 7.]~~ ~~AND~~ ~~Not performing testing that adds heat to the suppression pool.~~ 1	Insert A.1 from next page 1 A.~~1~~ 2 1 Verify suppression pool average temperature ~~is~~ 3 ≤ [110]°F. 2 AND A.~~2~~ 3 1 Restore suppression pool average temperature to ≤ [95]°F. 105 2	Once per hour 24 hours

(continued)

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.2.1 Act b> B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER ~~[until all OPERABLE IRM channels are ≤ [25/40] divisions of full scale on Range 7.]~~ TSTF-206 2 [to ≤1% RTP]	12 hours 2
~~C. Suppression pool average temperature > [105]°F. AND Any OPERABLE IRM channel > [25/40] divisions of full scale on Range 7. AND Performing testing that adds heat to the suppression pool.~~ TSTF-206 not shown 1	C.1 A Suspend all testing that adds heat to the suppression pool. AND	Immediately Move to previous Page as Insert A.1
<3.6.2.1 Act b.1> <4.6.2.1.b.3> D C 1 Suppression pool average temperature > [110]°F but ≤ [120]°F. 2	1 C D.1 Place the reactor mode switch in the shutdown position.	Immediately
	AND	
	D.2 Verify suppression pool average temperature is 3 ≤ [120]°F. 2	Once per 30 minutes
	AND	
	D.3 Be in MODE 4.	36 hours

(continued)

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
<LCO 3.6.2.1.a.2.b)> <3.6.2.1 Act 6.2> ~~E~~ D [1] Suppression pool average temperature > [120] °F. [2]	~~E~~ D.1 [1]	Depressurize the reactor vessel to < [200] psig. [2]	12 hours
	AND		
	~~E~~ D.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<4.6.2.1.b> <4.6.2.1.b.1> 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

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.6.2.1 - SUPPRESSION POOL AVERAGE TEMPERATURE

1. The LaSalle CTS does not contain the lower 95°F limit. The CTS limit is always 105°F, whether or not testing that adds heat to the suppression pool is being performed. Therefore, the corresponding limitations as specified in ISTS LCO 3.6.2.1.a and b, including ACTIONS A and C, have been modified or deleted, as necessary, to correspond with the current licensing basis. In addition, TSTF-206 changes that affect the deleted requirements have not been added.

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

3. These additional words have been deleted for consistency. These words do not appear in the BWR/4 ISTS (NUREG-1433 Rev. 1). These words were approved to be deleted from NUREG-1434, Rev. 1 per change package BWR-6, C.4, but apparently were not deleted.

<CTS>

3.6 CONTAINMENT SYSTEMS

3.6.2.2 Suppression Pool Water Level

<LCO 3.6.2.1.a> <LCO 3.5.3.a> LCO 3.6.2.2 Suppression pool water level shall be ≥ ~~18 ft~~ − 4.5 inches and ≤ ~~18 ft 9.75~~ +3 inches. 1

<Appl 3.6.2.1> <Appl 3.5.3.a> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.2.1 Act a> <3.5.3 Act a> A. Suppression pool water level not within limits.	A.1 Restore suppression pool water level to within limits.	2 hours
<3.6.2.1 Act a> <3.5.3 Act a> 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
<4.6.2.1.a> <4.5.3.1.a.1> SR 3.6.2.2.1 Verify suppression pool water level is within limits.	24 hours

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, 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

LCO 3.6.2.3 LCO 3.6.2.3 Two RHR suppression pool cooling subsystems shall be OPERABLE.

Appl 3.6.2.3 APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
3.6.2.3 Act a	A. One RHR suppression pool cooling subsystem inoperable.	A.1 Restore RHR suppression pool cooling subsystem to OPERABLE status.	7 days
<3.6.2.3 Act a> <3.6.2.3 Act b>	~~B.~~ C. Required Action and associated Completion Time ~~of Condition A~~ ~~or B~~ 6 not met. OR	~~B.~~C.1 Be in MODE 3. AND ~~B.~~C.2 Be in MODE 4.	12 hours 36 hours
<3.6.2.3 Act b>	B. Two RHR suppression pool cooling subsystems inoperable.	B.1 Restore one RHR suppression pool cooling subsystem to OPERABLE Status.	8 hours

TSTF-230

<CTS>

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
<4.6.2.3.a>	SR 3.6.2.3.1	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. 2	31 days 1
<4.6.2.3.b>	SR 3.6.2.3.2	Verify each required 3 RHR pump develops a flow rate ≥ ~~[7450]~~ 7200 4 gpm through the associated heat exchanger while operating in the suppression pool cooling mode.	[In accordance with the Inservice Testing Program ~~or 92 days~~] 4

<INSERT ITS 3.6.2.4 (BWR/4 ISTS 3.6.2.4)> 5

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.6.2.3 - RHR SUPPRESSION POOL COOLING

1. The LaSalle 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.

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

3. The LaSalle 1 and 2 design only uses two of the three RHR pumps in the suppression pool cooling mode. 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. A new Specification has been added, ITS 3.6.2.4. This Specification is from the BWR/4 ISTS (NUREG-1433 ISTS 3.6.2.4), since the LaSalle 1 and 2 design is similar to the BWR/4 design with regard to RHR suppression pool spray. Therefore, the BWR/4 LCO is used and any deviations from the BWR/4 ISTS are discussed in the Justification for Deviations for ITS: 3.6.2.4.

6. 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.

Insert BWR/4 ISTS 3.6.2.4* $\boxed{1}$

<CTS>

3.6 CONTAINMENT SYSTEMS

3.6.2.4 Residual Heat Removal (RHR) Suppression Pool Spray

<LCO 3.6.2.2> LCO 3.6.2.4 Two RHR suppression pool spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.2.2 Act a> A. One RHR suppression pool spray subsystem inoperable.	A.1 Restore RHR suppression pool spray subsystem to OPERABLE status.	7 days
<3.6.2.2 Act b> B. Two RHR suppression pool spray subsystems inoperable.	B.1 Restore one RHR suppression pool spray subsystem to OPERABLE status.	8 hours
<3.6.2.2 Act a> <3.6.2.2 Act b> 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

$\boxed{1}$ * This BWR/4 Specification Insert was used because it best represented the LaSalle 1 and 2 design

Insert Page 3.6-34a

Insert BWR/4 ISTS 3.6.2.4 1
(continued)

<CTS>

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
<4.6.2.2.a>	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. 2 3	31 days
<4.6.2.2.b>	SR 3.6.2.4.2	Verify each required 5 RHR pump develops a flow rate ≥ ~~400~~ 450 4 gpm ~~through the heat exchanger while~~ operating in the suppression pool spray mode. through the spray sparger 6	In accordance with the Inservice Testing Program ~~or 92 days~~ 4

Insert Page 3.6-34b

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.6.2.4 - RHR SUPPRESSION POOL SPRAY

1. A new Specification has been added, ITS 3.6.2.4. This Specification is from the BWR/4 ISTS (NUREG-1433 ISTS 3.6.2.4), since the LaSalle 1 and 2 design is similar to the BWR/4 design with regard to RHR suppression pool spray. Therefore, the BWR/4 LCO is used and any deviations from the BWR/4 ISTS are discussed below.

2. The LaSalle 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).

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

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

5. The LaSalle 1 and 2 design only uses two of the three RHR pumps in the suppression pool spray mode. Therefore, ISTS SR 3.6.2.4.2 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.

6. The LaSalle 1 and 2 accident analysis does not credit the cooling effect of the RHR heat exchangers during the suppression pool spray mode. Therefore, this requirement has been deleted. Clarification of required flow through the spray sparger has been added, consistent with the LaSalle 1 and 2 current licensing basis.

1

3.6 CONTAINMENT SYSTEMS

3.6.2.4 Suppression Pool Makeup (SPMU) System

LCO 3.6.2.4 Two SPMU subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Upper containment pool water level not within limit.	A.1	Restore upper containment pool water level to within limit.	4 hours
B. Upper containment pool water temperature not within limit.	B.1	Restore upper containment pool water temperature to within limit.	24 hours
C. One SPMU subsystem inoperable for reasons other than Condition A or B.	C.1	Restore SPMU subsystem to OPERABLE status.	7 days
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	12 hours
	AND		
	D.2	Be in MODE 4.	36 hours

1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.2.4.1	Verify upper containment pool water level is ≥ [23 ft 3 inches] above the pool bottom.	24 hours
SR 3.6.2.4.2	Verify upper containment pool water temperature is ≤ [125]°F.	24 hours
SR 3.6.2.4.3	Verify each SPMU subsystem manual, power operated, and automatic valve that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
[SR 3.6.2.4.4	Verify all upper containment pool gates are in the stored position or are otherwise removed from the upper containment pool.	31 days]
SR 3.6.2.4.5	------------------NOTE------------------ Actual makeup to the suppression pool may be excluded. -- Verify each SPMU subsystem automatic valve actuates to the correct position on an actual or simulated automatic initiation signal.	[18] months

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ISTS: 3.6.2.4 - SUPPRESSION POOL MAKEUP (SPMU) SYSTEM

1. The LaSalle 1 and 2 design does not include a Suppression Pool Makeup System. Therefore, this Specification has been deleted.

<CTS>

3.6 CONTAINMENT SYSTEMS

3.6.3.1 Primary Containment Hydrogen Recombiners (~~If~~ permanently ~~installed~~) 1

<LCO 3.6.6.1> LCO 3.6.3.1 Two primary containment hydrogen recombiners shall be OPERABLE.

<Appl 3.6.6.1> APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.6.1 Act> 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
<DOC L.2> [B. Two primary containment hydrogen recombiners inoperable.	B.1 Verify by administrative means that the hydrogen control function is maintained. AND B.2 Restore one primary containment hydrogen recombiner to OPERABLE status.	1 hour AND c 2 One per 12 hours thereafter 7 days] 3

(continued)

<CTS>

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.6.1 Act>	C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
<4.6.6.1.b>	SR 3.6.3.1.1 Perform a system functional test for each primary containment hydrogen recombiner.	[18] 24 months 3
	~~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~~ 4
<4.6.6.1.c.2>	SR 3.6.3.1.3 2 3 Perform a resistance to ground test for each heater phase.	[18] 24 months 3

<Insert ITS 3.6.3.2 (BWR/4 ISTS 3.6.3.3)>

5

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.6.3.1 - PRIMARY CONTAINMENT HYDROGEN RECOMBINERS

1. This reviewer's type of note has been deleted. This information is for the NRC reviewer to be keyed in to what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific submittal.

2. Typographical error corrected.

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

4. The Current LaSalle 1 and 2 Licensing Basis does not include ISTS SR 3.6.3.1.2, which requires a visual examination of each primary containment hydrogen recombiner enclosure and verification that there is no evidence of abnormal conditions. CTS 4.6.6.1.b (ITS SR 3.6.3.1.1) and CTS 4.6.6.1.c.2 (ITS SR 3.6.3.1.2) require a Hydrogen Recombiner system functional test and a heater resistance to ground test, respectively. This CTS testing, which is maintained in the ITS, provides adequate periodic surveillance testing to ensure the Operability of the Hydrogen Recombiners. This testing includes verification of system leak tightness during Integrated Leak Rate Testing. Accordingly, ComEd concludes that requiring a visual examination of each primary containment recombiner enclosure at periodic intervals is not necessary and ISTS SR 3.6.3.1.2 has not been included in the LaSalle 1 and 2 ITS.

5. A new Specification has been added, ITS 3.6.3.2. This Specification is from the BWR/4 ISTS (NUREG-1433 ISTS 3.6.3.3), since the LaSalle 1 and 2 design is similar to the BWR/4 design with regard to oxygen concentration requirement (LaSalle 1 and 2 inerts the primary containment since the containment is a Mark II). Therefore, the BWR/4 LCO is used and any deviations from the BWR/4 ISTS are discussed in the Justification for Deviations for ITS: 3.6.3.2.

1

3.6 CONTAINMENT SYSTEMS

3.6.3.2 Primary Containment and Drywell Hydrogen Ignitors

LCO 3.6.3.2 Two divisions of primary containment and drywell hydrogen ignitors shall be OPERABLE, each with > 90% of the associated ignitor assemblies OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One primary containment and drywell hydrogen ignitor division inoperable.	A.1	---------NOTE--------- LCO 3.0.4 is not applicable. --------------------- Restore primary containment and drywell hydrogen ignitor division to OPERABLE status.	30 days
B. Two primary containment and drywell hydrogen ignitor divisions 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 and drywell hydrogen ignitor division 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 3.6.3.2.1	Energize each primary containment and drywell hydrogen ignitor division and perform current versus voltage measurements to verify required ignitors in service.	184 days
SR 3.6.3.2.2	------------------NOTE------------------ Not required to be performed until 92 days after discovery of four or more ignitors in the division inoperable. -- Energize each primary containment and drywell hydrogen ignitor division and perform current versus voltage measurements to verify required ignitors in service.	92 days
SR 3.6.3.2.3	Verify each required ignitor in inaccessible areas develops sufficient current draw for a ≥ [1700]°F surface temperature.	[18] months

(continued)

1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.2.4	Verify each required ignitor in accessible areas develops a surface temperature of ≥ [1700]°F.	[18] months

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ISTS: 3.6.3.2 - PRIMARY CONTAINMENT AND DRYWELL HYDROGEN IGNITORS

1. The LaSalle 1 and 2 design does not include Primary Containment and Drywell Hydrogen Ignitors. Therefore, this Specification has been deleted.

1 Insert BWR/4 ISTS 3.6.3.3*

<CTS>

3.6 CONTAINMENT SYSTEMS

3.6.3.3 2 1 Primary Containment Oxygen Concentration

<LCO 3.6.6.2> LCO 3.6.3.3 The primary containment oxygen concentration shall be < 4.0 volume percent.

<Appl 3.6.6.2> 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.

2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.6.2 Act> A. Primary containment oxygen concentration not within limit.	A.1 Restore oxygen concentration to within limit.	24 hours
<3.6.6.2 Act> B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ [15]% RTP. 2	8 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<4.6.6.2> SR 3.6.3.3.1 2 1	Verify primary containment oxygen concentration is within limits.	7 days

1 * This BWR/4 Specification Insert was used because it best represented the LaSalle 1 and 2 design

Insert Page 3.6-38a

1. A new Specification has been added, ITS 3.6.3.2. This Specification is from the BWR/4 ISTS (NUREG-1433 ISTS 3.6.3.3), since the LaSalle 1 and 2 design is similar to the BWR/4 design with regard to the inerting requirements of the primary containment. Therefore, the BWR/4 LCO is used and any deviations from the BWR/4 ISTS are discussed below.

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

1

3.6 CONTAINMENT SYSTEMS

3.6.3.3 [Drywell Purge System]

LCO 3.6.3.3 Two [drywell purge] subsystems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One [drywell purge] subsystem inoperable.	A.1	---------NOTE--------- LCO 3.0.4 is not applicable. --------------------- Restore [drywell purge] subsystem to OPERABLE status.	30 days
B. Two [drywell purge] 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 [drywell purge] subsystem 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 3.6.3.3.1	Operate each [drywell purge] subsystem for ≥ [15] minutes.	92 days
[SR 3.6.3.3.2	Verify each [drywell purge] subsystem flow rate is ≥ [500] scfm.	[18] months]

1

1. The LaSalle 1 and 2 design includes a Drywell Purge System; however, it is not utilized as an Engineered Safety Feature, and is not included in the CTS. The LaSalle primary containment is a Mark II design, and does not credit a Drywell Purge System for reducing hydrogen concentration. Consequently, the Drywell Purge System has not been included in the ITS for LaSalle 1 and 2.

<CTS>

$\boxed{1}$

3.6 CONTAINMENT SYSTEMS

3.6.4.1 [Secondary Containment]

<LCO 3.6.5.1> LCO 3.6.4.1 The [secondary containment] shall be OPERABLE.

<Appl 3.6.5.1> APPLICABILITY: MODES 1, 2, and 3,
[During movement of irradiated fuel assemblies in the $\boxed{1}$ [~~primary or~~ secondary containment],
$\boxed{1}$ During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).]

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.5.1 Act a> A. [Secondary containment] inoperable [in MODE 1, 2, or 3]. $\boxed{1}$	A.1 Restore $\boxed{1}$ [secondary containment] to OPERABLE status.	4 hours
<3.6.5.1 Act a> B. Required Action and associated Completion Time [of Condition A] not met. $\boxed{1}$	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.5.1 Act b> C. Secondary containment inoperable during movement of irradiated fuel assemblies in the ~~[primary or]~~ secondary containment, during CORE ALTERATIONS, or during OPDRVs. 1	C.1 ---------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Suspend movement of irradiated fuel assemblies in the ~~[primary and]~~ secondary containment. 1	Immediately
	AND	
	C.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	C.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<4.6.5.1.a> 1 SR 3.6.4.1.1	Verify secondary containment vacuum is ≥ 0.25 inch of vacuum water gauge. 1	24 hours
~~SR 3.6.4.1.2~~	~~Verify all [secondary containment] equipment hatches are closed and sealed.~~	~~31 days~~ 2

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE TSTF-18	FREQUENCY
<4.6.5.1.b.1> SR 3.6.4.1.3 2 2 Verify ~~each~~ one [secondary containment] 1 access door is closed, [except when the access opening is being used for entry and exit[, then at least one door shall be closed]]. in each access opening TSTF-18	3 31 days TSTF-18
<4.6.5.1.c.1> [SR 3.6.4.1.4 3 2 Verify ~~each~~ 4 standby gas treatment (SGT) subsystem ~~will draw down~~ can be drawn down 4 the [secondary containment] to ≥ [0.25] inch of vacuum water gauge in ≤ [120] 1 300 seconds using one 4]	[18] 24 1 months on a STAGGERED TEST BASIS. 1
<4.6.5.1.c.1> SR 3.6.4.1.5 4 2 Verify ~~each~~ SGT subsystem can be 4 maintained ≥ [0.266] 1 0.25 inch of vacuum water gauge in 2 the [secondary containment] for 1 hour at a flow rate ≤ [4000] 4400 1 cfm. 1 using one 4	[18] 24 1 months on a STAGGERED TEST BASIS. for each SGT subsystem 4

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

2. ISTS SR 3.6.4.1.2 requires verification that all secondary containment equipment hatches are closed and sealed every 31 days. This Surveillance Requirement is not required in the LaSalle 1 and 2 CTS and has not been included in the ITS. At LaSalle 1 and 2, all equipment access openings are provided with inner and outer doors and are treated as access doors. As a result, they will be subject to the verification requirements of ITS SR 3.6.4.1.2 (ISTS SR 3.6.4.1.3), which verifies the position of secondary containment access doors. Therefore, ISTS SR 3.6.4.1.2 is not required. This is consistent with the current licensing basis. In addition, the following SRs have been renumbered due to this deletion.

3. The ISTS SR 3.6.4.1.3 (ITS SR 3.6.4.1.2) allowance that both doors can be open during entry and exit has been deleted. This is consistent with the same SR in NUREG-1433, Rev. 1. The LaSalle 1 and 2 design with respect to the number of doors in an access opening is consistent with the BWR/4 design (2 doors per access opening), not the BWR/6 design (one door per access opening).

4. ISTS SRs 3.6.4.1.4 and 3.6.1.4.5 are tests that ensure the Secondary Containment is Operable; the leak tightness of the Secondary Containment boundary is within the assumptions of the accident analyses. However, they are written in such a manner that they imply that if a SGT subsystem is inoperable, the SRs are failed ("Verify each standby gas treatment (SGT) subsystem will/can..."). As stated above, this is not the intent of the SRs. Therefore, to ensure this misinterpretation cannot occur, the SRs have been rephrased to more clearly convey the original intent of the SRs, to verify the Secondary Containment is Operable. With the new wording, if a SGT subsystem is inoperable, ITS SRs 3.6.4.1.3 and 3.6.4.1.4 will still be met and only the SGT System Specification, LCO 3.6.4.3, will be required to be entered. The SRs will still ensure each SGT subsystem is used (on a STAGGERED TEST BASIS) to perform the SRs. This change is also consistent with TSTF-322.

<CTS>

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

<LCO 3.6.5.2> LCO 3.6.4.2 Each SCIV shall be OPERABLE.

<Appl 3.6.5.2> APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the ~~[primary or]~~ secondary containment~~s,~~ $\boxed{1}$
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.2> 2. Separate Condition entry is allowed for each penetration flow path.

<DOC A.2> 3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIVs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.5.2 Act a> <4.6.5.1.b.2> 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)

<CTS>

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
<3.6.5.2 Act a> A. (continued) 2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative controls.	A.2	--------NOTE-------- 5 1. 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
<DOC L.2> 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
<3.6.5.2 Act a> C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 AND C.2	Be in MODE 3. Be in MODE 4.	12 hours 36 hours

TSTF-269

(continued)

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
<3.6.5.2 Act c> D. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies in the ~~[primary or]~~ secondary containment, ~~*~~ during CORE ALTERATIONS, or during OPDRVs. 1	D.1	--------NOTE--------- LCO 3.0.3 is not applicable. ---------------------	
		Suspend movement of irradiated fuel assemblies in the ~~[primary and]~~ secondary containment ~~*~~. 1	Immediately
	AND		
	D.2	Suspend CORE ALTERATIONS.	Immediately
	AND		
	D.3	Initiate action to suspend OPDRVs.	Immediately

<CTS>

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
<4.6.5.1.b.2> SR 3.6.4.2.1	------------------NOTES------------------ 1. Valves and blind flanges in high radiation areas may be verified by use of administrative ~~controls~~. means 2. Not required to be met for SCIVs that are open under administrative ~~means~~. controls 2 --- Verify each secondary containment isolation manual valve and blind flange that is not locked, sealed, or otherwise secured in position and is required to be closed during accident conditions is closed.	31 days TSTF-45 rev.2
<4.6.5.2.c> SR 3.6.4.2.2 TSTF-46	Verify the isolation time of each power operated ~~and each~~ automatic SCIV is within limits.	[~~In accordance with the Inservice Testing Program or~~ 92 days] 1
<4.6.5.2.b> SR 3.6.4.2.3	Verify each automatic SCIV actuates to the isolation position on an actual or simulated automatic isolation signal.	~~18~~ 24 months 1

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

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

<CTS>

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

<LCO 3.6.5.3> LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

<Appl 3.6.5.3> APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the ~~primary or~~ secondary containment~~s~~, $\boxed{1}$
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.5.3 Act a> A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
<3.6.5.3 Act a.1> 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
<3.6.5.3 Act a.2> C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the ~~primary or~~ secondary containment~~s~~, during CORE ALTERATIONS, or during OPDRVs. $\boxed{1}$	------------NOTE------------ LCO 3.0.3 is not applicable. ---------------------------- C.1 Place OPERABLE SGT subsystem in operation. OR	Immediately (continued)

<CTS>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.5.2 Act a.2> C. (continued)	C.2.1 Suspend movement of irradiated fuel assemblies in the ~~primary and~~ secondary containment~~s~~. $\boxed{1}$	Immediately
	AND	
	C.2.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	C.2.3 Initiate action to suspend OPDRVs.	Immediately
<DOC A.3> D. Two SGT subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3. $\boxed{2}$	Immediately
<3.6.5.3 Act b> E. Two SGT subsystems inoperable during movement of irradiated fuel assemblies in the ~~primary or~~ secondary containment~~s~~, during CORE ALTERATIONS, or during OPDRVs. $\boxed{1}$	E.1 --------NOTE--------- LCO 3.0.3 is not applicable. --------------------- Suspend movement of irradiated fuel assemblies in the ~~primary and~~ secondary containment~~s~~. $\boxed{1}$	Immediately
	AND	
		(continued)

<CTS>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.5.3 Act b> E. (continued)	E.2 Suspend CORE ALTERATIONS. AND E.3 Initiate action to suspend OPDRVs.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<4.6.5.3.a> SR 3.6.4.3.1 $\boxed{1}$	Operate each SGT subsystem for ≥ ~~[~~10~~]~~ continuous hours ~~[~~with heaters operating~~]~~.	31 days
<4.6.5.3.b> 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
<4.6.5.3.d.2> SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	~~[18]~~ 24 $\boxed{1}$ months
~~[SR 3.6.4.3.4~~	~~Verify each SGT filter cooler bypass damper can be opened and the fan started.~~	~~[18] months]~~ $\boxed{1}$

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

2. Typographical/grammatical error corrected.

1

3.6 CONTAINMENT SYSTEMS

3.6.5.1 Drywell

LCO 3.6.5.1 The drywell shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell inoperable.	A.1 Restore drywell 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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1.1	Verify 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 is ≤ [10%] of the drywell bypass leakage limit.	[18] months

(continued)

1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.5.1.2	Visually inspect the exposed accessible interior and exterior surfaces of the drywell.	[40] months

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ISTS: 3.6.5.1 - DRYWELL

1. The LaSalle 1 and 2 design does not include a drywell internal to the primary containment (NUREG-1434 is based on a Mark III containment; LaSalle 1 and 2 has a Mark II containment). Therefore, this Specification has been deleted.

1

3.6 CONTAINMENT SYSTEMS

3.6.5.2 Drywell Air Lock

LCO 3.6.5.2 The drywell air lock shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

1. Entry and exit is permissible to perform repairs of the affected air lock components.
2. Enter applicable Conditions and Required Actions of LCO 3.6.5.1, "Drywell," when air lock leakage results in exceeding overall drywell bypass leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One drywell 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
	<u>AND</u>	
		(continued)

1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Lock the OPERABLE door closed.	24 hours
	AND	
	A.3 Verify by administrative means the OPERABLE door is locked closed.	Once per 31 days
B. Drywell 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 and exit is permissible under the control of a dedicated individual. ---------------------------	
	B.1 Verify an OPERABLE door is closed.	1 hour
	AND	
	B.2 Lock an OPERABLE door closed.	24 hours
	AND	
	B.3 Verify by administrative means an OPERABLE door is locked closed.	Once per 31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Drywell air lock inoperable for reasons other than Condition A or B.	C.1	Initiate action to evaluate drywell overall leakage rate per LCO 3.6.5.1, "Drywell," using current air lock test results.	Immediately
	AND		
	C.2	Verify a door is closed.	1 hour
	AND		
	C.3	Restore air lock to OPERABLE status.	24 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	12 hours
	AND		
	D.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.2.1	-------------------NOTE------------------- Only required to be performed once after each closing. -- Verify seal leakage rate is ≤ [200] scfh when the gap between the door seals is pressurized to ≥ [11.5] psig.	72 hours
SR 3.6.5.2.2	Verify drywell air lock seal air flask pressure is ≥ [90] psig.	7 days
SR 3.6.5.2.3	-------------------NOTE------------------- Only required to be performed upon entry into drywell. -- Verify only one door in the drywell air lock can be opened at a time.	18 months
SR 3.6.5.2.4	-------------------NOTE------------------- An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. -- Verify overall drywell air lock leakage rate is ≤ [200] scfh by performing an overall air lock leakage test at ≥ [11.5] psig.	18 months

(continued)

I

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.5.2.5	Verify, from an initial pressure of [90] psig, the drywell air lock seal pneumatic system pressure does not decay at a rate equivalent to > [30] psig for a period of [10] days.	[18] months

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ISTS: 3.6.5.2 - DRYWELL AIR LOCK

1. The LaSalle 1 and 2 design does not include a drywell internal to the primary containment (NUREG-1434 is based on a Mark III containment; LaSalle 1 and 2 has a Mark II containment). Therefore, this Specification has been deleted.

3.6 CONTAINMENT SYSTEMS

3.6.5.3 Drywell Isolation Valve[s]

LCO 3.6.5.3 Each drywell isolation valve [, except for Drywell Vacuum Relief System valves,] shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

1. Penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by drywell isolation valves.
4. Enter applicable Conditions and Required Actions of LCO 3.6.5.1, "Drywell," when drywell isolation valve leakage results in exceeding overall drywell bypass leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one drywell isolation valve inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. AND	8 hours
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	--------NOTE--------- Isolation devices in high radiation areas may be verified by use of administrative means. --------------------- Verify the affected penetration flow path is isolated.	Prior to entering MODE 2 or 3 from MODE 4, if not performed within the previous 92 days
B. ---------NOTE--------- Only applicable to penetration flow paths with two isolation valves. ---------------------- One or more penetration flow paths with two drywell isolation valves inoperable.	B.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	4 hours
C. Required Action and associated Completion Time not met.	C.1 AND C.2	Be in MODE 3. Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
[SR 3.6.5.3.1	Verify each [ ] inch drywell purge isolation valve is sealed closed.	31 days]
[SR 3.6.5.3.2	-------------------NOTE------------------- Not required to be met when the drywell purge supply or exhaust valves are open for pressure control, ALARA or air quality considerations for personnel entry, or Surveillances that require the valves to be open [provided the [20] inch containment [purge system supply and exhaust] lines are isolated]. --	
	Verify each [20] inch drywell purge isolation valve is closed.	31 days]
SR 3.6.5.3.3	-------------------NOTE------------------- Not required to be met for drywell isolation valves that are open under administrative controls. --	
	Verify each drywell isolation manual valve and blind flange that is required to be closed during accident conditions is closed.	Prior to entering MODE 2 or 3 from MODE 4, if not performed in the previous 92 days

(continued)

1

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.6.5.3.4	Verify the isolation time of each power operated, ~~and each~~ automatic drywell isolation valve is within limits. TSTF-46	[In accordance with the Inservice Testing Program or 92 days]
SR 3.6.5.3.5	Verify each automatic drywell isolation valve actuates to the isolation position on an actual or simulated isolation signal.	[18] months
[SR 3.6.5.3.6	Verify each [ ] inch drywell purge isolation valve is blocked to restrict the valve from opening > [50]%.	[18] months]

1. The LaSalle 1 and 2 design does not include a drywell internal to the primary containment (NUREG-1434 is based on a Mark III containment; LaSalle 1 and 2 has a Mark II containment). Therefore, this Specification has been deleted.

1

3.6 CONTAINMENT SYSTEMS

3.6.5.4 Drywell Pressure

LCO 3.6.5.4 Drywell-to-primary containment differential pressure shall be [≥ -0.26 psid and ≤ 2.0 psid].

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell-to-primary containment differential pressure not within limits.	A.1 Restore drywell-to-primary containment differential pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.5.4.1 Verify drywell-to-primary containment differential pressure is within limits.	12 hours

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ISTS: 3.6.5.4 - DRYWELL PRESSURE

1. The LaSalle 1 and 2 design does not include a drywell internal to the primary containment (NUREG-1434 is based on a Mark III containment; LaSalle 1 and 2 has a Mark II containment). Therefore, this Specification has been deleted.

1

3.6 CONTAINMENT SYSTEMS

3.6.5.5 Drywell Air Temperature

LCO 3.6.5.5 Drywell average air temperature shall be ≤ [135]°F.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Drywell average air temperature not within limit.	A.1	Restore drywell average air temperature to within limit.	8 hours
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
SR 3.6.5.5.1	Verify drywell average air temperature is within limit.	24 hours

1. The LaSalle 1 and 2 design does not include a drywell internal to the primary containment (NUREG-1434 is based on a Mark III containment; LaSalle 1 and 2 has a Mark II containment). Therefore, this Specification has been deleted.

3.6 CONTAINMENT SYSTEMS

3.6.5.6 Drywell Vacuum Relief System

LCO 3.6.5.6 [Two] drywell post-LOCA and [two] drywell purge vacuum relief subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

-----------------------------------NOTE-----------------------------------
Enter applicable Conditions and Required Actions of LCO 3.6.5.1, "Drywell," when inoperable drywell purge vacuum relief subsystem(s) results in exceeding overall drywell bypass leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Separate Condition entry is allowed for each vacuum relief subsystem. ---------------------- One or more vacuum relief subsystems not closed.	A.1 Close the subsystem.	4 hours
B. One or [two] drywell post-LOCA vacuum relief subsystems inoperable for reasons other than Condition A.	B.1 Restore drywell post-LOCA vacuum relief subsystem(s) to OPERABLE status.	30 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One drywell purge vacuum relief subsystem inoperable for reasons other than Condition A.	C.1 Restore drywell purge vacuum relief subsystem to OPERABLE status.	30 days
D. [Two] drywell purge vacuum relief subsystems inoperable for reasons other than Condition A.	D.1 Restore one drywell purge vacuum relief subsystem to OPERABLE status.	72 hours
E. [Two] drywell post-LOCA vacuum relief subsystems inoperable for reasons other than Condition A. AND One drywell purge vacuum relief subsystem inoperable for reasons other than Condition A.	E.1 Restore one drywell post-LOCA vacuum relief or drywell purge vacuum relief subsystem to OPERABLE status.	72 hours
F. Required Action and associated Completion Time of Condition A, B, C, D, or E not met.	F.1 Be in MODE 3. AND F.2 Be in MODE 4.	12 hours 36 hours

(continued)

1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. [Two] drywell purge vacuum relief subsystems inoperable for reasons other than Condition A. AND One or [two] drywell post-LOCA vacuum relief subsystems inoperable for reasons other than Condition A.	G.1 Be in MODE 3. AND G.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.6.1	-----NOTES----- 1. Not required to be met for drywell purge vacuum relief breakers open during Surveillances. 2. Not required to be met for vacuum breakers open when performing their intended function. --------------- Verify each vacuum breaker and its associated isolation valve is closed.	7 days
SR 3.6.5.6.2	Perform a functional test of each vacuum breaker and its associated isolation valve.	31 days

(continued)

[1]

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.5.6.3	Verify the opening setpoint of each vacuum breaker is ≤ [1.0] psid.	[18] months

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ISTS: 3.6.5.6 - DRYWELL VACUUM RELIEF SYSTEM

1. The LaSalle 1 and 2 design does not include a drywell internal to the primary containment (NUREG-1434 is based on a Mark III containment; LaSalle 1 and 2 has a Mark II containment). Therefore, this Specification has been deleted.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.1 Primary Containment

BASES

BACKGROUND

The function of the primary containment is to isolate and contain fission products released from the Reactor Primary System following a ~~Design Basis Accident (DBA)~~ Loss of Coolant (LOCA) ~~and to~~ confine the postulated release of radioactive material to within limits. The primary containment consists of a steel lined, reinforced concrete vessel, which surrounds the Reactor Primary System and provides an essentially leak tight barrier against an uncontrolled release of radioactive material to the environment. Additionally, this structure provides shielding from the fission products that may be present in the primary containment atmosphere following accident conditions. 1

The isolation devices for the penetrations in the primary containment boundary are a part of the primary 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. Primary containment air locks are OPERABLE, except as provided in LCO 3.6.1.2, "Primary Containment Air Locks";

c. All equipment hatches are closed ~~; and~~ and sealed. 1

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

The sealing mechanism associated with each primary containment penetration (e.g., welds, bellows, or O-rings) is OPERABLE. 5

(continued)

BASES

BACKGROUND (continued)

This Specification ensures that the performance of the primary containment, in the event of a DBA, meets the assumptions used in the safety analyses of References 1 and 2. SR 3.6.1.1.1 leakage rate requirements are in conformance with 10 CFR 50, Appendix J (Ref. 3), as modified by approved exemptions.

Design Basis Accident (DBA) 1

, Option B 1

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.

0.635 5 1

The maximum allowable leakage rate for the primary containment (L_a) is [0.437]% by weight of the containment and drywell air per 24 hours at the maximum peak containment pressure (P_a) of [11.5] psig (Ref. 4).

1 design basis LOCA

39.6 5

Primary containment satisfies Criterion 3 of the NRC Policy Statement.

10 CFR 50.36(c)(2)(ii). 1

LCO

Primary Containment Leakage Rate Testing Program

3 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, leakage test. At this time, the combined Type B and C leakage must be < 0.6 L_a, and the overall Type A leakage must be < 0.75 L_a. Compliance with this LCO will ensure a primary containment

met. limits

applicable

(continued)

In addition, the leakage from the drywell to the suppression chamber must be limited to ensure the primary containment pressure does not exceed design limits.

BASES

LCO (continued)	configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis. Individual leakage rates specified for the primary containment air locks 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 that 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 that is 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 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 3.6.1.1.1

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 (SR 3.6.1.2.1 and SR 3.6.1.2.4), secondary containment bypass leakage (SR 3.6.1.3.9), resilient seal primary containment purge valve leakage testing (SR 3.6.1.3.6), or main steam isolation valve leakage (SR 3.6.1.3.10) 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 10 CFR 50, Appendix J, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply.

4 limit

the Primary Containment Leakage Rate Testing Program 3

3 2 3 4 4 1

SR 3.6.1.1.2

The structural integrity of the primary containment is ensured by the successful completion of the Primary Containment Tendon Surveillance Program and by associated visual inspections of the steel liner and penetrations for evidence of deterioration or breach of integrity. This ensures that the structural integrity of the primary containment will be maintained in accordance with the provisions of the Primary Containment Tendon Surveillance Program. Testing and Frequency are consistent with the recommendations of Regulatory Guide 1.35 (Ref. 5).

1 Inservice Inspection Program for Post Tensioning Tendons

5

REFERENCES 1

1. U FSAR, Section [6.2]. 5
2. U FSAR, Section [15.6.5].

, except that the Unit 1 and 2 primary containments shall be treated as twin containments even though the Initial Structural Integrity tests were not within two years of each other. 1

(continued)

<Insert SR 3.6.1.1.3> 3

Insert SR 3.6.1.1.3

SR 3.6.1.1.3

The analyses results in Reference 6 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 0.030 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 0.052 ft^2, corresponding to a 3-in line size.

As left bypass leakage, prior to the first startup after performing a required bypass leakage test, is required to be $\leq$ 10% of the drywell-to-suppression chamber bypass leakage limit when tested with an initial differential pressure of 1.5 psi. 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

REFERENCES (continued)

3. 10 CFR 50, Appendix J, Option B 1 1
4. UFSAR, Section ~~[ ]~~ 6.2.6.1 5
5. Regulatory Guide 1.35, Revision ~~[1]~~ 3 5
6. UFSAR, Section 6.2.1.1.5. 1

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, 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, or analysis description.

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

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 3.6.1.2 Primary Containment Air Lock~~s~~

BASES

BACKGROUND

A ~~Two~~ double door primary containment air lock~~s have~~ has been built into the primary containment to provide personnel access to the primary containment and to provide primary containment isolation during the process of personnel entry and exit. The air lock~~s are~~ 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 pressure in excess of the maximum expected pressure following a DBA in primary containment. double, compressible ~~Each of the doors has inflatable seals that are maintained > [60] psig by the seal air flask and pneumatic system, which is maintained at a pressure ≥ [90] psig. Each door has two seals to ensure they are single failure proof in maintaining the leak tight boundary of primary containment.~~

and local leak rate testing capability to ensure pressure integrity. To effect a leak tight seal, the air lock design uses pressure sealed doors (i.e., an increase in primary containment internal pressure results in an increased sealing on each door.)

The ~~Each~~ air lock is nominally a right circular cylinder, 10 ft ~~2 inches~~ in diameter, with doors at each end that are interlocked to prevent simultaneous opening. The air lock~~s~~ is ~~are~~ provided with limit switches on both doors ~~in each air lock~~ that provide remote ~~control room~~ indication of door position ~~[Additionally, control room indication is provided to alert the operator whenever an air lock interlock mechanism is defeated.]~~ via an alarm in the control room that indicates when an air lock door is open. During periods when primary containment is not required to be OPERABLE, the air lock interlock mechanism may be disabled, allowing both doors of the ~~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 door interlock mechanism has failed, by manually performing the interlock function.

The primary containment air lock~~s~~ form~~s~~ part of the primary containment pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining primary containment leakage rate to within limits in the event of a

(continued)

BASES

BACKGROUND (continued)

DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the ~~unit~~ 2 safety analysis.

APPLICABLE SAFETY ANALYSES

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 mass 1 2 ~~[0.437]~~ 0.635 3 % by weight of the containment ~~and drywell~~ air per 24 hours at the ~~calculated~~ Design Basis LOCA 1 maximum peak containment pressure (P_a) of ~~[11.5]~~ 39.6 3 psig (Ref. 2) 1. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock~~s~~ 1.

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.

1 Primary containment air lock~~s~~ satisf~~y~~ies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii).

LCO

2 As part of the primary containment pressure boundary, the air lock~~s~~ 4 safety function is related to control of containment leakage ~~rates~~ 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~~s~~ 1 ~~are~~ is 1 required to be OPERABLE. For ~~each~~ 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 open at a time. This provision ensures that a gross breach of primary containment does not exist when primary containment is required to be OPERABLE. Closure of a single door in ~~each~~ the 1 air lock is

(continued)

BASES

LCO (continued)

sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into ~~and~~ or exit from primary containment. 4

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 for most repairs. ~~It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door, then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which~~ the primary containment boundary is not intact (during access through the OPERABLE door). The ~~ability~~ allowance 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. ~~After each entry and exit, the OPERABLE door must be immediately closed.~~ 2

1 If the inner door is the one that is inoperable, however, then a short time exists when

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

~~Note 2 has been included to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required Actions may allow for continued operation, and a subsequent inoperable air lock is governed by subsequent Condition entry and application of associated Required Actions.~~ 5

(continued)

BASES

ACTIONS (continued)

second 5

The ACTIONS are modified by a ~~third~~ Note, which ensures appropriate remedial actions 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.

2 La

; if airlock leakage results in exceeding overall containment leakage rate acceptance criteria

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

5

With one primary containment air lock door inoperable ~~in one or more primary containment air locks~~, the OPERABLE door must be verified closed (Required Action A.1) ~~in each affected air lock~~. This ensures that a leak tight primary 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 ~~affected~~ 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 the OPERABLE door of the ~~affected~~ air lock is being maintained closed.

2 penetration

Required Action A.3 ensures that the ~~affected~~ 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.

2 given

2 Required Action A.3 is modified by a Note that applies to air lock doors located in high radiation areas and allows these doors to be verified locked closed by use of administrative 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.

5 or areas with limited access due to inerting

(continued)

BASES

ACTIONS

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

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 if both air locks have an inoperable door. This 7 day restriction begins when the second air lock is discovered inoperable. 5

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

Primary containment entry may be required to perform Technical Specifications (TS) Surveillances and Required 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. 6

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

With an the air lock interlock mechanism inoperable in one or both primary containment air locks, the Required Actions and associated Completion Times are consistent with those specified in Condition A. 5

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 one 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

(continued)

BASES

ACTIONS

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

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 5 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.

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

With ~~one or more~~ the 5 air lock~~s~~ 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~~ the 5 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.

Required Action C.2 requires that one door in the ~~affected~~ 5 primary containment air locks must be verified closed. This Required 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.

2 (Required Action C.3)

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

5

(continued)

BASES

ACTIONS (continued)

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.

SURVEILLANCE REQUIREMENTS

5 the Primary Containment Leakage Rate Testing Program

SR 3.6.1.2.1

the J 1

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 regard 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, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply~~.

as a small fraction of the total allowable primary containment leakage 3

5 the Primary Containment Leakage Rate Testing Program

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR, requiring the results to be evaluated against the acceptance criteria of SR 3.6.1.1.1. This ensures that air lock leakage is properly accounted for in determining the ~~overall~~ primary containment leakage rate.

5 which is applicable to

Combined Types B and C 1

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

~~SR 3.6.1.2.2~~

~~The seal air flask pressure is verified to be at ≥ [90] psig every 7 days to ensure that the seal system remains viable. It must be checked because it could bleed down during or following access through the air lock, which occurs regularly. The 7 day Frequency has been shown to be acceptable through operating experience and is considered adequate in view of the other indications available to operations personnel that the seal air flask pressure is low.~~ 5

SR 3.6.1.2.~~3~~ 2 5

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~~ the air lock are designed to withstand the maximum expected post accident primary containment pressure (Ref. ~~3~~ 2), 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~~ not normally challenged when the primary containment air lock door is ~~opened~~ used for entry and exit (procedures require strict adherence to single door opening), this test is only required to be performed ~~upon entering or exiting a primary containment air lock, but is not required more frequently than once per 184 days.~~ The ~~184 day~~ 24 month Frequency is based on engineering judgment and is considered adequate ~~in view of other administrative controls [such as indications of interlock mechanism status available to operations personnel].~~ given that the interlock is not challenged during use of the air lock.

TSTF-17 1 2 7 Insert SR 3.6.1.2.2 TSTF-17 4

~~SR 3.6.1.2.4~~

~~A seal pneumatic system test to ensure that pressure does not decay at a rate equivalent to > [2] psig for a period of [48] hours from an initial pressure of [90] psig is an effective leakage rate test to verify system performance. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a~~ 5

(continued)

Insert SR 3.6.1.2.2

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. 7

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.6.1.2.4 (continued)~~

~~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, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~ 5

REFERENCES

1 u
1. FSAR, Section ~~[3.8]~~ 1.1.3.5.1 3

1
~~2. 10 CFR 50, Appendix J.~~

1 2 u
~~3.~~ FSAR, ~~Table [6.2-13]~~ Section 6.2.6.1 3

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, 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, or analysis description.

2. Editorial change made for enhanced clarity.

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

4. Typographical/grammatical error corrected.

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

6. 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.

7. 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 PCIVs 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 the primary containment function assumed in the safety analysis will be maintained. These isolation devices consist of either passive devices or active (automatic) devices. Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analysis. One of these barriers may be a closed system.

(which include plugs and caps, as listed in Ref. 1)

1

1

, except for penetrations isolated by excess flow check valves,

8 2 26 26 2

2 The 6 and 20 inch primary containment purge valves are 8 PCIVs that are qualified for use during all operational conditions. The 6 and 20 inch primary containment purge valves are normally maintained closed in MODES 1, 2, and 3 to ensure ~~leak tightness~~. The purge valves must be closed when not being used for pressure control, ALARA, or air quality considerations ~~to ensure that the primary containment boundary assumed in the safety analysis will be maintained~~.

3

the primary containment boundary is maintained

However, these

may

open

1

Since they are fully qualified.

1 inerting, de-inerting,

(continued)

BASES (continued)

5 4

APPLICABLE SAFETY ANALYSES

The PCIV 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 for which the consequences are mitigated by PCIVs are a loss of coolant accident (LOCA), and a main steam line break (MSLB), ~~and a fuel handling accident inside primary containment~~ (Refs. 1 and 2). In the analysis for each of these accidents, it is assumed that PCIVs are either closed or function to close within the required isolation time 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 1, 2, and 3, the MSLB LOCA is the ~~most~~ limiting event due to radiological consequences. For the MSLB, 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 MSLB analysis (Ref. 3). ~~The safety analyses assume that the purge valves are closed at event initiation.~~ Likewise, it is assumed that the primary containment ~~is isolated~~ isolates such that release of fission products to the environment is controlled.

3 second closure time is assumed in the MSIV closure (the most severe overpressurization transient) analysis (Ref. 4) and the

The DBA analysis assumes that ~~within 60 seconds after the accident,~~ isolation of the primary containment is complete and leakage terminated, except for the maximum allowable leakage, L_a, prior to fuel damage. ~~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.~~

5

2 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 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~[The purge valves may be unable to close in the environment following a LOCA. Therefore, each of the purge valves is required to remain sealed closed during MODES 1, 2, 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. Again, 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.]~~ 6

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. ~~[Primary containment purge valves that are not qualified to close under accident conditions must be sealed closed [or blocked to prevent full opening] to be OPERABLE.]~~ 1 7 The valves covered by this LCO are listed with their associated stroke times in the ~~FSAR~~ Technical Requirements Manual (Ref. ~~2~~ 1). 1

The normally closed manual PCIVs and blind flanges are in place, are considered OPERABLE when the ~~manual~~ valves are closed or open ~~in accordance with appropriate~~ under administrative controls, ~~automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact.~~ 15 These passive isolation valves and devices are those listed in Reference ~~2~~ 1. ~~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. 1 7

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.

(continued)

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

BASES (continued)

APPLICABILITY

Normally 1

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 are required to be OPERABLE, however, to prevent inadvertent reactor vessel draindown and release of radioactive material during a postulated fuel handling accident. 1 These valves are those whose associated instrumentation is required to be OPERABLE according to LCO 3.3.6.1, "Primary Containment Isolation Instrumentation." (This does not include the valves that isolate the associated instrumentation.)

ACTIONS

The ACTIONS are modified by a Note allowing penetration flow path(s) ~~[except for the [ ] inch primary containment purge valve flow path(s)]~~ 7 to be unisolated intermittently under administrative controls. ~~[The primary containment purge valve exception applies to primary containment purge valves that are not qualified to close under accident conditions.]~~ 6 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. ~~Due to the size of the containment purge line penetration and the fact that those penetrations exhaust directly from the primary containment atmosphere to the environment, the penetration flow path containing these valves may not be opened under administrative controls. A single purge valve in a penetration flow path may be opened to effect repairs to an inoperable valve, as allowed by the exception to SR 3.6.1.3.1 and Note 2 to SR 3.6.1.3.2.~~ 7

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

(continued)

BASES

ACTIONS (continued)

subsequent Condition entry and application of associated Required Actions.

The ACTIONS are modified by Notes 3 and 4. Note 3 ensures 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 ~~are~~ be 9 taken.

A.1 and A.2

16 MSIV leakage rate or hydrostatically tested line leakage rate

With one or more penetration flow paths with one PCIV inoperable except for ~~purge valve or secondary containment bypass leakage~~ 2 not within limits, 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, a blind flange, and a check valve with flow through the valve secured. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available one to the primary containment. The Required Action must be completed within the 4 hour Completion Time (8 hours for main steam lines). The specified time period 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 must be verified to be isolated on a periodic basis. This is necessary to ensure that primary containment penetrations required to be isolated following

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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 the primary containment, drywell, and steam tunnel and capable of being mispositioned are in the correct position. The Completion Time for this verification of "once per 31 days for isolation devices outside primary containment, drywell, and steam tunnel" is appropriate because the devices are operated under administrative controls and the probability of their misalignment is low. For devices inside the primary containment, drywell, or steam tunnel, the specified time period of "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 devices and the existence of other administrative controls ensuring that device misalignment is an unlikely possibility. 7

7 if primary containment was de-inerted while in MODE 4

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

16 or more

two 5 Note 1 TSTF-269

Required Action A.2 is modified by a Note that applies to isolation devices 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 devices, once they have been verified to be in the proper position, is low. 10

TSTF-269 Insert A.1 and A.2

10

B.1

16 except for MSIV leakage rate or hydrostatically tested line leakage rate not within limit

With one or more penetration flow paths with two PCIVs inoperable, 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. or more 16

(continued)

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)

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.

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.

7

16

except for MSIV leakage rate or hydrostatically tested line leakage rate not within limit

C.1 and C.2

When one or more penetration flow paths with one PCIV inoperable, except for MSIV leakage rate or hydrostatically tested line leakage rate not within limit, 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 ~~[4]~~ 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 ~~[4]~~ 4 hour Completion Time 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 MODES 1, 2, and 3. The closed system must meet the requirements of Reference 5. In the event the affected penetration is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This 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 that each affected penetration is isolated~~ is appropriate because the ~~valves~~ devices are operated under administrative controls and the probability of their misalignment is low.

TSTF-30

72

16

for a penetration with a closed system

11

4 hours except for excess flow check valve (EFCV) lines and penetrations with a closed system and

11 16

for EFCVs and penetrations with a closed system

16

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.

TSTF-30

7

Condition C is modified by a Note indicating this Condition is applicable only to those penetration flow paths with only one PCIV. For penetration flow paths with two or more PCIVs, Conditions A and B provide the appropriate Required Actions. This Note is necessary since this Condition is written

11 16

The Completion Time of 72 hours for EFCVs is also reasonable considering the mitigating effects of a small pipe diameter and restricting orifice and the isolation boundary provided by the instrument.

(continued)

1

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.

BASES

ACTIONS

C.1 and C.2 (continued)

specifically to address those penetrations with a single PCIV.

two (5) Note 1 TSTF-269

Required Action C.2 is modified by a Note ~~that~~ applies to ~~valves and blind flanges~~ 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.

8 isolation devices 8

Insert C.1 and C.2 TSTF-269

D.1

MSIV 16 (SR 3.6.1.3.10) or hydrostatically tested line leakage rate (SR 3.6.1.3.11)

With the ~~secondary containment bypass~~ leakage rate~~s~~ not within limit, the assumptions of the safety analysis ~~are~~ not met. ~~Therefore, the leakage must be restored to within limit within 4 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 isolat~~ion~~ed 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 to be the lesser actual pathway leakage of the two devices. ~~The 4 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.~~

may be 12 4 2

Insert D.1a 16 Insert D.1b

~~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~~

7

(continued)

Insert C.1 and 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.

Insert D.1a

Therefore, the leakage rate must be restored to within limit within the Completion Times appropriate for each type of valve leakage: a) hydrostatically tested line leakage not on a closed system is required to be restored within 4 hours; b) MSIV leakage is required to be restored within 8 hours; and c) hydrostatically tested line leakage on a closed system is required to be restored within 72 hours.

Insert D.1b

The 4 hour Completion Time for hydrostatically tested line leakage not on a closed system is reasonable considering the time required to restore the leakage by isolating the penetration and the relative importance of the hydrostatically tested line leakage to the overall containment function. The Completion Time of 8 hours for MSIV leakage allows a period of time to restore the 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. The 72 hour Completion Time for hydrostatically tested line leakage on a closed system is acceptable based on the available water seal expected to remain as a gaseous fission product boundary during the accident and in many cases, the associated closed system. The closed system must meet the requirements of Reference 5.

BASES

ACTIONS

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

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 3.6.1.3.6. The specified Completion Time is reasonable, considering that one containment purge valve remains closed so that a gross breach of containment does not exist.

In accordance with Required Action 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.

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TSTF-269 Changes not shown

For the containment purge valve with resilient seal that is isolated in accordance with Required Action E.1, SR 3.6.1.3.6 must be performed at least once every [ ] days. This provides assurance that degradation of the resilient seal is detected and confirms that the leakage rate of the containment purge valve does not increase during the time the penetration is isolated. The normal Frequency for SR 3.6.1.3.6 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 acceptable based on operating experience.

(continued)

BASES

7

E

ACTIONS (continued)

E.1 and E.2

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

~~G.1, H.1, I.1,~~ and F.2 7

7

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

If any Required Action and associated Completion Time cannot be met, the plant 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. If suspending the OPDRVs would result in closing the residual heat removal (RHR) shutdown cooling isolation valves, an alternative Required Action is provided to immediately initiate action to restore the valves to OPERABLE status. This allows RHR shutdown cooling to remain in service while actions are being taken to restore the valve.

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2

14

SURVEILLANCE REQUIREMENTS

~~SR 3.6.1.3.1~~

~~Each [ ] inch primary containment purge valve is required to be verified sealed closed at 31 day intervals. This SR is intended to apply to primary containment purge valves that are not fully qualified to open under accident conditions. 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~~

7

(continued)

TSTF-30 changes not shown 13

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.1 (continued)

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. 5), related to primary containment purge valve use during unit operations.

7

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 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 movement of irradiated fuel assemblies), pressurization concerns are not present and the purge valves are allowed to be open.

SR 3.6.1.3.~~2~~ 1 7 8 inch and 26 2 4 ed

2

This SR verifies that the 20 inch 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

7

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.2 (continued) 1 7

2 ~~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.~~ 7

7 ~~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 movement of irradiated fuel assemblies) pressurization concerns are not present and the purge valves are allowed to be open.~~

The SR is modified by a Note (Note 2) 7 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, 7 pressure control, ALARA, or air quality considerations for personnel entry, or for Surveillances that require the valves to be open, ~~provided the drywell [purge supply and exhaust] lines are isolated.~~ Insert SR 3.6.1.3.1 7 These primary containment 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 primary containment ~~purge~~ isolation valve requirements discussed in SR 3.6.1.3.4. 2 7

SR 3.6.1.3.3 TSTF-45

This SR verifies that each primary containment isolation manual valve and blind flange that is located outside primary containment, ~~drywell, and steam tunnel,~~ 1 and not locked, sealed, or otherwise secured 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 of the primary containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those PCIVs outside primary

(continued)

Insert SR 3.6.1.3.1

purge valves and suppression chamber purge valves are not open simultaneously. This is required to prevent a bypass path between the suppression chamber and the drywell, which would allow steam and gases from a LOCA to bypass the downcomers to the suppression pool.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.~~2~~ 2 7 (continued)

containment, and capable of being mispositioned, 8 are in the correct position. Since verification of valve 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.

Two Notes are 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 PCIVs, once they have been verified to be in the proper position, is low. A second Note is included to clarify that PCIVs open under administrative controls are not required to meet the SR during the time 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. 14

SR 3.6.1.3.~~4~~ 3 7

This SR verifies that each 1 primary containment manual isolation valve and blind flange located inside primary containment ~~, drywell, or steam tunnel,~~ and not locked, sealed, or otherwise secured TSTF-45 and required to be closed during accident conditions, is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside the primary containment boundary is within design limits. For PCIVs inside primary containment, 1 ~~drywell, or steam tunnel~~ the Frequency of "prior to entering MODE 2 or 3 from MODE 4 if primary containment was de-inerted while in MODE 4 7, 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.

Two Notes are 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 and personnel safety 7. Therefore, the probability of misalignment of these PCIVs, once they have been verified to be in their proper position, is low. A second Note is included to clarify that PCIVs that are open

(continued)

BASES

14

SURVEILLANCE REQUIREMENTS

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.

3 7

SR 3.6.1.3.6 (continued)

under administrative controls are not required to meet the SR during the time that the PCIVs are open.

7

INSERT SR 3.6.1.3.4

SR 3.6.1.3.5

J 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 3.6.1.3.6. The isolation time test ensures that the valve will isolate in a time period less than or equal to that assumed in the safety analysis. The ~~isolation time and~~ Frequency of this SR ~~are~~ is in accordance with the Inservice Testing Program ~~or 92 days~~.

each

8

5 2

15 5

SR 3.6.1.3.6

For primary containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J (Ref. 4), is required to ensure OPERABILITY. Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation, and the importance of maintaining this penetration leak tight (due to the direct path between primary containment and the environment), a Frequency of 184 days was established. Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (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.

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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

(continued)

Insert SR 3.6.1.3.4

SR 3.6.1.3.4

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 and operating life, as applicable, 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.

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.6.1.3.6 (continued)~~

~~(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.~~

7

SR 3.6.1.3.~~7~~ 6 7

Verifying that the full closure isolation time of each MSIV is within the specified limits is required to demonstrate OPERABILITY. The full closure 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. The Frequency of this SR is ~~[~~in accordance with the Inservice Testing Program ~~or 18 months]~~.

2 1

SR 3.6.1.3.~~8~~ 7 7

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 3.3.6.1.6~~ LCO 3.3.6.1, "Primary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. The ~~[18]~~ 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 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.

8 2

7 Insert SR3.6.1.3.8 and Insert SR3.6.1.3.9

~~SR 3.6.1.3.9~~

~~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 6 are met. The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of~~

7

13 TSTF-30 Changes Not Shown

(continued)

Insert SR 3.6.1.3.8

SR 3.6.1.3.8

This SR requires a demonstration that each EFCV is OPERABLE by verifying that the valve actuates to the isolation position on an actual or simulated instrument line break condition. This SR provides assurance that the instrumentation line EFCVs will perform as designed. 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 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 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Insert SR 3.6.1.3.9

SR 3.6.1.3.9

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 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. Other administrative controls, such as those that limit the shelf life and operating life, as applicable, of the explosive charges, must be followed. The Frequency of 24 months on a STAGGERED TEST BASIS is considered adequate given the administrative controls on replacement charges and the frequency checks of circuit continuity (SR 3.6.1.3.4).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.3.9 (continued)

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.

7

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].]

SR 3.6.1.3.10

2 1 (STF-30 changes not shown)

7 Main steam lines

The analyses in References 2 and 3 are based on leakage that is less than the specified leakage rate. Leakage through all four MSIVs must be ≤ [100] scfh when tested at P_a ([11.5] psig). The MSIV leakage rate must be verified to be in accordance with the leakage test requirements of Reference 4, 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 (Ref. 4), as modified by approved

2 400

25.0

any one mainsteam line must be ≤100 scfh and through

7

8

7 the Primary Containment Leakage Rate Testing Program

(continued)

The acceptance criteria for the combined leakage of all hydrostatically tested lines is 1 gpm times the total number of hydrostatically tested PCIVs when tested at ≥ 1.1 P_a. 7

BASES

SURVEILLANCE REQUIREMENTS

SR ~~3.6.1.3.10~~ (continued)

~~exemptions; thus, SR 3.0.2 (which allows Frequency extensions) does not apply.~~ 7

SR 3.6.1.3.11

Surveillance of hydrostatically tested lines provides assurance that the calculation assumptions of References ~~2~~ 2 ~~and 3~~ are met. The combined leakage rates must be demonstrated ~~to be~~ in accordance with the leakage test ~~frequency of Reference 4, as modified by approved exemptions; thus SR 3.0.2 (which allows Frequency extensions) does not apply.~~ required by the Primary Containment Leakage Rate Testing Program.

1 7 8

TSTF-30 Changes not shown

~~[This SR is 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.]~~

~~[SR 3.6.1.3.12~~

~~[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, or 3] and having blocking devices on the valves that are not permanently installed.]~~

~~[Verifying that 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 time limits assumed in the analyses of References 2 and 3.~~

~~[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 inside primary containment occurs in these MODES, 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]~~ 7

(continued)

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.6.1.3.12 (continued)~~

~~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.~~

7

REFERENCES

1. Technical Requirements Manual. 1

2. ~~1.~~ UFSAR, ~~Chapter~~ Section ~~[15]~~ 16.5. 1 2

3. ~~2.~~ UFSAR, Section ~~[6.2]~~ 15.6.4.

4. ~~3.~~ UFSAR, ~~[Table 6.2-44]~~ Section 15.2.4.

5. UFSAR, Section 6.2.4.2.3.

~~4. 10 CFR 50, Appendix J.~~ 7

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, 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, or analysis description.

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

3. This change was approved to be made in NUREG-1434, Rev. 1 per change package BWR-15, C.9, but apparently was not made. This change was made to the BWR/4 ISTS, NUREG-1433, Rev. 1.

4. Typographical/grammatical error corrected.

5. 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 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.

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

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.

9. This change was approved to be made in NUREG-1434, Rev. 1 per change package BWR-16, C.23, Rev. 1, but apparently was not made. This change was made to the BWR/4 ISTS, NUREG-1433, Rev. 1.

10. This change was approved to be made in NUREG-1434, Rev. 1 per change package BWR-15, C.5, but apparently was not made. A similar change was made to NUREG-1434, Rev. 1, Bases 3.6.4.2, Required Actions A.1 and A.2.

11. The LaSalle 1 and 2 design includes EFCVs and TIPs, similar to the BWR/4 design. Therefore, the Bases for Required Actions C.1 and C.2 has been modified and proposed Bases for SR 3.6.1.3.4, SR 3.6.1.3.8, and SR 3.6.1.3.9 have been added, consistent with the BWR/4 ISTS (NUREG-1433, Rev. 1).

12. This change was approved to be made in NUREG-1434, Rev. 1 per change package BWR-15, C.4, but apparently was not made. This change was made to the BWR/4 ISTS, NUREG-1433, Rev. 1.

13. Some of the Bases changes for TSTF-30 and TSTF-269 have not been adopted since the SRs/information is not applicable to LaSalle 1 and 2.

14. Editorial change made for enhanced clarity.

15. 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.

16. 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.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 ~~Primary Containment~~ Drywell and Suppression Chamber Pressure 1

BASES

BACKGROUND

The ~~primary containment~~ drywell and suppression chamber internal pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a Design Basis Accident (DBA) or loss of coolant accident (LOCA).

2 ~~The limits on primary containment [to secondary containment differential] pressure have been developed based on operating experience. The auxiliary building, which is part of the secondary containment, completely surrounds the lower portion of the primary containment. Therefore, the primary containment design external differential pressure, and consequently the Specification limit, are established relative to the auxiliary building pressure. The auxiliary building pressure is kept slightly negative relative to the atmospheric pressure to prevent leakage to the atmosphere.~~

Transient events, ~~which include inadvertent~~ ~~containment~~ drywell 2 spray initiation, can reduce ~~the primary containment~~ drywell and suppression chamber internal 2 pressure ~~(Ref. 1)~~. Without an appropriate limit on the ~~negative containment~~ minimum drywell and suppression chamber internal pressure, the design limit for negative ~~internal~~ Containment differential 2 pressure of ~~[3.0]~~ 5.0 3 psid could be exceeded. Therefore, the Specification pressure limits of ~~[-0.1 and 1.0 psid]~~ (-0.5 psig) 4 were established ~~(Ref. 2)~~ (Ref. 1) 4.

The limitation on the ~~primary [to secondary containment differential]~~ maximum drywell and suppression chamber internal pressure provides added assurance that the peak LOCA ~~primary containment~~ drywell and suppression chamber 2 pressure does not exceed the design value of ~~15~~ 45 2 psig (Ref. 1). (0.15 psig) 4

APPLICABLE SAFETY ANALYSES

~~Primary containment performance for the DBA is evaluated for the entire spectrum of break sizes for postulated LOCAs inside containment (Ref. 3).~~ 2 (Ref. 2) Among the inputs to the design basis analysis is the initial ~~primary containment~~ internal pressure. ~~The primary containment [to secondary containment differential] pressure can affect the initial containment internal pressure.~~ 2 The initial pressure limitation requirements ensure that peak primary containment pressure for a DBA LOCA does not exceed the design value of ~~15~~ 45 psig and that peak negative pressure for an inadvertent

(continued)

Drywell and Suppression Chamber 1

BASES

drywell 2

APPLICABLE SAFETY ANALYSES (continued)

containment spray event does not exceed the design value of [-5.0] psid. 5.0 3

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

LCO 1

A limitation on the primary [to secondary containment differential] pressure of [≥ 0.0] and ≤ [5.0 psid] is required to ensure that primary containment initial conditions are consistent with the initial safety analyses assumptions so that containment pressures remain within design values during a LOCA and the design value of containment negative pressure is not exceeded during an inadvertent operation of containment sprays.

drywell and suppression chamber internal

+0.75 psig 3 -0.5 psig

drywell 2

APPLICABILITY

In MODES 1, 2, and 3, a DBA could result in 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 primary containment pressure within limits is not required in MODE 4 or 5.

1 drywell and suppression chamber internal

ACTIONS

A.1

drywell or suppression chamber internal 1

When primary [to secondary containment differential] pressure is not within the limits of the LCO, differential pressure must be restored to within limits 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 primary [to secondary containment differential] pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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 ~~primary containment [to secondary containment differential]~~ drywell and suppression chamber internal [1] pressure is within limits ensures that operation remains within the limits assumed in the primary containment analysis. The 12 hour Frequency of this SR was developed based on operating experience related to trending primary containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal primary containment pressure condition.

REFERENCES

[2] U

1. U FSAR, Section [6.2.1.~~2~~] 1.3. [3]
2. U FSAR, Section [6.2.1.1.~~3.2~~] 3.1.
3. ~~FSAR, Section [6.2].~~ [2]

1. Changes have been made to reflect the changes made to the Specification.

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

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

4. The last sentence in the third paragraph of the Background Section is describing both the upper and lower pressure limit, but it follows the description of the lower limit and comes before the description of the upper limit. For clarity, the lower limit value is identified in the description of the lower limit and the upper limit value is identified in the description of the upper limit. In addition, the statement specifies a Reference that is different than the Reference provided for the descriptions of the upper and lower limits. At LaSalle 1 and 2, the Reference for the actual limits is the same as the Reference for the descriptions of the limits. Therefore, the single Reference is identified at the end of each of the limit descriptions (lower limit and upper limit).

Drywell 1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.5 Primary Containment Air Temperature

BASES

BACKGROUND

2

Heat loads from the drywell, as well as piping and equipment ~~in the primary containment~~, add energy to the ~~primary containment~~ airspace and raise airspace temperature. Coolers included in the unit design remove this energy and maintain an appropriate average temperature ~~inside primary containment~~. The average airspace temperature affects the calculated response to postulated Design Basis Accidents (DBAs). This ~~primary containment~~ drywell air temperature limit is an initial condition input for the Reference 1 safety analyses.

APPLICABLE SAFETY ANALYSES

Primary containment performance for the DBA is evaluated for a entire spectrum of break sizes for postulated loss of coolant accidents (LOCAs) inside containment (Ref. 1). Among the inputs to the design basis analysis is the initial ~~primary containment~~ drywell 2 average air temperature. Analyses ~~assume an initial average~~ ~~primary containment~~ air ~~and suppression pool~~ 2 temperature of ~~95~~ 135 3 °F. Maintaining the expected initial conditions ensures that safety analyses remain valid and ensures that the peak LOCA primary ~~containment~~ drywell temperature does not exceed the maximum allowable temperature of ~~185~~ 340 2 °F (Ref. 1). Exceeding this design temperature may result in the degradation of the primary containment structure under accident loads. Equipment inside primary containment, and needed to mitigate the effects of a DBA, is designed to operate and be capable of operating under environmental conditions expected for the accident.

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

LCO

drywell 1

With an initial ~~primary containment~~ average air temperature less than or equal to the LCO temperature limit, the peak accident temperature is maintained below the ~~primary containment~~ design temperature. As a result, the ability of

(continued)

BASES

LCO (continued)

primary containment to perform its design function is ensured.

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 ~~primary containment~~ drywell 1 average air temperature within the limit is not required in MODE 4 or 5.

ACTIONS

A.1

When ~~primary containment~~ drywell 1 average air temperature is not within the limit of the LCO, it must be restored within 8 hours. This Required Action is necessary to return operation to within the bounds of the 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

If the ~~primary containment~~ drywell 1 average air temperature cannot be restored to within the 4 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

Verifying that the ~~primary containment~~ drywell 1 average air temperature is within the LCO limit ensures that operation remains within the limits assumed for the primary

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.5.1 (continued)

containment analyses. ~~In order to determine the primary containment average air temperature, an arithmetic average is calculated, using measurements taken at locations within the primary containment selected to provide a representative sample of the overall primary containment atmosphere.~~

The 24 hour Frequency of this SR ~~is considered acceptable based on observed slow rates of temperature increase within primary containment as a result of environmental heat sources (due to large volume of the primary containment).~~ Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal ~~primary containment~~ drywell 1 air temperature condition.

2
Insert
SR 3.6.1.5.1-1

REFERENCES

1. (u) 2 FSAR, Section ~~[~~6.2~~]~~. 3

Insert ITS B 3.6.1.6 (BWR/4 ISTS B 3.6.1.8) 5

The drywell average air temperature is determined using the average temperature at the operating return air plenum upstream of the primary containment ventilation heat exchanger coil and cabinet located at elevation 740ft 0 inches, azimuth 248°, and elevation 740 ft 0 inches, azimuth 76°. This provides a representative sample of the overall drywell atmosphere. 2

Insert SR 3.6.1.5.1-1

was developed based on operating experience related to drywell average air temperature variations and temperature dependent drift of instrumentation located in the drywell during the applicable MODES and the low probability of a DBA occurring between Surveillances.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.6.1.5 - DRYWELL AIR TEMPERATURE

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

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

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

4. Typographical/grammatical error corrected.

5. A new Bases has been added, ITS Bases 3.6.1.6. This Bases is from the BWR/4 ISTS (NUREG-1433 ISTS B 3.6.1.8), since the LaSalle 1 and 2 design is similar to the BWR/4 design with regard to the vacuum breakers. Therefore, the BWR/4 Bases are used and any deviations from the BWR/4 ISTS Bases are discussed in the Justification for Deviations for ITS Bases: 3.6.1.6.

Insert BWR/4 ISTS B 3.6.1.8* 1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.8 6 1 Suppression Chamber-to-Drywell Vacuum Breakers

BASES

outside the primary containment which form an extension of the primary containment boundary. The vacuum relief valves are mounted in special piping 3

BACKGROUND

The function 3 of the suppression-chamber-to-drywell vacuum breakers is to relieve vacuum in the drywell. There are (12) four 2 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 wetwell drywell boundary. Each vacuum breaker is a self actuating valve, ~~similar to a check valve, which can be remotely operated for testing purposes~~. with one vacuum breaker in each line 3

Manual isolation valves are located on each side of each vacuum breaker 3

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 3 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 ~~water leg~~ water column 3 in the Mark ~~I~~ II 3 Vent System downcomer is controlled by the drywell-to-suppression chamber differential pressure. If the drywell pressure is

(continued)

* This BWR/4 Bases insert was used to match the BWR/4 Specification inserted in the LCO section.

Insert Page B 3.6-38b

Insert BWR/4 ISTS B3.6.1.8 [1]
(continued)

downcomer water column height [3]

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~~ downcomer during normal operation. [3]

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~~ to maintain the structural integrity of primary containment. [3]

The safety analyses assume that the ~~internal~~ vacuum breakers are closed initially and ~~are~~ fully open at a differential pressure of ~~[0.5]~~ 1.0 [2] psid (~~Ref. 1~~ Refs. 1 and 2 [3]). Additionally, ~~3~~ one of the ~~12~~ four ~~internal~~ vacuum breakers ~~are~~ is assumed to fail in a closed position (~~Ref. 1~~). [3] 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]~~ four [2] vacuum breakers be OPERABLE, are a result of the requirement placed on the vacuum breakers to limit the ~~vent system~~ downcomer waterleg height (the additional vacuum breaker is required to meet the single failure criterion). [3] ~~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~~ assume the vacuum breakers to be closed initially and to remain closed and leak tight ~~with~~ until the suppression pool is at a positive pressure relative to the drywell. [4]

(continued)

Insert Page B 3.6-38c

Insert BWR/4 ISTS B 3.6.1.8 1
(continued)

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

5 A11 ~~Only [9] of the [12]~~ All vacuum breakers must be OPERABLE ~~for opening~~ to open. ~~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). This LCO also ensures that ~~The vacuum breaker OPERABILITY requirement~~ provides assurance that the drywell-to-suppression chamber negative differential pressure remains below the design value. Vacuum breakers will open so that The requirement that the vacuum breakers be closed ensures that there is no excessive bypass leakage should a LOCA occur.

3 The manual isolation valves in each vacuum breaker line must also be open for the associated vacuum breaker to be considered OPERABLE.

APPLICABILITY

6 ~~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.~~ drywell sprays

~~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 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. 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 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)

Insert Page B 3.6-38d

Insert BWR/4 ISTS B3.6.1.8 [1]
(continued)

BASES (continued)

ACTIONS

A.1

With one of the ~~required~~ [5] 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~~ three [2] 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~~ four [2] ~~required~~ vacuum breakers inoperable, 72 hours is allowed to restore ~~at least one of the~~ [5] inoperable vacuum breaker~~s~~ 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.

B.1 and B.2 [5]

[7] With one vacuum breaker not closed, exists [7]

~~An open vacuum breaker allows~~ communication between the drywell and suppression chamber airspace~~, and, as a result,~~ there is the potential for ~~suppression chamber~~ primary containment [3] overpressurization due to this bypass leakage if a LOCA were to occur. Therefore, ~~the open~~ both manual isolation valves in the affected line [5] vacuum breaker must be closed. A short time is allowed to close the ~~vacuum breaker~~ manual valves 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~~ 4 hour Completion Time is considered adequate to perform this ~~test~~ activity [5]. Insert B.2 [5]

C.1 and C.2

~~If the inoperable suppression chamber-to-drywell vacuum breaker cannot be closed or restored to OPERABLE status within the required Completion Time~~ If any Required Action and associated Completion Time cannot be met [8], the plant must be brought to a MODE in which the LCO does not apply. To

(continued)

Insert Page B 3.6-38e

Insert B.2

With both manual isolation valves closed, the vacuum breaker is not capable of performing the vacuum relief function. While the remaining three OPERABLE vacuum breakers are capable of providing the vacuum relief function, the overall 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, under this condition, 72 hours is allowed to restore the inoperable vacuum breaker to OPERABLE status so that the 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.

Insert BWR/4 B 3.6.1.8 1
(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.

5 Insert D.1

SURVEILLANCE REQUIREMENTS

SR 3.6.1.~~8~~.1 6 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]~~ 0.25 2 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.~~ 5

5 ~~A~~ Two Notes are added to this SR. The first Note ~~is added to this SR 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.

SR 3.6.1.~~8~~.2 1 6

Each ~~required~~ 5 vacuum breaker must be manually 3 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~~ 92 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~~ 5

(continued)

Insert Page B 3.6-38g

Insert D.1

With two or more vacuum breakers inoperable, an excessive suppression chamber-to-drywell differential pressure could occur during a DBA. Therefore, an immediate plant shutdown in accordance with LCO 3.0.3 is required.

Insert BWR/4 ISTS B3.6.1.8 1

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.~~7~~ 6 1.2 (continued)

~~additional assurance that the vacuum breakers are OPERABLE, since they are located in a harsh environment (the suppression chamber airspace).~~ 5 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~~. 5

SR 3.6.1.~~8~~ 6 1.3

Verification of the vacuum breaker opening setpoint of ≤ 0.5 psid from the closed position 3 is necessary to ensure that the safety analysis assumption regarding vacuum breaker full open differential pressure of ~~[0.5]~~ 1.0 2 psid is valid. 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. 8 (~~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.

REFERENCES

1. U 3 FSAR, Section ~~[6.2]~~ 6.2.1 2.

2. FSAR, Response to NRC Question 021.4. 3

Insert Page B 3.6-38i

1. A new Bases has been added, ITS Bases 3.6.1.6. This Bases is from BWR/4 ISTS 3.6.1.8 (NUREG-1433), since the LaSalle 1 and 2 design is similar to the BWR/4 design with regard to the vacuum breakers. Therefore, the BWR/4 Bases are used and any deviations from the BWR/4 ISTS are discussed below.

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, or analysis 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. Changes have been made to reflect those changes made to the Specification.

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. 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 3.6.1.6 Low-Low Set (LLS) Valves

BASES

BACKGROUND

The safety/relief valves (S/RVs) can actuate either in the relief mode, 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 overcome the spring force and open the pilot valve. As in the safety mode, opening the pilot valve allows a differential pressure to develop across the main valve piston and thus opens the main valve. The main valve can stay open with valve inlet steam pressure as low as [0] psig. 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.

[Six] 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, such that reopening of 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.

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.

APPLICABLE SAFETY ANALYSES

The LLS relief mode functions to ensure that the containment design basis of one S/RV operating on "subsequent actuations" is met (Ref. 1). 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 multiple simultaneous S/RV openings occur only on the initial actuation for DBAs. Even though [six]

(continued)

1

BASES

APPLICABLE SAFETY ANALYSES (continued)

LLS S/RVs are specified, all [six] LLS S/RVs do not operate in any DBA analysis.

LLS valves satisfy Criterion 3 of the NRC Policy Statement.

LCO

[Six] LLS valves are required to be OPERABLE to satisfy the assumptions of the safety analysis (Ref. 2). The requirements of this LCO are applicable to the mechanical and electrical/pneumatic capability of the LLS valves to function for controlling the opening and closing of the S/RVs.

APPLICABILITY

In MODES 1, 2, and 3, an event could cause pressurization of the reactor and opening of S/RVs. 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 valves OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

With one LLS valve inoperable, the remaining OPERABLE LLS valves are 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 S/RVs and the low probability of an event in which the remaining LLS S/RV capability would be inadequate.

B.1 and B.2

If two or more LLS valves are inoperable or if the inoperable LLS 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 allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1

A manual actuation of each LLS valve is performed to verify that the valve and solenoids are functioning properly and that 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 ≥ [950] 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 ADS 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 developed based on the S/RV tests required by the ASME Boiler and Pressure Vessel Code, Section XI (Ref. 3). 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 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.

Since steam pressure is required in order to perform the Surveillance, however, and steam may not be available during a unit outage, the Surveillance may be performed during the shutdown prior to or the startup following a unit outage. Unit startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified by Reference 3 prior to valve installation. After adequate reactor steam dome pressure and flow are reached, 12 hours are allowed to prepare for and perform the test.

(continued)

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.6.2

The LLS designed S/RVs 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 automatic LLS function operate as designed when initiated either by an actual or simulated automatic initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.5.4 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. 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.

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

REFERENCES

[1. GESSAR-II, Appendix 3BA.8.]

2. FSAR, Section [5.5.17].

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

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ISTS BASES: 3.6.1.6 - LOW-LOW SET (LLS) VALVES

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

1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.7 Residual Heat Removal (RHR) Containment Spray System

BASES

BACKGROUND

The primary containment is designed with a suppression pool so that, in the event of a loss of coolant accident (LOCA), steam released from the primary system is channeled through the suppression pool water and condensed without producing significant pressurization of the primary containment. The primary containment is designed so that with the pool initially at the minimum water volume and the worst single failure of the primary containment heat removal systems, suppression pool energy absorption combined with subsequent operator controlled pool cooling will prevent the primary containment pressure from exceeding its design value. However, the primary containment must also withstand a postulated bypass leakage pathway that allows the passage of steam from the drywell directly into the primary containment airspace, bypassing the suppression pool. The primary containment also must withstand a low energy steam release into the primary containment airspace. The RHR Containment Spray System is designed to mitigate the effects of bypass leakage and low energy line breaks.

There are two redundant, 100% capacity RHR containment spray subsystems. Each subsystem consists of a suction line from the suppression pool, an RHR pump, a heat exchanger, and three spray spargers inside the primary containment (outside of the drywell) above the refueling floor. Dispersion of the spray water is accomplished by 350 nozzles in each subsystem.

The RHR containment spray mode will be automatically initiated, if required, following a LOCA, or it may be manually initiated according to emergency procedures.

APPLICABLE SAFETY ANALYSES

Reference 1 contains the results of analyses that predict the primary containment pressure response for a LOCA with the maximum allowable bypass leakage area.

The equivalent flow path area for bypass leakage has been specified to be [0.9] ft^2. The analysis demonstrates that

(continued)

1

BASES

APPLICABLE SAFETY ANALYSES (continued)

with containment spray operation the primary containment pressure remains within design limits.

The RHR Containment Spray System satisfies Criterion 3 of the NRC Policy Statement.

LCO

In the event of a Design Basis Accident (DBA), a minimum of one RHR containment spray subsystem is required to mitigate potential bypass leakage paths and maintain the primary containment peak pressure below design limits. To ensure that these requirements are met, two RHR containment spray subsystems must be OPERABLE. Therefore, in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active failure. An RHR containment spray subsystem is OPERABLE when the pump, 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 containment spray subsystems OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

With one RHR containment spray subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE RHR containment spray 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 was chosen in light of the redundant RHR containment capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

(continued)

[1]

BASES

ACTIONS (continued)

B.1

With two RHR containment spray 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 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.

C.1 and C.2

If the inoperable RHR containment spray 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 3.6.1.7.1

Verifying the correct alignment for manual, power operated, and automatic valves in the RHR containment 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 were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency of this SR is justified because the valves are operated under procedural control and because improper valve position would affect only a single subsystem. This Frequency has been shown to be acceptable based on operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.7.1 (continued)

A Note has been added to this SR that allows RHR containment spray subsystems to be considered OPERABLE during alignment to and operation in the RHR shutdown cooling mode when below [the RHR cut in permissive pressure in MODE 3], if capable of being manually realigned and not otherwise inoperable. At these low pressures and decay heat levels (the reactor is shut down in MODE 3), a reduced complement of subsystems can provide the required containment pressure mitigation function thereby allowing operation of an RHR shutdown cooling loop when necessary.

SR 3.6.1.7.2

Verifying each RHR pump develops a flow rate ≥ [5650] 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. It is tested in the pool cooling mode to demonstrate pump OPERABILITY without spraying down equipment in primary containment. Flow is a normal test of centrifugal pump performance required by the ASME Code, Section XI (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 or 92 days.]

SR 3.6.1.7.3

This SR verifies that each RHR containment spray subsystem automatic valve actuates to its correct position upon receipt of an actual or simulated automatic actuation signal. Actual spray initiation is not required to meet this SR. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.3.6 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. Operating experience has shown that these components usually pass the Surveillance when performed at

(continued)

[1]

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.7.3 (continued)

the [18] month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.7.4

This Surveillance is performed every 10 years to verify that the spray nozzles are not obstructed and that flow will be provided when required. The 10 year Frequency is adequate to detect degradation in performance due to the passive nozzle design and its normally dry state and has been shown to be acceptable through operating experience.

REFERENCES

1. FSAR, Section [6.2.1.1.5].

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

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

1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.8 Penetration Valve Leakage Control System (PVLCS)

BASES

BACKGROUND

The PVLCS supplements the isolation function of primary containment isolation valves (PCIVs) in process lines that also penetrate the secondary containment. These penetrations are sealed by air from the PVLCS to prevent fission products leaking past the isolation valves and bypassing the secondary containment after a Design Basis Accident (DBA) loss of coolant accident (LOCA).

The PVLCS consists of [two] independent, manually initiated subsystems, either of which is capable of preventing fission product leakage from the containment post LOCA. Each subsystem is comprised of an air compressor, an accumulator, an injection valve, and three injection headers with separate isolation valves. This system has additional headers, which serve the Main Steam Isolation Valve Leakage Control System and safety/relief valve (S/RV) actuator air accumulators.

Each process line has two PCIVs and an additional manual isolation valve outside of the outboard PCIV. The two outboard valves are double disk gate valves. Each valve is provided sealing air from its electrically associated division of PVLCS to the area between the dual disk seats. The PVLCS is started manually.

APPLICABLE SAFETY ANALYSES

The analyses described in Reference 1 provide the evaluation of offsite dose consequences during accident conditions. During the first 25 minutes following an accident, the isolation valves on lines that penetrate primary containment and also penetrate secondary containment are assumed to leak fission products directly to the environment, without being processed by the Standby Gas Treatment System. The analyses take credit for manually initiating PVLCS after 25 minutes and do not assume any further secondary containment bypass leakage.

The PVLCS satisfies Criterion 3 of the NRC Policy Statement.

(continued)

1

BASES (continued)

LCO

[Two] PVLCS subsystems must be OPERABLE such that in the event of an accident, at least one subsystem is OPERABLE assuming the worst case single active failure. A PVLCS subsystem is OPERABLE when all necessary components are available to supply each associated dual seat isolation valve with sufficient air pressure to preclude containment leakage when the containment atmosphere is at the maximum peak containment pressure, P_a.

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 PVLCS is not required to be OPERABLE in MODES 4 and 5 to prevent leakage of radioactive material from primary containment.

ACTIONS

A.1

With one PVLCS subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE PVLCS subsystem is adequate to perform the leakage control function. The 30 day Completion Time is based on the low probability of the occurrence of a LOCA, the amount of time available after the event for operator action to prevent exceeding this limit, the low probability of failure of the OPERABLE PVLCS subsystem, and the availability of the PCIVs.

B.1

With [two] PVLCS 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, the availability of 25 minutes for operator action, and the availability of the PCIVs.

(continued)

1

BASES

ACTIONS (continued)

C.1 and C.2

If the inoperable PVLCS 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 3.6.1.8.1

The minimum air supply necessary for PVLCS OPERABILITY varies with the system being supplied with compressed air from the PVLCS accumulators. Due to the support system function of PVLCS for S/RV actuator air, however, the specified minimum pressure of [101] psig is required, which provides sufficient air for [ ] S/RV actuations with the drywell pressure at 30 psig. This minimum air pressure alone is sufficient for PVLCS to support the OPERABILITY of these S/RV systems and is verified every 24 hours. The 24 hour Frequency is considered adequate in view of other indications available in the control room, such as alarms, to alert the operator to an abnormal PVLCS air pressure condition.

SR 3.6.1.8.2

A simulated system operation is performed every [18] months to ensure that the PVLCS will function throughout its operating sequence. This includes correct automatic positioning of valves once the system is initiated manually. Proper functioning of the compressor and valves is verified by this Surveillance. The [18] month Frequency was developed considering it is prudent that many Surveillances be performed only during a plant outage. 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.

(continued)

BASES (continued)

~~REFERENCES 1. FSAR, Section [15.6.5].~~

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ISTS BASES: 3.6.1.8 - PENETRATION VALVE LEAKAGE CONTROL SYSTEM (PVLCS)

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

B 3.6 CONTAINMENT SYSTEMS

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

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, which is connected between the inboard and outboard MSIVs; and an outboard subsystem, which is 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 (four blowers for the inboard subsystem and two blowers for the outboard subsystem), valves, piping, and heaters (for the inboard subsystem only). The 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 that leakage through the closed MSIVs is collected by the MSIV LCS. In both process modes, the effluent is discharged to the auxiliary building, which encloses a volume served 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 auxiliary building and ultimately filtered by the SGT System. The analyses in Reference 3 provide the evaluation of offsite dose consequences. The operation of the MSIV LCS prevents a release of untreated leakage for this type of event.

(continued)

1

1

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)

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 subsystem occurring between surveillances and has been shown to be acceptable through operating experience.

SR 3.6.1.9.2

The electrical continuity of each inboard MSIV LCS subsystem heater is verified by a resistance check, by verifying the rate of temperature increase meets specifications, or by verifying 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 the upstream heaters meet current or wattage draw requirements (if not used to verify electrical continuity in SR 3.6.1.9.2). The 18 month

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.9.3 (continued)

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.

REFERENCES

1. FSAR, Section [6.7].
2. Regulatory Guide 1.96, Revision [1].
3. FSAR, Section [15.6.5].

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, 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

Primary containment utilizes a Mark II over/under pressure suppression configuration, with the

BASES

BACKGROUND

1

Insert B 3.6.2.1 Background

The suppression pool ~~is a concentric open container of water with a stainless steel liner that is~~ located at the bottom of the primary containment. The suppression pool is designed to absorb the decay heat and sensible heat released during a reactor blowdown from safety/relief valve discharges or from a loss of coolant accident (LOCA). The suppression pool must also condense steam from the Reactor Core Isolation Cooling System turbine exhaust and provides the main emergency water supply source for the reactor vessel. ~~The amount of energy that the pool can absorb as it condenses steam is dependent upon the initial average suppression pool temperature. The lower the initial pool temperature, the more heat it can absorb without heating up excessively. Since it is an open pool, its temperature will affect both primary containment pressure and average air temperature. Using conservative inputs and methods, the maximum calculated primary containment pressure during and following a Design Basis Accident (DBA) must remain below the primary containment design pressure of [15] psig. In addition, the maximum primary containment average air temperature must remain < [185]°F.~~

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

2

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 ~~[— the design pressure is [15] psig and design temperature is [185]°F]~~;

c. Condensation oscillation (CO) loads ~~[— a maximum allowable initial temperature of [100]°F ensures that CO loads do not exceed the Mark III CO load definition]~~; and

d. Chugging loads ~~[— a maximum allowable initial temperature of [100]°F ensures that expected LOCA~~

(continued)

Insert B 3.6.2.1 BACKGROUND

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 design value (45 psig). 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.

BASES

BACKGROUND (continued)

~~temperatures are within the range of Mark III tested conditions]~~. 2

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 suppression pool temperature analyses required by Reference 3). An initial pool temperature of ~~[95]~~ 105 °F is assumed for the Reference 1 ~~and 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 plant testing.~~ 1

5 1 and 1 and 2 1 3

Suppression pool average temperature satisfies Criteria 2 and 3 of ~~the NRC Policy Statement~~ 10CFR50.36(c)(2)(ii). 1

LCO

A limitation on the suppression pool average temperature is required to assure that the primary 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 as follows:

TSTF-206 Changes not shown

a. ~~Average temperature ≤ [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.~~ 4 ~~This requirement ensures that licensing bases initial conditions are met.~~ 1

TSTF-206

4 a ~~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] 3 and testing that adds heat to the suppression pool is being performed. This requirement ensures that the plant has testing flexibility, and was also 4

4

(continued)

BASES

LCO (continued)

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

3 4 TSTF-206

6 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 plant 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.

4 {with THERMAL POWER ≤1% RTP} 3

[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~~, heat input is approximately equal to normal system heat losses.

TSTF-206 {1% RTP}

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, and A.3 4

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, the initial conditions exceed the conditions assumed for the Reference 1 and 2 analyses. However, primary containment cooling capability still exists, and the primary containment pressure suppression function will occur at temperatures well above that 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 temperature to be restored to below the limit. Additionally, when pool temperature is > [95]°F,

4 limit 4 2 1 105 3

(continued)

BASES

ACTIONS

~~A.1 and A.2~~ and A.3 (continued) 4

3

increased monitoring of the pool temperature is required to ensure it remains ≤ ~~[~~110~~]~~°F. The once per hour Completion Time is adequate based on past experience, which has shown that suppression pool temperature increases relatively slowly except when testing that adds heat to the 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.

4 In addition,

B.1

~~[~~1% RTP~~]~~ 3 TSTF-206

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, THERMAL POWER must be reduced to ≤ ~~[25/40] divisions of full scale on Range 7 for all OPERABLE IRM channels~~ within 12 hours. The 12 hour Completion Time is reasonable, based on operating experience, to reduce reactor power from full power in an orderly manner and without challenging plant systems. 3

5

~~C.1~~

TSTF-206 Changes not Shown

~~Suppression pool average temperature is allowed to be > [95]°F with any OPERABLE IRM channel > [25/40] divisions of full scale on Range 7 when testing that adds heat to the suppression pool is being performed. However, if temperature is > [105]°F, the~~ testing must be immediately ~~suspended to preserve the pool's~~ heat absorption capability. ~~With the testing suspended, Condition A is entered and the Required Actions and associated Completion Times are applicable.~~ 4

that adds heat to the suppression pool

4

~~D.1 and D.2~~ C.1, C.2 and C.3

3

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

within 36 hours 6

(continued)

BASES

4

ACTIONS

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

rates (provided pool temperature remains ≤ 120°F). Additionally, when 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 experience. Given the high pool temperature in this ~~C~~ondition, 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. 3 5 1

~~E.1 and E.2~~ D.1 and D.2 4

If suppression pool average temperature cannot be maintained ≤ 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 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 without challenging plant systems. 3

Continued addition of heat to the suppression pool with 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 temperature was > 120°F, the maximum allowable bulk and local temperatures could be exceeded very quickly. 3

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1.1

The suppression pool average temperature is regularly monitored to ensure that the required limits are satisfied. Average temperature is determined by taking an arithmetic average of the OPERABLE suppression pool water temperature channels, and may include an allowance for temperature stratification 1. The 24 hour Frequency has been shown to be acceptable based on operating experience. When heat is being added to the suppression pool by testing, however, it

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1.1 (continued)

is necessary to monitor suppression pool temperature more frequently. The 5 minute Frequency during testing is justified by the rates at which testing will heat up the suppression pool, has been shown to be acceptable based on 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

1 U

1. FSAR, Section ~~[6.2]~~. 3
2. ~~FSAR, Section [15.2].~~ LaSalle County Station Mark II Design Assessment Report, Section 6.2, June 1981. 1
3. NUREG-0783.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.6.2.1 - SUPPRESSION POOL AVERAGE TEMPERATURE

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

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

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

5. Typographical error corrected.

6. Editorial change made for enhanced clarity.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.2 Suppression Pool Water Level

primary containment utilizes a Mark II over/under pressure suppression configuration, with the

1

BASES

BACKGROUND

The suppression pool ~~is a concentric open container of water with a stainless steel liner, which is located~~ at the bottom of the primary containment. The suppression pool is designed to absorb the decay heat and sensible heat released during a reactor blowdown from safety/relief valve (S/RV) discharges or from a loss of coolant accident (LOCA). The suppression pool must also condense steam from the Reactor Core Isolation Cooling (RCIC) System turbine exhaust and provides the main emergency water supply source for the reactor vessel. The suppression pool volume ranges between 128,800 ~~135,291~~ ft^3 at the low water level limit of ~~18 ft~~ — 4.5 inches and 131,900 ~~138,701~~ ft^3 at the high water level limit of ~~18 ft 9.75~~ inches. The level is referenced to a plant elevation of 699 ft 11 inches.

2 3 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 S/RV quenchers, main vents, or 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.

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 resulting from a Design Basis Accident (DBA) LOCA. ~~An inadvertent upper pool dump could also overflow the weir wall into the drywell.~~ 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. 1

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

1

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

within the limits specified so that the safety analysis of Reference 1 remains valid.

Suppression pool water level satisfies Criteria 2 and 3 of ~~the NRC Policy Statement~~. 10 CFR 50.36(c)(2)(ii) 1

LCO

A limit that suppression pool 3 water level be ≥ (18 ft) (–) 4.5 inches) and ≤ (18 ft 9.75) inches) is required to ensure that the primary containment conditions assumed for the safety analysis are met. Either the high or low water level limits were used in the safety analysis, depending upon which is conservative for a particular calculation. 2 1 (referenced to plant elevation 699 ft 11 inches)

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause significant loads on the primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced because of 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

With suppression pool water level outside the limits, the conditions assumed for the safety analysis are not met. If water level is below the minimum level, the pressure suppression function still exists as long as ~~main vents~~ 1 the downcomers are covered, 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 ~~or as long as the drywell~~ 3 and the capability of the 1 suppression pool 3 sprays ~~are OPERABLE~~. ~~Prompt action to restore the suppression pool water level to within the normal range is prudent, however, to retain the margin to weir wall overflow from an inadvertent upper pool dump and reduce the risks of increased pool swell and dynamic loading.~~ 1 Therefore, continued operation for a limited time is allowed. The 2 hour Completion Time is sufficient to restore suppression pool water level to within specified limits. Also, it takes

(continued)

BASES

ACTIONS

A.1 (continued)

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.

1 has been shown to be acceptable based on operating experience.

REFERENCES

1. U 1 ~~F~~SAR, Section ~~[~~6.2~~]~~. 2

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.6.2.2 - SUPPRESSION POOL WATER LEVEL

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

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

3. Editorial change made for enhanced clarity.

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.

1 Each RHR subsystem contains a pump and ~~two~~ a heat exchanger~~s~~ ~~in series~~ and is manually initiated and independently controlled. The two RHR subsystems perform the suppression pool cooling function by circulating water from the suppression pool through the RHR heat exchanger~~s~~ 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 subsystem is sufficient to meet the overall DBA pool cooling requirement to limit peak temperature to ~~185~~ 200 °F for loss of coolant accidents (LOCAs) and transient events such as a turbine trip or a stuck open safety/relief valve (S/RV). S/RV leakage and Reactor Core Isolation Cooling System 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. 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~~ 10CFR50.36(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 the 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~~. 3 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 the pump, ~~two~~ a heat ~~exchangers~~ exchanger, and associated piping, valves, instrumentation, and controls are OPERABLE. 1

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. 4 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 RHR suppression pool cooling subsystem is adequate to perform the primary containment cooling function. 6 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

(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

5

any

associated

TSTF-230

B.1 and B.2

If ~~the~~ Required Action and ~~required~~ Completion Time ~~of~~ 5 ~~Condition A~~ cannot be met ~~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.

or

5.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.3.1

5

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 being locked, sealed, or secured. 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.

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
4 ~~sub~~system is a manually initiated system. This Frequency

(continued)

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 5

7200 2

Verifying each RHR pump develops a flow rate ≥ 7450 gpm, while operating in the suppression pool cooling mode with flow through the associated heat exchanger ~~at least every 92 days~~, ensures that ~~pump performance has not degraded during the cycle.~~ Flow is a normal test of centrifugal pump performance required by ASME 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 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 ~~or 92 days~~.

5 1 also

verification 4

7

2

peak suppression pool temperature can be maintained below the design limits during a DBA (Ref. 1). The 1

6 tests

REFERENCES

1. (U) 1 FSAR, Section ~~X~~6.2~~X~~. 2

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

Insert ITS B 3.6.2.4 (BWR/4 ISTS B3.6.2.4)

8

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, 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, or analysis description.

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

3. The specific requirement for the subsystems to be powered from two safety related independent power supplies has been deleted since the design of the system already reflects this. There are only two subsystems, and each is powered from a separate power supply; the power supplies cannot be cross-connected. This statement is not used in other LCO Bases where the system is designed with independent power supplies (e.g., Bases 3.6.1.6, "RHR Drywell Spray," and Bases 3.6.3.1, "Primary Containment Hydrogen Recombiners"). The BWR/4 ISTS Bases has this statement since some BWR/4s have two pumps per subsystem, with only one required for the subsystem to be Operable (as described in the BWR/4 ISTS Bases), and due to the electrical design of the system, one pump in each subsystem is powered from the same electrical division. Thus, for this design, the words in the NUREG are necessary. However, as described above, LaSalle 1 and 2 does not have this design.

4. Editorial change made for enhanced clarity.

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

6. Typographical/grammatical error corrected.

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

8. A new Bases has been added, ITS Bases 3.6.2.4. This Bases is from the BWR/4 ISTS (NUREG-1433 ISTS B 3.6.2.4), since the LaSalle 1 and 2 design is similar to the BWR/4 design with regard to RHR suppression pool spray. Therefore, the BWR/4 Bases is used and any deviations from the BWR/4 ISTS Bases are discussed in the Justification for Deviations for ITS Bases: 3.6.2.4.

Insert BWR/4 ISTS B3.6.2.4* [1]

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.

Each of the ~~two RHR~~ one [2] suppression pool spray subsystems contains two pump~~s~~ 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~~s~~ [2] (provided the associated valve is open). Thus, both suppression pool cooling and suppression pool spray functions are normally [2] 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.~~ [3] 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.

(continued)

*This BWR/4 Specification was used to match the BWR/4 Specification inserted in the LCO Section

Insert Page B3.6-68a

Insert BWR/4 ISTS B 3.6.2.4 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) 2

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~~. 5 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 6 RHR suppression pool spray subsystem is adequate to perform the primary containment bypass leakage mitigation function.

(continued)

Insert Page B 3.6-68b

Insert BWR/4 ISTS B3.6.2.4 1
(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 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~~ reduce pressure in the primary containment are available. 6 2

C.1 and C.2

If ~~the inoperable RHR suppression pool spray subsystem cannot be restored to OPERABLE status within the~~ any Required Action and associated Completion Time 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 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. 6

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 7

(continued)

Insert Page B 3.6-68c

Insert BWR/4 ISTS B 3.6.2.4 [1]
(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 [9] 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.

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 subsystem is a manually initiated system. This Frequency has been shown to be acceptable based on operating experience.

SR 3.6.2.4.2

Verifying each required [7] RHR pump develops a flow rate ≥ ~~[400]~~ 450 [8] gpm through the spray sparger [3] while operating in the suppression pool spray mode ~~with flow through the heat exchanger~~ [3] ~~ensures that pump performance has not degraded during the cycle. Flow is a~~ helps ensure that the primary containment pressure can be maintained below the design limits during a DBA (Ref. 1). The [2] ~~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.~~ [2] The Frequency of this SR is ~~in accordance with the Inservice Testing Program~~ is covered by the requirements of LCO 3.6.2.3, "RHR Suppression Pool Cooling." [2] ~~but the Frequency must not exceed 92 days~~ [8].

REFERENCES [2]

1. (U) FSAR, Section ~~[~~6.2~~]~~.1.1.3 [8].
2. ASME, Boiler and Pressure Vessel Code, Section XI.

Insert Page B 3.6-68d

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.6.2.4 - RHR SUPPRESSION POOL SPRAY

1. A new Bases has been added, ITS Bases 3.6.2.4. This Bases is from the BWR/4 ISTS (NUREG-1433 ISTS B 3.6.2.4), since the LaSalle 1 and 2 design is similar to the BWR/4 design with regard to RHR suppression pool spray. Therefore, the BWR/4 Bases is used and any deviations from the BWR/4 ISTS Bases are discussed below.

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

3. The RHR suppression pool spray mode does not credit the cooling effect of the heat exchangers, therefore, the requirement to demonstrate flow through the heat exchangers has been deleted. Additionally, clarification of flow demonstration through the spray sparger is provided.

4. Not used.

5. The specific requirement for the subsystems to be powered from two safety related independent power supplies has been deleted since the design of the system already reflects this. There are only two subsystems, and each is powered from a separate power supply; the power supplies cannot be cross-connected. This statement is not used in other LCO Bases where the system is designed with independent power supplies (e.g., Bases 3.6.3.1, "Primary Containment Hydrogen Recombiners"). The BWR/4 ISTS Bases has this statement since some BWR/4s have two pumps per subsystem, with only one required for the subsystem to be Operable (as described in the BWR/4 ISTS Bases), and due to the electrical design of the system, one pump in each subsystem is powered from the same electrical division. Thus, for this design, the words in the NUREG are necessary. However, as described above, LaSalle 1 and 2 does not have this design.

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

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

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

9. Editorial change made to be consistent with context of the Specification.

1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2.4 Suppression Pool Makeup (SPMU) System

BASES

BACKGROUND

The function of the SPMU System is to transfer water from the upper containment pool to the suppression pool after a loss of coolant accident (LOCA). For a LOCA, with Emergency Core Cooling System injection from the suppression pool, a large volume of water can be held up in the drywell behind the weir wall. This holdup can significantly lower suppression pool water level. The water transfer from the SPMU System ensures a post LOCA suppression pool vent coverage of ≥ 2 ft above the top of the top row vents so that long term steam condensation is maintained. The additional makeup water is used as part of the long term suppression pool heat sink. The post LOCA delayed transfer of this water to the suppression pool provides an initially low vent submergence, which results in lower drywell pressure loading and lower pool dynamic loading during a Design Basis Accident (DBA) LOCA as compared to higher vent submergence. The sizing of the residual heat removal heat exchanger takes credit for the additional SPMU System water mass in the calculation of the post LOCA peak containment pressure and suppression pool temperature.

The required water dump volume from the upper containment pool is equal to the difference between the total post LOCA drawdown volume and the assumed volume loss from the suppression pool. The total drawdown volume is the volume of suppression pool water that can be entrapped outside of the suppression pool following a LOCA. The post LOCA entrapment volumes causing suppression pool level drawdown include:

a. The free volume inside and below the top of the drywell weir wall;

b. The added volume required to fill the reactor pressure vessel from a condition of normal power operation to a post accident complete fill of the vessel, including the top dome;

c. The volume in the steam lines out to the inboard main steam isolation valve (MSIV) on three lines and out to the outboard MSIV on one line; and

(continued)

BASES

BACKGROUND (continued)

d. Allowances for primary containment spray holdup on equipment and structural surfaces.

The SPMU System consists of two redundant subsystems, each capable of dumping the makeup volume from the upper containment pool to the suppression pool by gravity flow. Each dump line includes two normally closed valves in series. The upper pool is dumped automatically on a suppression pool water level Low-Low signal (with a LOCA signal permissive) or on the basis of a timer following a LOCA signal alone to ensure that the makeup volume is available as part of the long term energy sink for small breaks that might not cause dump on a suppression pool water level Low-Low signal. A 30 minute timer was chosen, since the initial suppression pool mass is adequate for any sequence of vessel blowdown energy and decay heat up to at least 30 minutes.

Although the minimum freeboard distance above the suppression pool high water level limit of LCO 3.6.2.2, "Suppression Pool Water Level," to the top of the weir wall is adequate to preclude flooding of the drywell, a LOCA permissive signal is used to prevent an erroneous suppression pool level signal from causing pool dump. In addition, the SPMU System mode switch may be keylocked in the "OFF" position to ensure that inadvertent dump will not occur. Inadvertent actuation of the SPMU System during MODE 4 or 5 could create a radiation hazard to plant personnel due to a loss of shield water from the upper pool if irradiated fuel were in an elevated position.

APPLICABLE SAFETY ANALYSES

Analyses used to predict suppression pool temperature following large and small break LOCAs, which are the applicable DBAs for the SPMU System, are contained in References 1 and 2. During these events, the SPMU System is relied upon to dump upper containment pool water to maintain drywell horizontal vent coverage and an adequate suppression pool heat sink volume to ensure that the primary containment internal pressure and temperature stay within design limits. The analysis assumes an SPMU System dump volume of [36,380] ft^3 at a temperature of [125]°F.

The SPMU System satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

During a DBA, a minimum of one SPMU subsystem is required to maintain peak suppression pool water temperature below the design limits (Ref. 1). To ensure that these requirements are met, two SPMU subsystems must be OPERABLE with power from two independent safety related power supplies. Therefore, in the event of an accident, at least one subsystem is OPERABLE, assuming the worst case single active failure. The SPMU System is OPERABLE when the upper containment pool water temperature is ≤ [125]°F, the water level is ≥ [23 ft 3 inches], gates are in the stored condition, the piping is intact, and the system valves are OPERABLE. The above temperature and water level conditions correspond to an SPMU System available dump volume of ≥ [36,380] ft^3.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause heatup and pressurization of 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. Therefore, maintaining the SPMU System OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

When upper containment pool water level is < [23 ft 3 inches], the volume is inadequate to ensure that the suppression pool heat sink capability matches the safety analysis assumptions. A sufficient quantity of water is necessary to ensure long term energy sink capabilities of the suppression pool and maintain water coverage over the uppermost drywell vents. Loss of water volume has a relatively large impact on heat sink capability. Therefore, the upper containment pool water level must be restored to within limit within 4 hours. The 4 hour Completion Time is sufficient to provide makeup water to the upper containment pool to restore level within specified limit. Also, it takes into account the low probability of an event occurring that would require the SPMU System.

(continued)

BASES

ACTIONS (continued)

B.1

When upper containment pool water temperature is > [125]°F, the heat absorption capacity is inadequate to ensure that the suppression pool heat sink capability matches the safety analysis assumptions. Increased temperature has a relatively smaller impact on heat sink capability. Therefore, the upper containment pool water temperature must be restored to within limit within 24 hours. The 24 hour Completion Time is sufficient to restore the upper containment pool to within the specified temperature limit. It also takes into account the low probability of an event occurring that would require the SPMU System.

C.1

With one SPMU subsystem inoperable for reasons other than Condition A or B, the inoperable subsystem must be restored to OPERABLE status within 7 days. The 7 day Completion Time is acceptable in light of the redundant SPMU System capabilities afforded by the OPERABLE subsystem and the low probability of a DBA occurring during this period.

D.1 and D.2

If any Required Action and required 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.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.4.1

The upper containment pool water level is regularly monitored to ensure that the required limits are satisfied. The 24 hour Frequency of this SR was developed, considering operating experience related to upper containment pool water level variations and water level instrument drift during the applicable MODES and considering the low probability of a

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.4.1 (continued)

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 upper containment pool water level condition.

SR 3.6.2.4.2

The upper containment pool water temperature is regularly monitored to ensure that the required limit is satisfied. The 24 hour Frequency was developed, based on operating experience related to upper containment pool temperature variations during the applicable MODES.

SR 3.6.2.4.3

Verifying the correct alignment for manual, power operated, and automatic valves in the SPMU System 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 are verified to be in the correct position prior to being locked, sealed, or secured. This SR does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The Frequency of 31 days is justified because the valves are operated under procedural control and because improper valve position would affect only a single subsystem. This Frequency has been shown to be acceptable through operating experience.

SR 3.6.2.4.4

The upper containment pool has two gates used to separate the pool into distinct sections to facilitate fuel transfer and maintenance during refueling operations and two additional gates in the separator pool weir wall extension,

(continued)

1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.4.4 (continued)

which, when installed, limit personnel exposure and ensure adequate water submergence of the separator when the separator is stored in the pool. The SPMU System dump line penetrations are located in the steam separator storage section of the pool. To provide the required SPMU System dump volume to the suppression pool, the gates must be removed (or placed in their stored position) to allow communication between the various pool sections. The 31 day Frequency is appropriate because the gates are moved under procedural control and only the infrequent movement of these gates is required in MODES 1, 2, and 3.

SR 3.6.2.4.5

This SR requires a verification that each SPMU subsystem automatic valve actuates to its correct position on receipt of an actual or simulated automatic initiation signal. This includes verification of the correct automatic positioning of the valves and of the operation of each interlock and timer. As noted, actual makeup to the suppression pool may be excluded. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.4.6 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. 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. FSAR, Section [6.2].
2. FSAR, Chapter [15].

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

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.1 Primary Containment Hydrogen Recombiners

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 in Light-Water-Cooled Reactors" (Ref. 1), and GDC 41, "Containment Atmosphere Cleanup" (Ref. 2). The primary containment hydrogen recombiners ~~is~~ are required to reduce the hydrogen concentration in the primary containment following a loss of coolant accident (LOCA). The primary containment hydrogen recombiners accomplish~~es~~ this by recombining hydrogen and oxygen to form water vapor. The vapor ~~remains in the primary containment~~ is condensed and returned to the suppression pool, 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~~ hours after a Design Basis Accident (DBA). 1

Two 100% capacity independent primary containment hydrogen recombiner subsystems are provided and are shared between Unit 1 and Unit 2. Each consists of ~~controls located in the control room~~ and in the auxiliary electric equipment room, a power supply, and a recombiner located in ~~primary containment. The recombiners have no moving parts.~~ the reactor building. Recombination is accomplished by heating a hydrogen air mixture to > [1150]°F. The resulting water vapor and discharge gases are cooled prior to discharge from the unit. Air flows through the unit at ~~[100]~~ 125 cfm, with ~~natural circulation~~ a blower in the unit providing the motive force. A single recombiner is capable of maintaining the hydrogen concentration in primary containment below the 4.0 volume percent (v/o) flammability limit. Two recombiners are provided to meet the requirement for redundancy and independence. Each recombiner is powered from a separate Engineered Safety Feature bus and is provided with separate power panel and control panel with one recombiner powered from Unit 1 and the other recombiner powered from Unit 2.

Emergency operating procedures direct that the hydrogen concentration in primary containment be monitored following a DBA and that the primary containment hydrogen recombiner be manually activated to prevent the primary containment atmosphere from reaching a bulk hydrogen concentration of 4.0 v/o.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

(s) 1

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 as a function of time following the initiation of the accident is calculated. Assumptions recommended by Reference 3 are used to maximize the amount of hydrogen calculated.

were complied with 1

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

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

10CFR50.36(c)(2)(ii) 1

LCO

Two primary containment hydrogen recombiners must be OPERABLE. This ensures operation of at least one primary containment hydrogen recombiner 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.

(continued)

BASES (continued)

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 production 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(s) ~~is~~ are not required in MODE 3. 1

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(s) ~~is~~ are not required in these MODES. 1

ACTIONS

A.1

With one primary containment hydrogen recombiner inoperable, the inoperable primary containment hydrogen recombiner must be restored to OPERABLE status within 30 days. In this 3 condition, the remaining OPERABLE primary containment 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 to prevent hydrogen accumulation 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 stating 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

(continued)

BASES

ACTIONS

A.1 (continued)

limit, the low probability of the failure of the OPERABLE recombiner, 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.]~~ 4

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 ~~(one division of the hydrogen ignitors)~~ the Primary Containment Vent and Purge System 2. 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:]~~ 4 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. 2 It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of the alternate hydrogen control system. If the ability to perform the hydrogen control function is maintained, continued operation is permitted with two hydrogen recombiners inoperable for up to 7 days. Seven days is a reasonable time to allow two hydrogen recombiners to be inoperable because the hydrogen control function is maintained and because of the low probability of the occurrence of a LOCA that would generate hydrogen in the amounts capable of exceeding the flammability limit.

(continued)

BASES

ACTIONS (continued)

C.1

associated [3]

If any Required Action and ~~required~~ 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 requires verification that the ~~minimum heater sheath~~ reaction chamber gas [1] temperature increases to ≥ [1200] 1175 [2] °F in ≤ [5] hours and that ~~it is maintained > [1150] and < [1300]°F for ≥ [4] hours to~~ check the capability of the recombiner to properly function (and that significant heater elements are not burned out). [2] [2] [5]

[5] by determining that the current in each phase differs by less than or equal to 5% from the other phases and is within 5% of the value observed in the original acceptance test, corrected for line voltage differences.

Operating experience has shown that these components usually pass the Surveillance when performed at the [18] 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. [2]

[6]

~~SR 3.6.3.1.2~~

~~This SR ensures that there are no physical problems that could affect primary containment hydrogen recombiner operation. Since the recombiners are mechanically passive, they are not subject to mechanical failure. The only credible failures involve loss of power, blockage of the internal flow path, missile impact, etc. A visual inspection is sufficient to determine abnormal conditions that could cause such failures.~~

~~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.~~

(continued)

BASES

5

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.1.~~3~~ 2 6

within 30 minutes following completion of SR 3.6.3.1.1

This SR requires performance of a resistance to ground test of each heater phase to ensure that there are no detectable grounds in any heater phase. This is accomplished by verifying that the resistance to ground for any heater phase is ≥ ~~[10,000]~~ 1.0 E5 2 ohms.

Operating experience has shown that 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.

REFERENCES

1. 10 CFR 50.44.
2. 10 CFR 50, Appendix A, GDC 41.
3. Regulatory Guide 1.7, Revision ~~[1]~~ 0 2, March 10, 1971 1.
4. U 1 FSAR, Section ~~[~~6.2.5~~]~~. 2

Insert ITS B 3.6.3.2 (BWR/4 ISTS B 3.6.3.3) 7

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis 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. This Reviewer's Note has been deleted. This information is for the NRC reviewer to be keyed in to what is needed to meet this requirement. This is not meant to be retained in the final version of the plant specific submittal.

5. Change made to provide the current licensing basis details related to performing the Surveillance.

6. Changes made to be consistent with changes made to the Specification. The following SR has been renumbered due to the changes.

7. A new Bases has been added, ITS Bases 3.6.3.2. This Bases is from the BWR/4 ISTS (NUREG-1433 ISTS B 3.6.3.3), since the LaSalle 1 and 2 design is similar to the BWR/4 design with regard to oxygen concentration requirement (LaSalle 1 and 2 inerts the primary containment since the containment is a Mark II). Therefore, the BWR/4 Bases is used and any deviations from the BWR/4 ISTS Bases are discussed in the Justification for Deviations for ITS Bases: 3.6.3.2.

1 * Insert BWR/4 ISTS B3.6.3.3

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.1 2 1 Primary Containment Oxygen Concentration

BASES

The primary containment is 2

BACKGROUND

~~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 inerted 2 ~~inert~~, 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]")~~ 3 to provide redundant and diverse methods to mitigate events that produce hydrogen. 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. This LCO ensures that oxygen concentration does not exceed 4.0 v/o during operation in the applicable conditions.

APPLICABLE SAFETY ANALYSES

The Reference 1 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.

Primary containment oxygen concentration satisfies Criterion 2 of ~~the NRC Policy Statement~~.
10 CFR 50.36(c)(2)(ii) 4

* This BWR/4 Bases Insert was used to match the BWR/4 Specification inserted in the LCO Section

(continued)

Insert Page B 3.6-89a

Insert BWR/4 ISTS B3.6.3.3 1
(continued)

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 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.

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 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. 5

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 mitigating systems (e.g., hydrogen recombiners) and the low probability and long duration of an event that would generate significant amounts of hydrogen occurring during this period.

(continued)

Insert Page B 3.6-80b

1 Insert BWR/4 ISTS B 3.6.3.3

(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. 5

SURVEILLANCE REQUIREMENTS

SR 3.6.3.~~8~~.1 2 1

The primary containment must be determined to be iner~~t~~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. 6 4

REFERENCES

1. UFSAR, Section ~~[~~6.2.5~~]~~. 4 5

Insert Page B 3.6-80c

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.6.3.2 - PRIMARY CONTAINMENT OXYGEN CONCENTRATION

1. A new Bases has been added, ITS Bases 3.6.3.2. This Bases is from the BWR/4 ISTS (NUREG-1433 ISTS B 3.6.3.3), since the LaSalle 1 and 2 design is similar to the BWR/4 design with regard to the inerting requirements of the primary containment. Therefore, the BWR/4 Bases are used and any deviations from the BWR/4 ISTS are discussed below.

2. Editorial change made for enhanced clarity.

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

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

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

6. Typographical/grammatical error corrected.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.2 Primary Containment and Drywell Hydrogen Ignitors

BASES

BACKGROUND

The primary containment and drywell hydrogen ignitors are a part of the combustible gas control required by 10 CFR 50.44 (Ref. 1) and GDC 41, "Containment Atmosphere Cleanup" (Ref. 2), to reduce the hydrogen concentration in the primary containment following a degraded core accident. The hydrogen ignitors ensure the combustion of hydrogen in a manner such that containment overpressure failure is prevented as a result of a postulated degraded core accident.

10 CFR 50.44 (Ref. 1) requires boiling water reactor units with Mark III containments to install suitable hydrogen control systems. The hydrogen ignitors are installed to accommodate an amount of hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water. This requirement was placed on reactor units with Mark III containments because they were not designed for inerting and because of their low design pressure. Calculations indicate that if hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water were to collect in primary containment, the resulting hydrogen concentration would be far above the lower flammability limit such that, without the hydrogen ignitors, if the hydrogen were ignited from a random ignition source, the resulting hydrogen burn would seriously challenge the primary containment.

The hydrogen ignitors are based on the concept of controlled ignition using thermal ignitors designed to be capable of functioning in a post accident environment, seismically supported and capable of actuation from the control room. Ignitors are distributed throughout the [32] regions of the drywell and primary containment in which hydrogen could be released or to which it could flow in significant quantities. The hydrogen ignitors are arranged in two independent divisions such that each containment region has two ignitors, one from each division, controlled and powered redundantly so that ignition would occur in each region even if one division failed to energize.

(continued)

BASES

BACKGROUND (continued)

When the hydrogen ignitors are energized they heat up to a surface temperature ≥ [1700]°F. At this temperature, they ignite the hydrogen gas that is present in the airspace in the vicinity of the ignitor. The hydrogen ignitors depend on the dispersed location of the ignitors so that local pockets of hydrogen at increased concentrations would burn before reaching a hydrogen concentration significantly higher than the lower flammability limit. Hydrogen ignition in the vicinity of the ignitors is assumed to occur when the local hydrogen concentration reaches [8.0] volume percent (v/o) and results in [85]% of the hydrogen present being consumed.

APPLICABLE SAFETY ANALYSES

The hydrogen ignitors cause hydrogen in containment to burn in a controlled manner as it accumulates following a degraded core accident (Ref. 3). Burning occurs at the lower flammability concentration, where the resulting temperatures and pressures are relatively benign. Without the system, hydrogen could build up to higher concentrations that could result in a violent reaction if ignited by a random ignition source after such a buildup.

The hydrogen ignitors are not included for mitigation of a Design Basis Accident (DBA) because an amount of hydrogen equivalent to that generated from the reaction of 75% of the fuel cladding with water is far in excess of the hydrogen calculated for the limiting DBA loss of coolant accident (LOCA). The hydrogen concentration resulting from a DBA can be maintained less than the flammability limit using the hydrogen recombiners. However, the hydrogen ignitors have been shown by probabilistic risk analysis to be a significant contributor to limiting the severity of accident sequences that are commonly found to dominate risk for units with Mark III containment.

The hydrogen ignitors are considered to be risk significant in accordance with the NRC Policy Statement.

LCO

Two divisions of primary containment and drywell hydrogen ignitors must be OPERABLE, each with more than 90% of the ignitors OPERABLE.

(continued)

1

BASES

LCO (continued)	This ensures operation of at least one ignitor division, with adequate coverage of the primary containment and drywell, in the event of a worst case single active failure. This will ensure that the hydrogen concentration remains near 4.0 v/o.
APPLICABILITY	In MODES 1 and 2, the hydrogen ignitor is required to control hydrogen concentration to near the flammability limit of 4.0 v/o following a degraded core event that would generate hydrogen in amounts equivalent to a metal water reaction of 75% of the core cladding. The control of hydrogen concentration prevents overpressurization of the primary containment. The event that could generate hydrogen in quantities sufficiently high enough to exceed the flammability limit is limited to MODES 1 and 2. In MODE 3, both the hydrogen production rate and the total hydrogen produced after a degraded core accident 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 hydrogen ignitor is low. Therefore, the hydrogen ignitor is not required in MODE 3. In MODES 4 and 5, the probability and consequences of a degraded core accident are reduced due to the pressure and temperature limitations. Therefore, the hydrogen ignitors are not required to be OPERABLE in MODES 4 and 5 to control hydrogen.
ACTIONS	<u>A.1</u> With one hydrogen ignitor division inoperable, the inoperable division must be restored to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE hydrogen ignitor division is adequate to perform the hydrogen burn function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced hydrogen control capability. The 30 day Completion Time is based on the low probability of the occurrence of a degraded core event that would generate hydrogen in amounts equivalent to a metal water reaction of

(continued)

BASES

ACTIONS

A.1 (continued)

75% of the core cladding, the amount of time available after the event for operator action to prevent hydrogen accumulation from exceeding the flammability limit, and the low probability of failure of the OPERABLE hydrogen ignitor division.

Required Action A.1 has been modified by a Note indicating the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one hydrogen ignitor division is inoperable or when one or more areas with adjacent ignitors are inoperable. The allowance is provided because of the low probability of the occurrence of an event that would generate hydrogen in amounts capable of exceeding the flammability limit, the low probability of the failure of both hydrogen ignitor divisions or adjacent ignitors, and the amount of time available after the event for operator action to prevent exceeding the flammability limit.

B.1 and B.2

With two primary containment and drywell ignitor divisions 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 one hydrogen recombiner and one drywell purge subsystem. The 1 hour Completion Time allows a reasonable period of time to verify that a loss of hydrogen control function does not exist. The verification may be performed as an administrative check by examining logs or other information to determine the availability of the alternate hydrogen control capabilities. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of the alternate hydrogen control capabilities. If the ability to perform the hydrogen control function is maintained, continued operation is permitted with two ignitor divisions inoperable for up to 7 days. Seven days is a reasonable time to allow two ignitor divisions to be inoperable because the hydrogen control function is maintained and because of the low probability of the occurrence of a LOCA that would generate hydrogen in the amounts capable of exceeding the flammability limit.

(continued)

BASES

ACTIONS (continued)

C.1

If any Required Action and required 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 and SR 3.6.3.2.2

These SRs verify that there are no physical problems that could affect the ignitor operation. Since the ignitors are mechanically passive, they are not subject to mechanical failure. The only credible failures are loss of power or burnout. The verification that each required ignitor is energized is performed by circuit current versus voltage measurement.

The Frequency of 184 days has been shown to be acceptable through operating experience because of the low failure occurrence, and provides assurance that hydrogen burn capability exists between the more rigorous 18 month Surveillances. Operating experience has shown these components usually pass the Surveillance when performed at a 184 day Frequency. Additionally, these surveillances must be performed every 92 days if four or more ignitors in any division are inoperable. The 92 day Frequency was chosen, recognizing that the failure occurrence is higher than normal. Thus, decreasing the Frequency from 184 days to 92 days is a prudent measure, since only two more inoperable ignitors (for a total of six) will result in an inoperable ignitor division. SR 3.6.3.2.2 is modified by a Note that indicates that the Surveillance is not required to be performed until 92 days after four or more ignitors in the division are discovered to be inoperable.

[SR 3.6.3.2.3 and SR 3.6.3.2.4

These functional tests are performed every 18 months to verify system OPERABILITY. The current draw to develop a]

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.2.3, and SR 3.6.3.2.4 (continued)

surface temperature of ≥ 1700°F is verified for ignitors in inaccessible areas, e.g., in a high radiation area. Additionally, the surface temperature of each accessible ignitor is measured to be ≥ 1700°F to demonstrate that a temperature sufficient for ignition is achieved. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. 10 CFR 50.44.
2. 10 CFR 50, Appendix A, GDC 41.
3. FSAR, Section [6.2.5].

1

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ISTS BASES: 3.6.3.2 - PRIMARY CONTAINMENT AND DRYWELL HYDROGEN IGNITORS

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

1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.3 [Drywell Purge System]

BASES

BACKGROUND

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

The [Drywell Purge System] is an Engineered Safety Feature and is designed to operate following a loss of coolant accident (LOCA) in post accident environments without loss of function. The system has two independent subsystems, each consisting of a compressor and associated valves, controls, and piping. Each subsystem is sized to pump [500] scfm. 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.

Following a LOCA, the drywell is immediately pressurized due to the release of steam into the drywell environment. This pressure is relieved by the lowering of the water level within the weir wall, clearing the drywell vents and allowing the mixture of steam and noncondensibles to flow into the primary containment through the suppression pool, removing much of the heat from the steam. The remaining steam in the drywell begins to condense as steam flow from the reactor pressure vessel ceases, the drywell pressure falls rapidly. Both drywell purge compressors start automatically 30 seconds after a LOCA signal is received from the Emergency Core Cooling System instrumentation, but only when drywell pressure has decreased to within approximately [0.087] psi above primary containment pressure. This ensures the blowdown from the drywell to the primary containment is complete. The drywell purge compressors force air from the primary containment into the drywell. Drywell pressure increases until the water level between the weir wall and the drywell is forced down to the first row of suppression pool vents forcing drywell atmosphere back into containment and mixing with containment atmosphere to dilute the hydrogen. While drywell purge continues following the LOCA, hydrogen continues to be produced. Eventually, the 4.0 v/o limit is again approached

(continued)

BASES

BACKGROUND (continued)

and the hydrogen recombiners are manually placed in operation.

APPLICABLE SAFETY ANALYSES

The [Drywell Purge System] provides the capability for reducing the drywell hydrogen concentration to approximately the bulk average primary containment 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; and

b. Radiolytic decomposition of water in the Reactor Coolant System and drywell sump.

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.

[Based on a conservative assumption used to calculate the hydrogen concentration versus time after a LOCA, the hydrogen concentration in the primary containment would reach [3.5 v/o about 6 days] after the LOCA and [4.0 v/o about 2 days] later if no hydrogen mixing and recombiner were functioning (Ref. 2).]

The [Drywell Purge System] satisfies Criterion 3 of the NRC Policy Statement.

LCO

Two [drywell purge] subsystems must be OPERABLE to ensure operation of at least one primary containment [drywell purge] subsystem in the event of a worst case single active failure. Operation with at least one OPERABLE [drywell purge] subsystem provides the capability of controlling the hydrogen concentration in the drywell without exceeding the flammability limit.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1 and 2, the two [drywell purge] subsystems ensure the capability to prevent localized hydrogen concentrations above the flammability limit of 4.0 v/o in the 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 Purge System] is low. Therefore, the [Drywell Purge System] is 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 Purge System] is not required in these MODES.

ACTIONS

A.1

With one [drywell purge] subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE subsystem is adequate to perform the drywell purge function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in reduced drywell purge capability. The 30 day Completion Time is based on the availability of the second subsystem, the low probability of a LOCA that would generate hydrogen in amounts capable of exceeding the flammability limit, and the amount of time available after the event for operator action to prevent hydrogen accumulation from exceeding this limit.

Required Action A.1 has been modified by a Note indicating the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when one subsystem 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.

(continued)

BASES

ACTIONS (continued)

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 purge] 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 [one division of the hydrogen ignitors]. 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 may [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 purge] subsystems inoperable for up to 7 days. Seven days is a reasonable time to allow two [drywell purge] 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.

C.1

If any Required Action and the required 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.

(continued)

1

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.3.3.1

Operating each [drywell purge] subsystem for ≥ 15 minutes ensures that each subsystem is OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, compressor failure, or excessive vibration can be detected for corrective action. The 92 day Frequency is consistent with Inservice Testing Program Frequencies, operating experience, the known reliability of the compressor and controls, and the two redundant subsystems available.

[SR 3.6.3.3.2

Verifying that each [drywell purge] subsystem flow rate is ≥ [500] scfm ensures that each subsystem is capable of maintaining drywell 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 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. Regulatory Guide 1.7, Revision [1].
2. FSAR, Section [6.2.5].

1

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

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.

The secondary containment is a structure that completely encloses the primary containment 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/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."

APPLICABLE SAFETY ANALYSES

There are ~~three~~ two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a LOCA (Ref. 1), ~~a fuel handling accident inside primary containment (Ref. 2),~~ and a fuel handling accident ~~in the auxiliary building~~ (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,

(continued)

1

BASES

APPLICABLE SAFETY ANALYSES (continued)

and that fission products entrapped within the secondary containment structure will be treated by the SGT System prior to discharge to the environment.

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

LCO 2

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. 1 1

, the hatches and blowout panels must be closed and sealed, the sealing mechanisms associated with each secondary containment penetration (e.g., welds, bellows, or O-rings) 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.

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. 1

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 ~~primary or~~ secondary containment. 1 1

ACTIONS

A.1

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)

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. 1

B.1 and B.2

If the 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 ~~primary or~~ 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. 1

1

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~~

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,

3

(continued)

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

3

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.

1

BASES

ACTIONS

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

3

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

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~~. 2 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

SR 3.6.4.1.2 ~~and SR 3.6.4.1.3~~ 4

one 4 1

in each access opening

Verifying that ~~one~~ {secondary containment} ~~equipment hatches and~~ access doors ~~are~~ 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~~ doors 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.~~ Maintaining {secondary containment} OPERABILITY requires verifying ~~each~~ one door in the access opening is closed~~, except when the access opening is being used for entry and exit; then, at least one door must remain closed~~. The 31 day Frequency for ~~these~~ this SR~~s~~ 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~~.

1 TSTF -18 one 4 1 TSTF -18

INSERT B3.6.4.1.2 6 1

this 4

existing administrative controls on door status 2

(continued)

TSTF-18

Insert B 3.6.4.1.2

1 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 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 or outer door. 6 The intent is to not 1 breach the secondary containment at any time when secondary containment is required. This is achieved by maintaining the inner or outer portion of the 1 barrier closed at all times, i.e., all inner doors closed or all outer doors closed. Thus each access opening has one door closed. 6 However, all 1 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.

Each SGT subsystem is designed to drawdown pressure in the secondary containment to ≥ 0.25 inches vacuum water guage in ≤ 300 seconds and maintain pressure in the secondary containment at ≥ 0.25 inches of vacuum water guage for 1 hour at a flow rate of ≤ 4400 cfm. 2

All changes are 1 unless otherwise indicated

BASES

SURVEILLANCE REQUIREMENTS (continued)

4 [SR 3.6.4.1 ~~4~~ 3 and] SR 3.6.4.1 ~~5~~ 4

released to secondary containment 5

y The SGT System exhausts the *secondary containment* atmosphere to the environment through appropriate treatment equipment. To ensure that all fission products are treated, SR 3.6.4.1.3 and ~~SR 3.6.4.1.4 verifies~~ that ~~the SGT System will rapidly establish and maintain~~ can rapidly be established and maintained a pressure in the *secondary containment* that is less than the ~~lowest postulated~~ pressure external to the *secondary containment* boundary. 5 Establishment of this pressure SR 3.6.4.1.3, which ~~This~~ is confirmed by demonstrat~~ing~~ that one ~~SGT subsystem will draw down~~ the *secondary containment* to ≥ ~~*0.25*~~ inches of vacuum water gauge in ≤ ~~*120*~~ 300 seconds can be drawn down using one SGT subsystem. 5 When the SGT System is operating as designed, the establishment and maintenance of secondary containment pressure ~~This~~ cannot be accomplished if the *secondary containment* boundary is not intact. SR 3.6.4.1 ~~5~~ 4 demonstrates that ~~each~~ ~~SGT subsystem can maintain~~ the pressure in the secondary containment can be maintained using one SGT subsystem 5 ≥ ~~*0.266*~~ 0.25 inches of vacuum water gauge for 1 hour at a flow rate ≤ ~~*4000*~~ 4400 cfm. This flow rate is the assumed secondary containment leak rate during the drawdown period. 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 subsystem ~~s are tested on a~~ used for these surveillances is ~~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 the ~~se~~ ~~components~~ secondary containment boundary usually pass the Surveillance when performed at the ~~*18*~~ 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. 4

The inoperability of the SGT System does not necessarily constitute a failure of these Surveillances relative to secondary containment OPERABILITY. 5 The primary purpose of the SRs is to ensure secondary containment boundary integrity. The secondary purpose of the SRs is to ensure the SGT subsystem being tested functions as designed. There is a separate LCO with Surveillance Requirements that serves the primary purpose of ensuring OPERABILITY of the SGT System. These SRs

REFERENCES

1. U *FSAR, Section *15.6.5*.

~~2. FSAR, Section [15.7.6].~~ 2

2 ~~3~~. U *FSAR, Section *15.7.4*.

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, or analysis description.

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

5. ISTS SRs 3.6.4.1.4 and 3.6.4.1.5 are tests that ensure the Secondary Containment is OPERABLE; the leak tightness of the Secondary Containment boundary is within the assumptions of the accident analyses. However, they are written in such a manner that they imply that if a SGT subsystem is inoperable, the SRs are failed ("Verify each standby gas treatment (SGT) subsystem will/can..."). As stated above, this is not the intent of the SRs. Therefore, to ensure this misinterpretation cannot occur, the SRs and this Bases description have been rephrased to more clearly convey the original intent of the SRs, to verify the Secondary Containment is OPERABLE. With the new wording, if a SGT subsystem is inoperable, ITS SRs 3.6.4.1.3 and 3.6.4.1.4 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. 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, that are released during certain operations when primary containment is not required to be OPERABLE, or that take place outside primary containment, are maintained within the secondary containment boundary.

1 5 4

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 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), and blind flanges are considered passive devices. ~~Check valves or other automatic valves designed to close without operator action following an accident are considered active devices. Isolation barrier(s) for the penetration are discussed in Reference 2.~~ 1

(i.e., dampers) 1

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

Required to be closed during accident conditions

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

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~~ 1

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

primary containment (Ref. ~~3~~ 2) ~~and a fuel handling accident in the auxiliary building (Ref. 4)~~. The secondary containment performs no active function in response to each of these limiting events, but the boundary 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. 1

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.

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

LCO

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

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The automatic ~~power operated~~ 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 5~~ the Technical Requirements Manual (Ref. 3). 1

The normally closed ~~isolation valves or blind flanges~~ manual SCIVs are considered OPERABLE when ~~manual~~ the valves are closed ~~or open in accordance with appropriate~~ under administrative controls ~~, or automatic SCIVs are de-activated and secured in their closed position, and blind flanges are in place~~. These passive isolation valves or devices are listed in Reference ~~5~~ 3. 4 2 8 1

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, OPERABILITY of SCIVs is required.

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

(continued)

BASES

APPLICABILITY (continued)

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 ~~. Moving irradiated fuel assemblies in the [primary or secondary containment] may also occur in MODES 1, 2, and 3.~~

in the Secondary Containment

5

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 the need for 3 secondary containment isolation is indicated.

4 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 criteria 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

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

secondary containment. This Required Action must be completed within the 8 hour Completion Time. The specified time period is reasonable considering the time required to isolate the penetration and the low probability of a DBA, which requires the SCIVs to close, occurring during this short time.

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. This Required Action does not require any testing or device manipulation. Rather, it involves verification that the affected penetration remains isolated.

5

The Completion Time of once per 31 days is appropriate because the isolation devices are operated under administrative controls and the probability of their misalignment is low.

Required Action A.2 is modified by ~~a Note that~~ two Notes. applies to devices 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. Therefore, the probability of misalignment, once they have been verified to be in the proper position, is low.

Note 1

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5 isolation

Insert A.1 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 low probability of a DBA, which requires the SCIVs to close, occurring during this short time.

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

(continued)

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 cannot be 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 ~~primary and~~ secondary ~~containment~~ must be immediately suspended. 3 Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action 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.

5 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 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 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 D.1, D.2, and D.3 5

(continued)

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.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

not locked, sealed, or otherwise secured, and is

TSTF-45 rev. 2

This SR verifies each secondary containment isolation manual 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.

3

Since these SCIVs are readily accessible to personnel during normal unit 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.

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

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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.

6

Insert SR 3.6.4.2-1

SR 3.6.4.2.2

Verifying 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].

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7

7

3

(continued)

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

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 3.3.6.2.5~~ 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. Operating experience has shown these components usually pass the Surveillance when performed at the ~~[18]~~ 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

3

1

LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation,"

5

While this Surveillance can be

1

1

which is based on the refueling cycle

24 3

REFERENCES

1. (4) FSAR, Section [15.6.5]. 3 1

~~2. FSAR, Section [6.2.3].~~

~~3. FSAR, Section [15.7.6].~~ 1

(2) ~~4~~. (4) FSAR, Section [15.7.4]. 3. 1

(3) ~~5~~. ~~FSAR, Section [ ]~~. Technical Requirements Manual 1

1

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

2. 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.

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

4. Typographical/grammatical error corrected.

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. Editorial change made for enhanced clarity.

7. 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 will not include the times for the SCIVs. They are located in the Technical Requirements Manual.

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

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.

1 that are shared between Unit 1 and Unit 2

The SGT System consists of two ~~fully redundant~~ independent 4 subsystems, each with its own set of ductwork, dampers, charcoal filter train, and controls. Each SGT System discharges to the plant vent stack through a common exhaust pipe. 1

Each charcoal filter train consists of (components listed in order of the direction of the air flow):

1

b a. A ~~moisture separator~~; demister

c b. An electric heater;

d c. A prefilter; bank

e d. A high efficiency particulate air (HEPA) filter; bank

f e. A charcoal adsorber;

g f. A second HEPA filter; bank. and

a g. A centrifugal filter unit fan ~~with inlet flow control vanes~~. and centrifugal cooling fans;

1 Each SGT subsystem is capable of processing the secondary containment volume which includes both Unit 1 and Unit 2.

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 auxiliary and enclosure building structures~~. 1 The internal pressure of the SGT System boundary region is maintained at a negative pressure of [0.25] 2 inch water gauge when the system is in operation, which represents the internal pressure required to ensure zero exfiltration of air from the building ~~when exposed to a [10] mph wind blowing at an angle of [45]° to the building~~. 1

The ~~moisture separator~~ demister 1 is provided to remove entrained water in the air, while the electric heater reduces the relative

(continued)

BASES

BACKGROUND (continued)

humidity of the airstream to ~~less than~~ 70% (Ref. 2). The prefilter removes large particulate matter, while the HEPA filter is provided to remove fine particulate matter and protect the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides, and the final HEPA filter is provided to collect any carbon fines exhausted from the charcoal adsorber.

≤ 1 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 enclosure building recirculation fans and both charcoal filter train~~ fans start. SGT System flows are controlled by ~~modulating inlet vanes installed on the charcoal filter train exhaust fans and two position volume control dampers installed in branch ducts to individual regions of the secondary containment~~.

from either Unit 1 or Unit 2

1

supply

automatically

flow control dampers located upstream of the supply fans

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. 3). 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.

1 5 and 4

The SGT System satisfies Criterion 3 of ~~the NRC Policy Statement~~.

10 CFR 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.

3

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.

(continued)

BASES

APPLICABILITY (continued)

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 System OPERABLE 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 ~~[primary or]~~ secondary containment~~s~~. 1

ACTIONS

A.1

With one SGT subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this 9 ~~C~~condition, the remaining OPERABLE SGT subsystem is adequate 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 subsystem 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 2 ~~[primary or]~~ secondary containment~~s~~, during CORE ALTERATIONS, or during OPDRVs, when Required Action A.1 cannot be

(continued)

BASES

ACTIONS

C.1, C.2.1, C.2.2, and C.2.3 (continued)

completed within the required Completion Time, the OPERABLE SGT subsystem should be immediately placed in operation. This Required Action ensures that the remaining subsystem is OPERABLE, that no failures that could prevent automatic actuation have occurred, and that any other failure would be readily detected. will

4

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 unit in a condition that minimizes risk. 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 position. Also, if applicable, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until OPDRVs are suspended.

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

Insert Action C

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.

D.1

If both SGT 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.

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

When two SGT subsystems are inoperable, if applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the ~~primary and~~ secondary containment~~s~~ must be immediately

2

(continued)

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.

BASES

ACTIONS

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

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,

suspended. 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 to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until OPDRVs are suspended. 4

Required Action E.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 sufficient reason to require a reactor shutdown.~~

4

Insert E.1, E.2, and E.3

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

(from the control room) 5

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. 2 2

SR 3.6.4.3.2

ANSI/ASME N510-1989 1

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. 4). 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). Specified test frequencies 5 1

(continued)

Insert E.1, E.2, and E.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

SR 3.6.4.3.2 (continued)

and additional information are discussed in detail in the VFTP.

SR 3.6.4.3.3

4 LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation,"

This SR requires verification that each SGT subsystem starts upon receipt of an actual or simulated initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in ~~SR 3.3.6.2.5~~ overlaps this SR to provide complete testing of the safety function. While this Surveillance can be performed with the reactor at power, operating experience has shown these components usually pass the Surveillance when performed at the 2 24 ~~[18]~~ month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

6

~~SR 3.6.4.3.4~~

~~This SR requires verification that the SGT 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 these components usually pass the Surveillance when performed at the [18] month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.
2. 1 U FSAR, Section ~~[~~6.~~2.3~~ 5.1 ~~]~~.
3. FSAR, Section ~~[~~15.6.5~~]~~. 2
4. ~~Regulatory Guide 1.52, Rev. [2]~~ ANSI/ASME N510-1989. 1 5 1
4. UFSAR, Section 15.7.4.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, 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, or analysis 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. Changes have been made to reflect those changes made to the Specification.

1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5.1 Drywell

BASES

BACKGROUND

The drywell houses the reactor pressure vessel (RPV), the reactor coolant recirculating loops, and branch connections of the Reactor Coolant System (RCS), which have isolation valves at the primary containment boundary. The function of the drywell is to maintain a pressure boundary that channels steam from a loss of coolant accident (LOCA) to the suppression pool, where it is condensed. Air forced from the drywell is released into the primary containment. The pressure suppression capability assures that peak LOCA temperature and pressure in the primary containment are within design limits. The drywell also protects accessible areas of the containment from radiation originating in the reactor core and RCS.

To ensure the drywell pressure suppression capability, the drywell bypass leakage must be minimized to prevent overpressurization of the primary containment during the drywell pressurization phase of a LOCA. This requires periodic testing of the drywell bypass leakage, confirmation that the drywell air lock is leak tight, OPERABILITY of the drywell isolation valves (DIVs), and confirmation that the drywell vacuum relief valves are closed.

The isolation devices for the drywell penetrations are a part of the drywell barrier. To maintain this barrier:

a. The drywell air lock is OPERABLE except as provided in LCO 3.6.5.2, "Drywell Air Lock";

b. The drywell penetrations required to be closed during accident conditions are either:

   1. capable of being closed by an OPERABLE automatic DIV, or

   2. closed by manual valves, blind flanges, or de-activated automatic valves secured in closed positions except as provided in LCO 3.6.5.3, "Drywell Isolation Valves (DIVs)"; and

(continued)

1

BASES

BACKGROUND (continued)

c. The Drywell Vacuum Relief System is OPERABLE except as provided in LCO 3.6.5.6, "Drywell Vacuum Relief System."

This Specification is intended to ensure that the performance of the drywell in the event of a DBA meets the assumptions used in the safety analyses (Ref. 1).

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving the drywell are presented in Reference 1. The safety analyses assume that for a high energy line break inside the drywell, the steam is directed to the suppression pool through the horizontal vents where it is condensed. Maintaining the pressure suppression capability assures that safety analyses remain valid and that the peak LOCA temperature and pressure in the primary containment are within design limits.

The drywell satisfies Criteria 2 and 3 of the NRC Policy Statement.

LCO

Maintaining the drywell OPERABLE is required to ensure that the pressure suppression design functions assumed in the safety analyses are met. The drywell is OPERABLE if the drywell structural integrity is intact and the bypass leakage is within limits, except prior to the first startup after performing a required drywell bypass leakage test. At this time, the drywell bypass leakage must be ≤ [10%] of the drywell bypass leakage limit.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to the 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 drywell is not required to be OPERABLE in MODES 4 and 5.

(continued)

BASES (continued)

ACTIONS

A.1

In the event the drywell is inoperable, it 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 the drywell OPERABLE during MODES 1, 2, and 3. This time period also ensures that the probability of an accident (requiring drywell OPERABILITY) occurring during periods when the drywell is inoperable is minimal. Also, the Completion Time is the same as that applied to inoperability of the primary containment in LCO 3.6.1.1, "Primary Containment."

B.1 and B.2

If the drywell 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 3.6.5.1.1

The analyses in Reference 2 are based on a maximum drywell bypass leakage. This Surveillance ensures that the actual drywell bypass leakage is less than or equal to the acceptable $A/\sqrt{k}$ design value of [1.0] ft^2 assumed in the safety analysis. As left drywell bypass leakage, prior to the first startup after performing a required drywell bypass leakage test, is required to be ≤ [10%] of the drywell bypass leakage limit. At all other times between required drywell 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 [18] months, consistent with the difficulty of performing the test, risk of high radiation exposure, and the remote possibility that a component failure that is not identified by some other drywell or primary containment SR

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.1.1 (continued)

might occur. 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 3.6.5.1.2

The exposed accessible drywell interior and exterior surfaces are inspected to ensure there are no apparent physical defects that would prevent the drywell from performing its intended function. This SR ensures that drywell structural integrity is maintained. The [40] month Frequency was chosen so that the interior and exterior surfaces of the drywell can be inspected at every other refueling outage. Due to the passive nature of the drywell structure, the [40] month Frequency is sufficient to identify component degradation that may affect drywell structural integrity.

REFERENCES

1. FSAR, Chapter [6] and Chapter [15].

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ISTS BASES: 3.6.5.1 - DRYWELL

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

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5.2 Drywell Air Lock

BASES

BACKGROUND

The drywell air lock forms part of the drywell boundary and provides a means for personnel access during MODES 2 and 3 during low power phase of unit startup. For this purpose, one double door drywell air lock has been provided, which maintains drywell isolation during personnel entry and exit from the drywell. Under the normal unit operation, the drywell air lock is kept sealed. The air pressure in the seals is maintained > [60] psig by the seal air flask and pneumatic system, which is maintained at a pressure > [90] psig.

The drywell air lock is designed to the same standards as the drywell boundary. Thus, the drywell air lock must withstand the pressure and temperature transients associated with the rupture of any primary system line inside the drywell and also the rapid reversal in pressure when the steam in the drywell is condensed by the Emergency Core Cooling System flow following loss of coolant accident flooding of the reactor pressure vessel (RPV). It is also designed to withstand the high temperature associated with the break of a small steam line in the drywell that does not result in rapid depressurization of the RPV.

The air lock is nominally a right circular cylinder, [10] ft in diameter, with doors at each end that are interlocked to prevent simultaneous opening. During periods when the drywell is not required to be OPERABLE, the air lock interlock mechanism may be disabled, allowing both doors of the air lock to remain open for extended periods when frequent drywell entry is necessary. 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 Design Basis Accident (DBA).

The air lock is provided with limit switches on both doors that provide control room indication of door position. Additionally, control room indication is provided to alert the operator whenever an air lock interlock mechanism is defeated.

(continued)

BASES

BACKGROUND (continued)

The drywell air lock forms part of the drywell pressure boundary. Not maintaining air lock OPERABILITY may result in degradation of the pressure suppression capability, which is assumed to be functional in the unit safety analyses. The drywell air lock does not need to meet the requirements of 10 CFR 50, Appendix J (Ref. 1), since it is not part of the primary containment leakage boundary. However, it is prudent to specify a leakage rate requirement for the drywell air lock. A seal leakage rate limit of ≤ 200 scfh and an air lock overall leakage rate limit of ≤ 200 scfh, at pressure ≥ P_a (11.5 psig), have been established to assure the integrity of the seals.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving the drywell are presented in Reference 2. The safety analyses assume that for a high energy line break inside the drywell, the steam is directed to the suppression pool through the horizontal vents where it is condensed. Since the drywell air lock is part of the drywell pressure boundary, its design and maintenance are essential to support drywell OPERABILITY, which assures that the safety analyses are met.

The drywell air lock satisfies Criterion 3 of the NRC Policy Statement.

LCO

The drywell air lock forms part of the drywell pressure boundary. The air lock safety function assures that steam resulting from a DBA is directed to the suppression pool. Thus, the air lock's structural integrity is essential to the successful mitigation of such an event.

The air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, air lock leakage must be within limits, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of the drywell does not exist when the drywell is required to be OPERABLE.

Closure of a single door in the air lock is necessary to support drywell OPERABILITY following postulated events.

(continued)

BASES

LCO (continued)	Nevertheless, both doors are kept closed when the air lock is not being used for entry into and exit from the drywell.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to 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. Therefore, the drywell air lock is not required to be OPERABLE in MODES 4 and 5.

ACTIONS

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

The ACTIONS are modified by a second Note, which ensures appropriate remedial actions are taken when necessary. Pursuant to LCO 3.0.6, ACTIONS are not required even if the drywell is exceeding its bypass leakage limit. Therefore, the Note is added to require ACTIONS for LCO 3.6.5.1 to be taken in this event.

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

With one drywell air lock door inoperable, the OPERABLE door must be verified closed (Required Action A.1). This ensures that a leak tight drywell 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 3.6.5.1, "Drywell," which requires that the drywell be restored to OPERABLE status within 1 hour.

(continued)

BASES

ACTIONS

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

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

Required Action A.3 verifies that the air lock has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable drywell 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 that ensure that the OPERABLE air lock door remains closed.

The Required Actions are modified by two Notes. Note 1 ensures only the Required Actions and associated Completion Times of Condition C are required if both doors in the air lock are inoperable. The exception of the Note does not affect tracking the Completion Times 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. Drywell entry may be required to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside the drywell that are required by TS or activities on equipment that support TS-required equipment. This Note is not intended to preclude performing other activities (i.e., non-TS-required activities) if the drywell 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 drywell during the short time that the OPERABLE door is expected to be open.

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

With the drywell air lock interlock mechanism inoperable, the Required Actions and associated Completion Times consistent with Condition A are applicable.

(continued)

BASES

ACTIONS

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

The Required Actions are modified by two Notes. Note 1 ensures only the Required Actions and associated Completion Times of Condition C are required if both doors in the air lock are inoperable. Note 2 allows entry and exit into the drywell 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).

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

With the air lock inoperable for reasons other than those described in Condition A or B, Required Action C.1 requires action to be immediately initiated to evaluate drywell bypass leakage using current air lock test results. An evaluation is acceptable, since it is overly conservative to immediately declare the drywell inoperable if both doors in an air lock have failed a seal test or the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), drywell remains OPERABLE, yet only 1 hour (per LCO 3.6.5.1) 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 drywell leakage rate can still be within limits.

Required Action C.2 requires that one door in the drywell air lock must be verified to be closed. This Required Action must be completed within the 1 hour Completion Time. This specified time period is consistent with the ACTIONS of LCO 3.6.5.1, which requires that the drywell be restored to OPERABLE status within 1 hour.

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.

(continued)

BASES

ACTIONS (continued)

D.1 and D.2

If the inoperable drywell air lock 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 3.6.5.2.1

This SR requires a test be performed to verify seal leakage of the drywell air lock doors at pressures ≥ [11.5] psig. A seal leakage rate limit of ≤ [200] scfh has been established to ensure the integrity of the seals. The Surveillance is only required to be performed once after each closing. The Frequency of 72 hours is based on operating experience and is considered adequate in view of the other indications available to plant operations personnel that the seal is intact.

SR 3.6.5.2.2

Every 7 days the drywell air lock seal air flask pressure is verified to be ≥ [90] psig to ensure that the seal system remains viable. It must be checked because it could bleed down during or following access through the air lock, which occurs regularly. The 7 day Frequency has been shown to be acceptable, based on operating experience, and is considered adequate in view of the other indications to the plant operations personnel that the seal air flask pressure is low.

SR 3.6.5.2.3

The air lock door interlock is designed to prevent simultaneous opening of both doors in the air lock. Since both the inner and outer doors of the air lock are designed to withstand the maximum expected post accident drywell

(continued)

1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.2.3 (continued)

pressure, closure of either door will support drywell OPERABILITY. Thus, the door interlock feature supports drywell OPERABILITY while the air lock is being used for personnel transit in and out of the drywell. 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 drywell is entered, this test is only required to be performed once every [18] months. 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, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

The Surveillance is modified by a Note requiring the Surveillance to be performed only upon entry into the drywell.

SR 3.6.5.2.4

This SR requires a test to be performed to verify overall air lock leakage of the drywell air lock at pressures ≥ [11.5] psig. 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, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR has been modified by a Note indicating that an inoperable air lock door does not invalidate the previous successful performance of an overall air lock leakage test. This is considered reasonable, since either air lock door is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.2.4 (continued)

capable of providing a fission product barrier in the event of a DBA.

SR 3.6.5.2.5

This SR ensures that the drywell air lock seal pneumatic system pressure does not decay at an unacceptable rate. The air lock seal will support drywell OPERABILITY down to a pneumatic pressure of [90] psig. Since the air lock seal air flask pressure is verified in SR 3.6.5.2.2 to be ≥ [90] psig, a decay rate ≤ [30] psig over [10] days is acceptable. The [10] day interval is based on engineering judgment, considering that there is no postulated DBA where the drywell is still pressurized [10] days after the event.

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

REFERENCES

1. 10 CFR 50, Appendix J.
2. FSAR, Chapters [6 and 15].

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ISTS BASES: 3.6.5.2 - DRYWELL AIR LOCK

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

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5.3 Drywell Isolation Valve[s]

BASES

BACKGROUND

The drywell isolation valves, in combination with other accident mitigation systems, function to ensure that steam and water releases to the drywell are channeled to the suppression pool to maintain the pressure suppression function of the drywell.

The OPERABILITY requirements for drywell isolation valves help ensure that valves are closed, when required, and isolation occurs within the time limits specified for those isolation valves designed to close automatically. Therefore, the OPERABILITY requirements support minimizing drywell bypass leakage assumed in the safety analysis (Ref. 1) for a DBA. 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, 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 so that no credible single failure or malfunction of an active component can result in a loss of isolation.

The Drywell Vacuum Relief System valves serve a dual function, one of which is drywell isolation. However, since the other safety function of vacuum relief would not be available if the normal drywell isolation valve actions were taken, the drywell isolation valve OPERABILITY requirements are not applicable to the Drywell Vacuum Relief System isolation valves. Similar surveillance requirements in the LCO for Drywell Vacuum Relief System provide assurance that the isolation capability is available without conflicting with the vacuum relief function.

The Drywell Vent and Purge System is a high capacity system with a [20] inch line, which has isolation valves covered by this LCO. The system supplies filtered outside air directly to the drywell through two lines, each containing two primary containment isolation valves (PCIVs) and two drywell isolation valves called drywell purge isolation valves. The

(continued)

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BASES

BACKGROUND (continued)

drywell air is exhausted through a line also containing two drywell purge isolation valves by means of two fan units, which are part of the Containment Cooling System charcoal filter trains located inside primary containment. After the air is conditioned and filtered, it is exhausted through two PCIVs. The system is used to remove trace radioactive airborne products prior to personnel entry. The Drywell Vent and Purge System is seldom used in MODE 1, 2, or 3; therefore, the drywell purge isolation valves are seldom open during power operation.

The drywell purge isolation valves fail closed on loss of instrument air or power. The drywell purge isolation valves are fast closing valves (approximately [4] seconds). These valves are qualified to close against the differential pressure induced by a loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

This LCO is intended to ensure that releases from the core do not bypass the suppression pool so that the pressure suppression capability of the drywell is maintained. Therefore, as part of the drywell boundary, drywell isolation valve OPERABILITY minimizes drywell bypass leakage. Therefore, the safety analysis of any event requiring isolation of the drywell is applicable to this LCO.

The DBA resulting in a release of steam, water, or radioactive material within the drywell is a LOCA. In the analysis for these accidents, it is assumed that drywell isolation valves either are closed or function to close within the required isolation time following event initiation. Analyses (Ref. 1) also assumed a 4 second drywell purge isolation valve closure time following a 1 second delay prior to closure.

The drywell isolation valves and drywell purge isolation valves satisfy Criterion 3 of the NRC Policy Statement.

LCO

The drywell isolation valve safety function is to form a part of the drywell boundary.

(continued)

BASES

LCO (continued)

The drywell isolation valves are required to have isolation times of automatic drywell isolation valves within limits, automatic drywell isolation valves actuate on an automatic isolation signal, drywell isolation manual valves closed, purge valves closed, and 20 inch purge valves blocked to restrict maximum valve opening. While the Drywell Vacuum Relief System valves isolate drywell penetrations, they are excluded from this Specification. Controls on their isolation function are adequately addressed in LCO 3.6.5.6, "Drywell Vacuum Relief System." The valves covered by this LCO are included (with their associated stroke time for automatic valves) in Reference 2.

The normally closed isolation valves or blind flanges 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 (including check valves with flow through the valve secured), and blind flanges are in place. These passive isolation valves and devices are those listed in Reference 2.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to 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. Therefore, the drywell isolation valves are not required to be OPERABLE in MODES 4 and 5.

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 valve. In this way, the penetration can be rapidly isolated when a need for drywell isolation is indicated.

The second Note provides clarification that for the purpose of this LCO separate Condition entry is allowed for each penetration flow path.

(continued)

BASES

ACTIONS (continued)

The third Note requires the OPERABILITY of affected systems to be evaluated when a drywell isolation valve is inoperable. This ensures appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by an inoperable drywell isolation valve.

The fourth Note ensures appropriate remedial actions are taken when the drywell bypass leakage limits are exceeded. Pursuant to LCO 3.0.6, these ACTIONS are not required even when the associated LCO is not met. Therefore, Note 4 is added to require the proper actions be taken.

A.1 and A.2

With one or more penetration flow paths with one drywell isolation valve inoperable, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic drywell isolation valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. In this Condition, the remaining OPERABLE drywell isolation valve is adequate to perform the isolation function. However, the overall reliability is reduced because a single failure in the OPERABLE drywell isolation valve could result in a loss of drywell isolation. The 8 hour Completion Time is acceptable, since the drywell design bypass leakage $A/\sqrt{k}$ of [1.0] ft^2 would be maintained even with a single failure due to application of ACTIONS Note 4. In addition, the Completion Time is reasonable, considering the time required to isolate the penetration and the relative importance of supporting drywell OPERABILITY during MODES 1, 2, and 3.

For affected penetration flow paths that have been isolated in accordance with Required Action A.1, the affected penetrations must be verified to be isolated on a periodic basis. This is necessary to ensure that drywell penetrations that are required to be isolated following an accident, and 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 device manipulation; rather, it involves verification that those devices outside drywell and capable of potentially

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

being mispositioned are in the correct position. Since these devices are inside primary 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 devices and other administrative controls that will ensure that device misalignment is an unlikely possibility. Also, this Completion Time is consistent with the Completion Time specified for PCIVs in LCO 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)."

Required Action A.2 is modified by a Note that applies to isolation devices 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. Therefore, the probability of misalignment once they have been verified to be in the proper position, is low.

B.1

With one or more penetration flow paths with two drywell isolation valves inoperable, the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, a blind flange, and a check valve with flow through the valve secured. The 4 hour Completion Time is acceptable, since the drywell design bypass leakage $A/\sqrt{k}$ of [1.0] ft^2 is maintained due to application of ACTIONS Note 4. The Completion Time is reasonable, considering the time required to isolate the penetration, and the probability of a DBA, which requires the drywell isolation valves to close, occurring during this short time is very low.

Condition B is modified by a Note indicating this Condition is only applicable to penetration flow paths with two isolation valves. For penetration flow paths with one

(continued)

[1]

BASES

ACTIONS

B.1 (continued)

drywell isolation valve, Condition A provides the appropriate Required Actions.

C.1 and C.2

If any Required Action and associated Completion Time cannot be met, the plant must be placed in 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 3.6.5.3.1

Each [ ] inch drywell purge isolation valve is required to be verified sealed closed at 31 day intervals. This Surveillance is intended to be used for drywell purge isolation valves that are not qualified to open under accident conditions. This SR is designed to ensure that a gross breach of drywell is not caused by an inadvertent or spurious drywell purge isolation valve opening. Detailed analysis of these [ ] inch drywell purge valves failed to conclusively demonstrate their ability to close during a LOCA in time to support drywell OPERABILITY. Therefore, these valves are required to be in sealed closed position during MODES 1, 2, and 3. These [ ] inch drywell 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 leakage within limits. The Frequency is a result of the NRC resolution of Generic Issue B-24 (Ref. 3) related to purge valve use during unit operations.]

(continued)

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.5.3.2

This SR ensures that the [20] inch drywell purge isolation valves are closed as required or, if open, open for an allowable reason. This SR is intended to be used for drywell purge isolation valves that are fully qualified to close under accident conditions; therefore, these valves are allowed to be open for limited periods of time. This SR has been modified by a Note indicating the SR is not required to be met when the drywell purge supply or exhaust valves are open for pressure control, ALARA or air quality considerations for personnel entry, or surveillances that require the valve to be open [provided the [20] inch containment [purge system supply and exhaust] lines are isolated]. The 31 day Frequency is consistent with the valve requirements discussed under SR 3.6.5.3.1.

SR 3.6.5.3.3

This SR requires verification that each drywell isolation manual valve and blind flange that is required to be closed during accident conditions is closed. The SR helps to ensure that drywell bypass leakage is maintained to a minimum. Since these valves are inside primary containment, the Frequency specified as "prior to entering MODE 2 or 3 from MODE 4, if not performed in the previous 92 days," is appropriate because of the inaccessibility of the drywell isolation valves and because these drywell isolation valves are operated under administrative controls and the probability of their misalignment is low.

A Note has been included to clarify that valves that are open under administrative controls are not required to meet the SR during the time the valves are open.

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SR 3.6.5.3.4

Verifying that the isolation time of each power operated, and each automatic drywell isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analysis. The isolation time and Frequency of this SR are [in accordance with the Inservice Testing Program or 92 days].

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.5.3.5

Verifying that each automatic drywell isolation valve closes on a drywell isolation signal is required to prevent bypass leakage from the drywell following a DBA. This SR ensures each automatic drywell isolation valve will actuate to its isolation position on a drywell isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.1.6 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, since isolation of penetrations would eliminate cooling water flow and disrupt the normal operation of many critical components. Operating experience has shown 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 3.6.5.3.6

Verifying that each [ ] inch drywell purge valve is blocked to restrict opening to > [50]% is required to ensure that the valves can be closed under DBA conditions within the time limits assumed in the safety analyses.

The [18] month Frequency is appropriate because the blocking devices are typically removed only during a refueling outage.

REFERENCES

1. FSAR, Section [6.2.4].
2. FSAR, Table [6.2-44].
3. Generic Issue B-24.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ISTS BASES: 3.6.5.3 - DRYWELL ISOLATION VALVE[S]

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

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B 3.6 CONTAINMENT SYSTEMS

B 3.6.5.4 Drywell Pressure

BASES

BACKGROUND

Drywell-to-primary containment differential pressure is an assumed initial condition in the analyses that determine the primary containment thermal hydraulic and dynamic loads during a postulated loss of coolant accident (LOCA).

If drywell pressure is less than the primary containment airspace pressure, the water level in the weir annulus will increase and, consequently, the liquid inertia above the top vent will increase. This will cause top vent clearing during a postulated LOCA to be delayed, and that would increase the peak drywell pressure. In addition, an inadvertent upper pool dump occurring with a negative drywell-to-primary containment differential pressure could result in overflow over the weir wall.

The limitation on negative drywell-to-primary containment differential pressure ensures that changes in calculated peak LOCA drywell pressures due to differences in water level of the suppression pool and the drywell weir annulus are negligible. It also ensures that the possibility of weir wall overflow after an inadvertent pool dump is minimized. The limitation on positive drywell-to-primary containment differential pressure helps ensure that the horizontal vents are not cleared with normal weir annulus water level.

APPLICABLE SAFETY ANALYSES

Primary containment performance is evaluated for the entire spectrum of break sizes for postulated LOCAs. Among the inputs to the design basis analysis is the initial drywell internal pressure (Ref. 1). The initial drywell internal pressure affects the drywell pressure response to a LOCA (Ref. 1) and the suppression pool swell load definition (Ref. 2).

Additional analyses (Refs. 3 and 4) have been performed to show that if initial drywell pressure does not exceed the negative pressure limit, the suppression pool swell and vent clearing loads will not be significantly increased and the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	probability of weir wall overflow is minimized after an inadvertent upper pool dump. Drywell pressure satisfies Criterion 2 of the NRC Policy Statement.
LCO	A limitation on the drywell-to-primary containment differential pressure of [≥ -0.26 psid and ≤ 2.0 psid] is required to ensure that suppression pool water is not forced over the weir wall, vent clearing does not occur during normal operation, containment conditions are consistent with the safety analyses, and LOCA drywell pressures and pool swell loads are within design values.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to the 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 the drywell-to-primary containment differential pressure limitation is not required in MODE 4 or 5.
ACTIONS	<u>A.1</u> With drywell-to-primary containment differential pressure not within the limits of the LCO, it must be restored within 1 hour. The Required Action is necessary to return operation to within the bounds of the safety analyses. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.5.1, "Drywell," which requires that the drywell be restored to OPERABLE status within 1 hour. <u>B.1 and B.2</u> If drywell-to-primary containment differential pressure 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

(continued)

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BASES

ACTIONS

B.1 and B.2 (continued)

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 3.6.5.4.1

This SR provides assurance that the limitations on drywell-to-primary containment differential pressure stated in the LCO are met. The 12 hour Frequency of this SR was developed, based on operating experience related to trending of drywell pressure variations during the applicable MODES and to assessing proximity to the specified LCO pressure limits. 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

1. FSAR, Section [6.2.1].
2. FSAR, Section [3.8].
3. FSAR, Section [6.2.1.1.6].
4. FSAR, Section [6.2.7].

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

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5.5 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 drywell average air temperature affects equipment OPERABILITY, personnel access, and the calculated response to postulated Design Basis Accidents (DBAs). The limitation on drywell average air temperature ensures that the peak drywell temperature during a design basis loss of coolant accident (LOCA) does not exceed the design temperature of [330]°F. The limiting DBA for drywell atmosphere temperature is a small steam line break, assuming no heat transfer to the passive steel and concrete heat sinks in the drywell.

APPLICABLE SAFETY ANALYSES

Primary containment performance for the DBA is evaluated for the entire spectrum of break sizes for postulated LOCAs inside containment (Ref. 1). Among the inputs to the design basis analysis is the initial drywell average air temperature. Increasing the initial drywell average air temperature could change the calculated results of the design bases analysis. The safety analyses (Ref. 1) assume an initial average drywell air temperature of [135]°F. This limitation ensures that the safety analyses remain valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell temperature does not exceed the maximum allowable temperature of [330]°F. The consequence of exceeding this design temperature may result in the degradation of the drywell structure under accident loads. Equipment inside the drywell that is required to mitigate the effects of a DBA is designed and qualified to operate under environmental conditions expected for the accident.

Drywell average air temperature satisfies Criterion 2 of the NRC Policy Statement.

LCO

If the initial drywell average air temperature is less than or equal to the LCO temperature limit, the peak accident temperature can be maintained below the drywell design

(continued)

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BASES

LCO (continued)	temperature during a DBA. This ensures the ability of the drywell to perform its design function.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to the 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	<u>A.1</u> When the drywell average air temperature is not within the limit of the LCO, it must be restored within 8 hours. The Required Action is necessary to return operation to within the bounds of the safety analyses. The 8 hour Completion Time is acceptable, considering the sensitivity of the analyses to variations in this parameter, and provides sufficient time to correct minor problems. <u>B.1 and B.2</u> If drywell average air temperature cannot be restored to within limit 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.
SURVEILLANCE REQUIREMENTS	<u>SR 3.6.5.5.1</u> Verifying that the drywell average air temperature is within the LCO limit ensures that operation remains within the limits assumed for the drywell analysis. Drywell air temperature is monitored in all quadrants and at various elevations. Since the measurements are uniformly

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.5.5.1 (continued)

distributed, an arithmetic average is an accurate representation of actual drywell average temperature.

The 24 hour Frequency of the SR was developed based on operating experience related to variations in drywell average air temperature variations during the applicable MODES. 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

1. FSAR, Section [6.2].

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ISTS BASES: 3.6.5.5 - DRYWELL AIR TEMPERATURE

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

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5.6 Drywell Vacuum Relief System

BASES

BACKGROUND

The Mark III pressure suppression containment is designed to condense, in the suppression pool, the steam released into the drywell in the event of a loss of coolant accident (LOCA). The steam discharging to the pool carries the noncondensibles from the drywell. Therefore, the drywell atmosphere changes from low humidity air to nearly 100% steam (no air) as the event progresses. When the drywell subsequently cools and depressurizes, noncondensibles in the drywell must be replaced to avoid excessive weir wall overflow into the drywell. Rapid weir wall overflow must be controlled in a large break LOCA, so that essential equipment and systems located above the weir wall in the drywell are not subjected to excessive drag and impact loads. The drywell post-LOCA and the drywell purge vacuum relief subsystems are the means by which noncondensibles are transferred from the primary containment back to the drywell.

The vacuum relief systems are a potential source of bypass leakage (i.e., some of the steam released into the drywell from a LOCA bypasses the suppression pool and leaks directly to the primary containment airspace). Since excessive bypass leakage could degrade the pressure suppression function, the Drywell Vacuum Relief System has been designed with at least two valves in series in each vacuum breaker line. This minimizes the potential for a stuck open valve to threaten drywell OPERABILITY. The [two] drywell purge vacuum relief subsystems use separate [10] inch lines penetrating the drywell, and each subsystem consists of a series arrangement of a motor operated isolation valve and two check valves. The [two] drywell post-LOCA vacuum relief subsystems use a common [10] inch line penetrating the drywell, and each subsystem consists of a motor operated valve in series with a check valve. At least two [10] inch lines must be available to provide adequate relief to control rapid weir wall overflow.

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[1]

BASES (continued)

APPLICABLE SAFETY ANALYSES

The Drywell Vacuum Relief System must function in the event of a large break LOCA to control rapid weir wall overflow that could cause drag and impact loadings on essential equipment and systems in the drywell above the weir wall. The Drywell Vacuum Relief System is not required to assist in hydrogen dilution or to protect the structural integrity of the drywell following a large break LOCA. Furthermore, their passive operation (remaining closed and not leaking during drywell pressurization) is implicit in all of the LOCA analyses (Ref. 1).

The Drywell Vacuum Relief System satisfies Criterion 3 of the NRC Policy Statement.

LCO

The LCO ensures that in the event of a LOCA, [two] drywell post-LOCA and [two] drywell purge vacuum relief subsystems are available to mitigate the potential subsequent drywell depressurization. Each vacuum relief subsystem is OPERABLE when capable of opening at the required setpoint but is maintained in the closed position during normal operation.

APPLICABILITY

In MODES 1, 2, and 3, a Design Basis Accident could cause pressurization of primary containment. Therefore, Drywell Vacuum Relief 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 Drywell Vacuum Relief System OPERABLE is not required in MODE 4 or 5.

ACTIONS

The ACTIONS Note ensures appropriate remedial actions are taken when the drywell bypass leakage limits are exceeded. Pursuant to LCO 3.0.6, these ACTIONS are not required even when the associated LCO is not met. Therefore, the Note is added to require the proper actions be taken.

(continued)

BASES

ACTIONS (continued)

A.1

With one or more vacuum relief subsystems open, the subsystem must be closed within 4 hours. This assures that drywell leakage would not result if a postulated LOCA were to occur. The 4 hour Completion Time is acceptable, since the drywell design bypass leakage ($A/\sqrt{K}$) of [1.0] ft^2 is maintained, and is considered a reasonable length of time needed to complete the Required Action.

A Note has been added to provide clarification that separate Condition entry is allowed for vacuum relief subsystems not closed.

B.1 and C.1

With one [or two] drywell post-LOCA vacuum relief subsystems inoperable or one drywell purge vacuum relief subsystem inoperable, for reasons other then being not closed, the inoperable subsystem(s) must be restored to OPERABLE status within 30 days. In these Conditions, the remaining OPERABLE vacuum relief subsystems are adequate to perform the depressurization mitigation function since two [10] inch lines remain available. The 30 day Completion Time takes into account the redundant capability afforded by the remaining subsystems, a reasonable time for repairs, and the low probability of an event requiring the vacuum relief subsystems to function occurring during this period.

D.1 and E.1

With [two] drywell purge vacuum relief subsystems inoperable or with [two] drywell post-LOCA and one drywell purge vacuum relief subsystems inoperable, for reasons other than being not closed, at least one inoperable subsystem must be restored to OPERABLE status within 72 hours. In these Conditions, only one [10] inch line remains available. The 72 hour Completion Time takes into account at least one vacuum relief subsystem is still OPERABLE, a reasonable time for repairs, and the low probability of an event requiring the vacuum relief subsystems to function occurring during this period.

(continued)

BASES

ACTIONS (continued)

F.1, F.2, G.1, and G.2

If the inoperable drywell vacuum relief subsystem(s) cannot be closed or restored to OPERABLE status within the required Completion Time, or if two drywell purge vacuum relief subsystems are inoperable, for reasons other than being not closed, and one or two drywell post-LOCA vacuum relief subsystem(s) are inoperable, for reasons other than being not closed, 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 3.6.5.6.1

Each vacuum breaker and its associated isolation valve is verified to be closed to ensure that this potential large bypass leakage path is not present. This Surveillance is performed by observing the vacuum breaker or associated isolation valve position indication or by verifying that the vacuum breakers are closed when a differential pressure of [1.0] psid between the drywell and primary containment is maintained for 1 hour without makeup. The 7 day Frequency is based on engineering judgment, is considered adequate in view of other indications of vacuum breaker or isolation valve status available to the plant personnel, and has been shown to be acceptable through operating experience.

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

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BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.5.6.2

Each vacuum breaker and its associated isolation valve must be cycled to ensure that it opens adequately to perform its design function and returns to the fully closed position. This provides assurance that the safety analysis assumptions are valid. A 31 day Frequency was chosen to provide additional assurance that the vacuum breakers and their associated isolation valves are OPERABLE.

SR 3.6.5.6.3

Verification of the vacuum breaker opening setpoint is necessary to ensure that the safety analysis assumption that the vacuum breaker will open fully at a differential pressure of [1.0] 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. Operating experience has shown these components usually pass the Surveillance when performed at the [18] month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section [6.2].

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ISTS BASES: 3.6.5.6 - DRYWELL VACUUM RELIEF SYSTEM

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

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.

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-1434, 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.

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 the drywell-to-suppression chamber bypass leakage outside of limits in MODE 1, 2, or 3, the current Technical Specifications do not provide 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 increase 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 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 accelerated test basis and elevated test pressure requirements of CTS 4.6.2.1.d.2. CTS 4.6.2.1.d.2 requires verification of drywell-to-suppression chamber bypass leakage on an accelerated test basis and at a higher test pressure in the event that the results of consecutive drywell-to-suppression chamber bypass leakage tests are outside Technical Specification specified limits. Under the proposed change, drywell-to-suppression chamber will continue to be verified at the frequency and at the test pressure described in CTS 4.6.2.1.d. 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, the proposed change does not significantly increase the probability of such accidents. The proposed change results only in a fixed test frequency and test pressure for drywell-to-suppression chamber bypass leakage testing. The change does not alter leakage limits, and therefore does not alter the consequences of any previously analyzed events. Therefore, this change does not significantly increase the consequences of any previously analyzed accidents.

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

   The 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 test frequency and pressure for drywell-to-suppression chamber leakage testing in the event of consecutive test failures, and does not 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 accident.

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

   The change impacts only the test frequency and pressure for drywell-to-suppression chamber leakage testing in the event of consecutive test failures. The methodology of the accident analysis and limits of the Technical Specifications are not affected, nor is the primary containment response. This change results in a fixed test frequency and test pressure that have been demonstrated to be adequate through the results of previous testing. 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?

   This change allows drywell-to-suppression chamber bypass leakage that is greater than 10% of the acceptable design $A/\sqrt{k}$ limit to be considered acceptable at times other than during the first unit startup following performance of bypass leakage testing in accordance with proposed ITS SR 3.6.1.1.3, provided it is less than or equal to the design $A/\sqrt{k}$ leakage limit. 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 allows continued operation with drywell-to-suppression chamber leakage that is greater than 10% of the acceptable design value, but less than or equal to the design leakage limit. 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.

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

   This change only 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 3.6.1.1.3. 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 calculated $A/\sqrt{k}$ 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.

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?

   This change deletes the requirement associated with CTS 4.6.2.1.d 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.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 the existing license; Specification 3.6.1.1 allows 1 hour to restore losses in containment

L.1 CHANGE

2. (continued)

integrity 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.

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?

   The proposed change would allow the temporary opening of the remaining OPERABLE door for a limited period of time for purposes other than making repairs if one air lock door is 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. The change to allow the temporary opening of the one OPERABLE door for purposes other than 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; 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 the transient of plant shutdown to follow (due to inability to perform preventive or corrective maintenance) 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 the existing license; Specification 3.6.1.1 allows 1 hour to restore losses in containment

L.2 CHANGE

2. (continued)

integrity 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.

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 to allow the temporary opening of the one OPERABLE door for purposes other than 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; 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.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 time to verify an OPERABLE air lock door is closed when the primary containment air lock is inoperable. This change does not affect the air lock design or function and one primary containment air lock door 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 the primary containment air lock is inoperable. This change does not affect the air lock design or function and one primary containment air lock door 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.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 verification that the primary containment air lock door is locked closed to be done by administrative means if the barrier is in a high radiation area or the access to it is limited due to inerting. Neither an open nor an inoperable air lock door 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.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?

   This change would allow continued operation with an inoperable primary containment air lock door interlock mechanism. Having both primary containment air lock doors open at the same time is not an initiator of any previously analyzed accident. Therefore, this change does not significantly increase the probability of any previously analyzed accident. The proposed change provides actions with appropriate compensatory measures to maintain a level of safety equivalent to compliance with the LCO. These actions do not result in air lock 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 in the equipment design or capabilities, but does allow operation of the plant with equipment that is capable of performing its safety function. However, since the change includes compensatory measures which maintain a level of safety equivalent to the capabilities of the equipment, the change does not crease 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 dedicated personnel to provide compensatory actions in place of automatic equipment for a limited time. These administrative controls continue to provide adequate containment should an accident occur. Therefore, the change does not involve a significant reduction in the 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 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.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 1 hour to 4 hours for purge valves, 4 hours to 8 hours for MSIVs, and 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 purge valves, MSIVs, and other PCIVs isolate to control leakage from the primary containment during accidents. Allowing the additional time to isolate the purge valves, MSIVs, and other PCIVs will not significantly increase the consequences of an accident. The consequences will be the same for the proposed times as for the current times. The additional times, however, will allow more time to repair the inoperable purge valve, MSIV, or other 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 for other PCIVs 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 purge valves, MSIVs, and other 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 1 hour to 4 hours for purge valves, 4 hours to 8 hours for MSIVs, and 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,

L.1 CHANGE

3. (continued)

for purge valve and MSIV penetrations, another purge valve or MSIV, as applicable, in the penetration flow path remains Operable and capable of isolating the penetrations, and for the other PCIVs, 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 purge valve, MSIV, and other PCIVs.

Isolating the MSIV penetrations will require a reduction in power and has the potential for tripping the plant. A reduction in power or a plant trip is considered a transient due to the thermal effects it has on plant equipment. During the additional time allowed, a limiting event would still be assumed to be within the bounds of the safety analysis, assuming no single active failure. 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?

   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 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 with an inoperable excess flow check valve is bounded by the leakage through the instrument line flow restricting orifice located inside containment, and is significantly below 10 CFR 100 release limits. 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.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?

   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.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 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.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?

   This change would remove a specific restriction to perform a surveillance, which requires closure of the primary containment isolation valves, during Cold Shutdown or Refueling. The change will allow the surveillance to be performed while operating in MODE 1, 2, or 3. 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 appropriate plant conditions for performance of the surveillance will continue to be controlled in plant procedures to assure the potential consequences are not significantly increased. This control method has been previously determined to be acceptable as indicated in Generic Letter 91-04. 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 removes a specific restriction on the plant conditions for performing a surveillance, but does not change the method of performance. The appropriate plant conditions for performance of the surveillance will continue to be controlled in plant procedures to assure the possibility for a new or different kind of accident are not created. This control method has been previously determined to be acceptable as indicated in Generic Letter 91-04. 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 appropriate plant conditions for performing the surveillance will continue to be controlled in plant procedures to assure that there is no significant reduction. This control method has been previously determined to be acceptable as indicated in Generic Letter 91-04. Therefore, the change does not involve a significant reduction in the 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?

   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.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 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. 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 LaSalle 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 orificed 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.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 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.

L.11 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.11 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.12 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 reasons that the large primary containment purge and exhaust isolation valves may be opened are not assumed in the initiation of any analyzed event. Expanding the reasons these valves may be opened and removing the 90 hour limit for having these valves open does not affect any assumptions of the accident analyses and still ensures the time period these valves may be opened in MODES 1, 2, and 3 is limited. In addition, these purge and exhaust valves are capable of closing in the environment following a design basis accident. Thus, the consequences of an accident are not affected by this change. This change will not alter assumptions relative to 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 proposed change will not involve any physical change to plant systems, structures, or components (SSC), or the manner in which these SSCs are maintained, modified, tested, or inspected. The change in methods governing normal plant operation is consistent with the current safety analysis assumptions. 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?

   The proposed change expands the reasons the primary containment purge and exhaust isolation valves may be opened in MODES 1, 2, and 3. This change does not involve a reduction in the margin of safety since these valves are capable of closing in the environment following a design basis accident and the accumulated time a purge or exhaust valve flow path exists will continue to be limited. This change does not affect the current safety analysis assumptions. As such, no question of safety exists. Therefore, this change does not involve a significant reduction in a margin of safety.

L.13 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 restore leakage of hydrostatically tested valves and MSIVs. The extension is from the current 4 hour or shutdown requirement (depending on the number of valves in the penetration affected) to 4 hours for valves in hydrostatically tested lines not on a closed system, 8 hours for MSIVs, and 72 hours for valves in hydrostatically tested lines on a closed system. The PCIV leakage is not assumed to be an initiator of any analyzed event. Therefore, this change will not involve an increase in the probability of an accident previously evaluated. Allowing additional time to restore leakage will not significantly increase the consequences of an accident. The consequences will be the same for the proposed times as with the current action requirements. The additional times, however, will allow more time to repair the inoperable valves and possibly avoid a shutdown. Shutting down the unit is a transient which puts thermal stress on components which could increase the chances of challenging safety systems. 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. This change will still require the leakage values to be restored within limits. 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 relaxes the allowed restoration time for restoring the leakage to within limits. The margin of safety is not significantly reduced because another Operable valves remains to isolate the flow path, the system is a closed system, or the line is hydrostatically sealed. The additional times, however, will allow more time to repair the inoperable valves and possibly avoid a shutdown. Shutting down the unit is a transient which puts thermal stress on components which could increase the chances of challenging safety systems. Therefore, this change does not involve a significant reduction in a margin of safety.

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 - 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?

   This change extends the frequency of vacuum breaker functional test (cycling) from monthly to quarterly. 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, are not in a harsh environment, and cycle properly when tested, 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 cycle the vacuum breakers. Operational history has shown these vacuum breakers are normally closed and cycle properly when tested. These vacuum breakers are not in a harsh environment, and other valves in similar environments are tested at these Frequencies (or less frequently).

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 C 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 C 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 96 hours to restore one loop of suppression pool cooling when it is found to be inoperable and 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 96 hour extension is based on similar current allowed outage times for emergency core cooling systems equipment. The proposed 8 hour extension is based on similar allowed outage times for 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: 3.6.3.1 - PRIMARY CONTAINMENT HYDROGEN RECOMBINERS

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?

   Mode changes are proposed to be allowed with one containment hydrogen recombiner inoperable. The hydrogen recombiners are not considered as initiators for any previously evaluated accidents. Therefore, the probability of an accident previously evaluated is not significantly increased. A second containment hydrogen recombiner remains OPERABLE and the recombiners are backed up by the availability of the nitrogen inerting and purge system. Each of these are adequate to perform the safety function required for each previously evaluated accident. Therefore, the consequences of 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?

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Since Mode changes do not involve any manipulation of the hydrogen recombiners, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

   The margin of safety for the recombiners is based on the capacity and redundancy of the recombiners. Since the capacity is not changed and the recombiners are backed by another hydrogen control method, the capability for adequate response to the need for the hydrogen control function is maintained. Therefore, the 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?

   Continued operation is proposed to be allowed for a limited time with both primary containment hydrogen recombiners inoperable. The hydrogen recombiners are not considered as initiators for any previously evaluated accidents. Therefore, the probability of an accident previously evaluated is not significantly increased. The hydrogen recombiners are backed up by the nitrogen inerting and purge system, which is adequate to perform the safety function required for the previously evaluated accident. Therefore, the consequences of 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?

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Since normal operation of the plant does not involve any manipulation of the hydrogen recombiner system, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

   The margin of safety for this system is based on the capacity and redundancy of the system. Since the capacity is not changed and the system is backed by another method to control hydrogen, the capability for adequate response to the need for the hydrogen control function is maintained. Additionally, the proposed change will prevent unnecessary shutdowns and the associated risk of potential transients. Therefore, the change does not involve a significant reduction in a 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 proposed change would eliminate a redundant, more frequent functional test of the hydrogen recombiner system. The hydrogen recombiners are not considered as initiators for any previously evaluated accidents. Therefore, the probability of an accident previously evaluated is not significantly increased. The proposed change does not impact the system design or operation, or its ability to accomplish its safety function. 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 changes in plant operation. The system will continue to function in the same way as before the change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

   The design, function, and OPERABILITY requirements for the hydrogen recombiner system are unchanged with this proposed revision. The system must continue to be capable of performing its function. In addition, this change is supported by the NRC in Generic Letter 93-05, item 8.5. Therefore, the proposed 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 hydrogen recombiner instrumentation is not assumed in the initiation of any analyzed event. The requirements for the hydrogen recombiner instrumentation do not need to be explicitly stated in the Technical Specifications. To perform the hydrogen recombiner system functional test required by SR 3.6.3.1.1, the hydrogen recombiner instrumentation must be OPERABLE. If the hydrogen recombiner instrumentation is inoperable, this functional test cannot be satisfied and the appropriate actions must be taken for an inoperable hydrogen recombiner in accordance with the ACTIONS of ITS 3.6.3.1. 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 hydrogen recombiner instrumentation requirements from Technical Specifications does not impact any margin of safety. The requirements for hydrogen recombiner instrumentation do not need to be explicitly stated in the Technical Specifications. To perform the hydrogen recombiner system functional test required by SR 3.6.3.1.1, the hydrogen recombiner instrumentation must be OPERABLE. If the hydrogen recombiner instrumentation is inoperable, this functional test cannot be satisfied and the appropriate actions must be taken for an inoperable hydrogen recombiner in accordance with the ACTIONS of ITS 3.6.3.1. As a result, the OPERABILITY of the hydrogen recombiner instrumentation will be maintained to satisfy the associated SR of ITS 3.6.3.1 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.

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.6.4.1 - SECONDARY CONTAINMENT

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.6.4.2 - SECONDARY CONTAINMENT ISOLATION VALVES (SCIVs)

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?

   This change would remove a specific restriction to perform a surveillance of the secondary containment isolation valves during Cold Shutdown or Refueling. The change will allow the Surveillance to be performed while operating in MODE 1, 2, or 3. 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 appropriate plant conditions for performance of the surveillance will continue to be controlled in plant procedures to assure the potential consequences are not significantly increased. This control method has been previously determined to be acceptable as indicated in Generic Letter 91-04. 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 removes a specific restriction on the plant conditions for performing a surveillance, but does not change the method of performance. The appropriate plant conditions for performance of the surveillance will continue to be controlled in plant procedures to assure the possibility for a new or different kind of accident are not created. This control method has been previously determined to be acceptable as indicated in Generic Letter 91-04. 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 appropriate plant conditions for performing the surveillance will continue to be controlled in plant procedures to assure that there is no significant reduction. This control method has been previously determined to be acceptable as indicated in Generic Letter 91-04. Therefore, 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 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.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?

   This change will allow the verification of closure of isolation devices such as valves and blind flanges located in high radiation areas be performed by the use of administrative means. The entry into these high radiation areas are 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. 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 and these additional administrative controls continue to provide adequate containment should an accident occur. Therefore, the change does not involve a significant reduction in the 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 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.7 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.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.6.4.3 - STANDBY GAS TREATMENT (SGT) SYSTEM

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 based on the following:

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 based on the following:

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.