MARK UP OF ITS CHANGE

Conforming Changes to bring ITS Submittal up to date with current Zion license.

Implement PTLR, Amendments 177/164. Implement 10 CFR , Appendix J, Option B, to conform to Amendments 175/162. Revise the definition of F* Distance in Steam Generator Tube Inspection Program discussion, 5.5.7, to conform to Amendments 174/161. Conforming changes to update ITS to amendment 173/160, incorporation of specific sections of CEN-629-P, Revision 00, ABB/CE Topical Report on welded sleeve process.

BASES

SR 3.0.2 (continued)

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is a Surveillance with a Frequency of "in accordance with **the Containment Leakage Rate Testing Program**~~10 CFR 50, Appendix J, as modified by approved exemptions~~." The requirements of regulations take precedence over the TS. The TS cannot in and of themselves extend a test interval specified in the regulations. Therefore, there is a Note in the Frequency stating "SR 3.0.2 is not applicable."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time which requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals other than those consistent with refueling outages or periodic Completion Time intervals beyond those specified.

(continued)

5.5 Programs and Manuals

5.5.7 Steam Generator (SG) Tube Surveillance Program (continued)

"Defect" means an imperfection of such severity that it exceeds the Plugging or Repair Limit, where applicable. A tube containing a defect is defective.

"F* distance" is the length of undegraded tube required to resist pullout. This distance is measured from the bottom of the upper hard roll transition toward the bottom of the tube sheet and has been conservatively determined to be 1.05 inches. ~~(not including eddy current uncertainties). F* is only valid for Z1R14 and through Z1C15.~~ The determination of F* Distance is satisfied by the mechanical tool (rolling pin) used to install the joint having an effective length of greater than or equal to 1.05 inches. The F* Distance is applicable only to tubes that are rerolled below the mid-plane of the tubesheet and have no degradation in the rerolled expansion joints. The F* Distance is not applicable to the original factory hard roll. 174/161

"F* tube" means a tube with indications below the F* distance greater than or equal to the Plugging or Repair Limit, and no indications within the F* distance.

"Imperfection" means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy current testing indications < 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

"Plugging or Repair Limit" means the imperfection depth at or beyond which the tube shall be repaired or removed from service by plugging because it may become unserviceable prior to the next inspection.

"Repaired Tube" means a tube that has undergone a process that re-establishes its serviceability. The sleeving process utilized shall be one of the following:

1. The Combustion Engineering welded sleeve process as described in Report CEN-331-P, Revision 1-P. The following sections of ABB/CE Report CEN-629-P, Revision 00, shall be implemented in conjunction with Report CEN-331-P, Revision 1-P: 173/160

(continued)

5.5 Programs and Manuals

Section 4.5.3 - Tube Brushing-Cleaning Equipment
Section 4.5.7 - Nondestructive Examination
Section 5.0 - Sleeve Examination Program
Section 9.3.1 - Cleaning Qualification

2. The Westinghouse Electric Corporation mechanical sleeve process as described in Report WCAP-11669.

(continued)

5.5 Programs and Manuals

5.5.14 Containment Leakage Rate Testing Program — 175/162

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995, with the following exception: Section 10.2.1 of NEI 94-01 Revision 0, is not applicable to Unit 1 penetration P-16. This exception for penetration P-16 shall expire at the completion of Unit 1 outage Z1R15.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 46.79 psig. The containment design pressure is 47 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be $\leq$ 0.1% of containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are < 0.60 L_a for the Type B and C tests and < 0.75 L_a for Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage is < 0.6 L_a when combined with all Type B and C test results at a test pressure of $\geq P_a$.

   2) For each door, leakage rate is $\leq$ 1.0 SCFH when tested at $\geq$ 2.5 psig and < 10 psig, or $\leq$ 4.75 SCFH when tested at $\geq$ 10 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.6 Reporting Requirements (continued)

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing, as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits";
LCO 3.4.10, "Pressurizer Safety Valves"; and
LCO 3.4.12, "Low Temperature Overpressure Protection."

b. **The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those identified in the following documents: License Amendment number 177 for Unit 1 and 164 for Unit 2.** ~~The RCS pressure and temperature limits for Units 1 and 2 will be those previously reviewed and approved in amendments 101 and 91 respectively. The Cold Overpressure Protection setpoint and arming temperature and RCS Safety Valve applicability for Units 1 and 2 will be those previously reviewed and approved in amendments 54 and 51 respectively, and amendments 153 and 141 respectively.~~

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluency period and for any revision or supplement thereto. ~~The first revision to the limits after initial implementation of the PTLR will be submitted to the NRC for review prior to approval. Subsequent revisions will be made without prior approval.~~

5.6.7 Post Accident Monitor Report

When a report is required by LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.1 Containment

LCO 3.6.1 Containment shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment inoperable.	A.1 Restore containment to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	-------------------NOTE------------------- SR 3.0.2 is not applicable. -- Perform required visual examinations and leakage rate testing, except for containment air lock testing, in accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~the 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 Type B and Type C tests, and < 0.75 L_a for the Type A tests.~~	In accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~the Containment Leakage Rate Testing Program
SR 3.6.1.2	Verify containment structural integrity in accordance with the Containment Tendon Surveillance Program.	In accordance with the Containment Tendon Surveillance Program

3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Air Locks

LCO 3.6.2 Two containment air locks shall be OPERABLE.

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

ACTIONS

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

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment air locks with one containment air lock door inoperable.	------------NOTES------------ 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls if both air locks are inoperable. ----------------------------	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.1	Verify the OPERABLE door is closed in the affected air lock.	1 hour
	<u>AND</u>		
	A.2	Lock the OPERABLE door closed in the affected air lock.	24 hours
	<u>AND</u>		
	A.3	Verify the OPERABLE door is locked closed in the affected air lock.	Once per 31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more containment air locks with containment air lock interlock mechanism inoperable.	------------NOTES------------ 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit of containment is permissible under the control of a dedicated individual. -----------------------------	
	B.1 Verify an OPERABLE door is closed in the affected air lock.	1 hour
	<u>AND</u>	
	B.2 Lock an OPERABLE door closed in the affected air lock.	24 hours
	<u>AND</u>	
	B.3 Verify an OPERABLE door is locked closed in the affected air lock.	Once per 31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more containment air locks inoperable for reasons other than Condition A or B.	C.1	Initiate action to evaluate overall containment leakage rate per LCO 3.6.1.	Immediately
	AND		
	C.2	Verify a door is closed in the affected air lock.	1 hour
	AND		
	C.3	Restore air lock to OPERABLE status.	24 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	-------------------NOTES------------------- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria ~~of SR 3.6.1.1 in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions~~ applicable to SR 3.6.1.1. ~~3. SR 3.0.2 is not applicable.~~ --- Perform required air lock leakage rate testing in accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~ the Containment Leakage Rate Testing Program. ~~The acceptance criteria for air lock testing are:~~ ~~a. Overall air lock leakage rate, when combined with other Type B and Type C test results, is $\leq 0.6\ L_a$ when tested at $\geq P_a$.~~ ~~b. For each door with testable seals, the leakage rate is ≤ 1.0 SCFH when tested at ≥ 2.5 psig and $<$ 10 psig, or ≤ 4.75 SCFH when tested at ≥ 10 psig.~~	In accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~ the Containment Leakage Rate Testing Program
SR 3.6.2.2	-------------------NOTE------------------- Only required to be performed upon entry or exit through the containment air lock. --- Verify only one door in the air lock can be opened at a time.	184 days

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of the concrete reactor building, its steel liner, and the penetrations through this structure which encloses the entire reactor and Reactor Coolant System. The structure is designed to contain radioactive material that may be released from the reactor core following a ~~D~~design ~~B~~basis Loss of Coolant Accident ~~(DBA)~~. In addition, this structure provides shielding from the fission products that may be present in the containment atmosphere from both normal and post accident conditions (Ref. 1).

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

The cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions, and the dome roof is prestressed utilizing a three way post tensioning system.

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

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

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

   1. capable of being closed by an OPERABLE automatic containment isolation signal in MODES 1, 2 and 3,

(continued)

BASES

BACKGROUND (continued)

2. capable of being closed by an OPERABLE manual containment isolation signal in MODE 4, or

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

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks."

c. The equipment hatch is installed.

APPLICABLE SAFETY ANALYSES

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

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a Loss of Coolant Accident (LOCA) and a main steam line break (Ref. 2). In addition, the release of significant fission product radioactivity within containment can occur from a LOCA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 1). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B, as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA ~~DBA~~. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.1% per day in the safety analysis at P_a = 46.79 psig (Ref. 2).

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, except prior to the first startup after performing a required Containment Leakage Rate Testing Program ~~10 CFR 50, Appendix J,~~ leakage test. At this time the leakage limits specified in the Containment Leakage Rate Testing Program must be met, i.e., the combined Type B and C leakage must be $< 0.6\ L_a$, and the overall Type A leakage must be $< 0.75\ L_a$.

Compliance with this LCO will ensure a containment configuration, including both air locks and the equipment hatch, that is structurally sound and will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates ~~specified~~ for the containment air lock~~s~~ door seals ~~(LCO 3.6.2)~~ are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J, Option B. Therefore, exceeding the individual air lock door seal leakage rate limits specified in the Containment Leakage Rate Testing Program only results in the containment being inoperable when the combined Type B and C leakage rate exceeds the overall acceptance criteria ~~of Appendix J~~ specified in that program.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

A.1

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

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of ~~10 CFR 50, Appendix J, as modified by approved exemptions~~ **the Containment Leakage Rate Testing Program**.

Failure to meet individual air lock door seal leakage limits specified in **the Containment Leakage Rate Testing Program** ~~LCO 3.6.2~~ does not invalidate the acceptability of these overall leakage determinations unless their contribution to the overall Type A, B, and C leakage causes that limit to be exceeded.

As left leakage prior to the first startup after performing a required ~~10 CFR 50, Appendix J,~~ **Containment Leakage Rate Testing Program** leakage test is required to be < 0.6 L_a for combined Type B and Type C leakage. ~~, and~~

As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be < 0.75 L_a for overall Type A leakage.

At all other times between required leakage rate tests, the acceptance criteria for Type B and Type C leakage tests are 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1 (continued)

SR Frequencies are as required by ~~Appendix J, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply since the Technical Specifications can not override the Frequency required by the regulations~~ the Containment Leakage Rate Testing Program. These periodic testing requirements verifv that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.2

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of Specification 5.5.5, "Containment Tendon Surveillance Program." Testing and Frequency are in general conformance with the recommendations of Regulatory Guide 1.35 (Ref. 3).

REFERENCES

1. UFSAR, Section 3.8.
2. UFSAR, Chapter 15.
3. Regulatory Guide 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures," Revision 2, January 1976.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

Two air locks are provided. The personnel air lock, used for normal ingress and egress of containment, and the emergency air lock typically reserved for entering and exiting containment in emergency situations. The personnel air lock penetrates the equipment hatch. The emergency air lock is a right circular cylinder located on the opposite side of containment at ground level. The doors in each air lock are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each of the air lock doors contains double gasketed seals and local leakage rate testing capability to ensure seal integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door). 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) in containment. As such, closure of a single air lock door supports containment OPERABILITY.

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

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident and a rod ejection accident (Ref. 1). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 1). ~~This leakage rate is defined in 10 CFR 50, Appendix J, as L_a = 0.1% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a = 46.79 psig following a DBA.~~ This leakage rate is defined in 10 CFR 50, Appendix J, Option B, as the maximum allowable containment leakage rate at the calculated peak containment internal pressure, P_a, 46.79 psig, following a design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of the NRC Policy Statement.

LCO

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

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

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

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

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock.

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

A.1, A.2 and A.3

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which

(continued)

ACTIONS

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

requires 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 reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls.

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 same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1, C.2 and C.3 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. The 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required on a periodic basis to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment.

This Note is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment is 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

(continued)

BASES

ACTIONS

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

pressurize the containment during the short time that the OPERABLE door is expected to be open.

B.1, B.2 and B.3

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

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1, C.2 and C.3 are the appropriate remedial actions. Note 2 allows entry into and exit from containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

C.1, C.2 and C.3

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

(continued)

BASES

ACTIONS

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

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

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

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of ~~10 CFR 50, Appendix J, as modified by approved exemptions~~ the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The periodic testing requirements verify that the air lock leakage, when combined with other Type B and C test results, does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by ~~Appendix J, as modified by approved exemptions~~ the Containment Leakage Rate Testing Program. The testing requirements for the individual doors provide early indication of potential degradation that may affect the ability to meet overall Type B leakage criteria.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1 (continued)

The SR has been modified by ~~three~~ 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. This ensures that air lock leakage is properly accounted for in determining the ~~overall~~ **combined Type B and C** containment leakage rate. ~~Note 3 states that SR 3.0.2 is not applicable since Technical specifications can not override the Frequency required by regulations. Thus, a surveillance Frequency extension is not permitted~~.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is only challenged when the containment air lock door is opened, this test is only required to be performed upon entering or exiting a containment air lock but is not required more frequently than every 184 days. Therefore, performance of the SR may be extended beyond 184 days (plus the allowance specified in SR 3.0.2) without violating the requirements of SR 3.0.1. The 184 day Frequency is based on engineering judgment and is considered adequate in view of other indications of door and interlock mechanism status available to operations personnel.

(continued)

BASES (continued)

REFERENCES 1. UFSAR, Chapter 15.

CLEAN ITS SPEC

SR 3.0.2 (continued)

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is a Surveillance with a Frequency of "in accordance with **the Containment Leakage Rate Testing Program**." The requirements of regulations take precedence over the TS. The TS cannot in and of themselves extend a test interval specified in the regulations. Therefore, there is a Note in the Frequency stating "SR 3.0.2 is not applicable."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time which requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals other than those consistent with refueling outages or periodic Completion Time intervals beyond those specified.

(continued)

5.5 Programs and Manuals

5.5.7 Steam Generator (SG) Tube Surveillance Program (continued)

"Defect" means an imperfection of such severity that it exceeds the Plugging or Repair Limit, where applicable. A tube containing a defect is defective.

"F* distance" is the length of undegraded tube required to resist pullout. This distance is measured from the bottom of the upper hard roll transition toward the bottom of the tube sheet and has been conservatively determined to be 1.05 inches. The determination of F* Distance is satisfied by the mechanical tool (rolling pin) used to install the joint having an effective length of greater than or equal to 1.05 inches. The F* Distance is applicable only to tubes that are rerolled below the mid-plane of the tubesheet and have no degradation in the rerolled expansion joints. The F* Distance is not applicable to the original factory hard roll.

"F* tube" means a tube with indications below the F* distance greater than or equal to the Plugging or Repair Limit, and no indications within the F* distance.

"Imperfection" means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy current testing indications < 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

"Plugging or Repair Limit" means the imperfection depth at or beyond which the tube shall be repaired or removed from service by plugging because it may become unserviceable prior to the next inspection.

"Repaired Tube" means a tube that has undergone a process that re-establishes its serviceability. The sleeving process utilized shall be one of the following:

1. The Combustion Engineering welded sleeve process as described in Report CEN-331-P, Revision 1-P. The following sections of ABB/CE Report CEN-629-P, Revision 00, shall be implemented in conjunction with Report CEN-331-P, Revision 1-P:

(continued)

5.5 Programs and Manuals

5.5.7 Steam Generator (SG) Tube Surveillance Program (continued)

Section 4.5.3 - Tube Brushing-Cleaning Equipment
Section 4.5.7 - Nondestructive Examination
Section 5.0 - Sleeve Examination Program
Section 9.3.1 - Cleaning Qualification

2. The Westinghouse Electric Corporation mechanical sleeve process as described in Report WCAP-11669.

3. The Bechtel-KWU Alliance welded sleeve process as described in Report BKAT-01-P, Revision 1.

"Sleeve Inspection" means an inspection of the sleeved portion of the tube. For a Combustion Engineering welded sleeve, this inspection will include the upper weld which forms the new pressure boundary and the sleeve material below the upper weld. For a Westinghouse Electric Corp. mechanical sleeve, this inspection will include the entire length of sleeve material. For a Bechtel-KWU Alliance welded sleeve, this inspection will include the upper weld which forms the new pressure boundary and the sleeve material below the upper weld.

"Tube Inspection" means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. If a tube does not permit the passage of the eddy current inspection probe the entire length and through the U-bend; this shall be recorded and an adjacent tube shall be inspected.

"Unserviceable" describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified above.

5.5.8 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

(continued)

5.5 Programs and Manuals

5.5.8 Secondary Water Chemistry Program (continued)

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.9 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Positions C.5 and C.6 of Regulatory Guide 1.52, Revision 2, dated March 1978, and conducted in general conformance with ANSI N510-1975. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

a. Demonstrate for each of the ESF filter systems that an inplace test of the high efficiency particulate air (HEPA) filters shows removal capability of ≥ 99% of dioctyl phthalate (DOP) when tested in general conformance with ANSI N510-1975 at the flowrate specified below. Verification of the specified flowrates may be accomplished during the performance of SR 3.7.9.4, SR 3.7.12.3, and SR 3.7.13.5 as applicable. The PTEFS and FHBEFS are shared and as such, will be tested in the FHBEFS mode.

(continued)

5.5 Programs and Manuals

5.5.9 Ventilation Filter Testing Program (VFTP) (continued)

ESF Filter System	Flowrate
Control Room Emergency	≥ 1600, ≤ 2200 cfm
Fuel Handling Building Exhaust	≤ 24,000 cfm
Emergency Core Cooling System and Containment Spray Cubicle Exhaust	≤ 48,000 cfm

b. Demonstrate for each of the ESF filter systems that an inplace test of the charcoal adsorber shows removal capability of ≥ 99.95% of halogenated refrigerant test gas when tested in general conformance with ANSI N510-1975 at the flowrate specified below. Verification of the specified flowrates may be accomplished during the performance of SR 3.7.9.4, and SR 3.7.13.5 as applicable. The charcoal filters are shared by the FHBEFS, PTEFS, and ECCS and CS CEFS and as such, will be tested in the FHBEFS alignment.

ESF Filter System	Flowrate
Control Room Emergency	≥ 1600, ≤ 2200 cfm
Fuel Handling Building Exhaust	≤ 24,000 cfm

c. Demonstrate for each of the ESF filter systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows a removal efficiency of an activated methyl iodide greater than the value specified below when tested in general conformance with ANSI N510-1975 at the temperature and relative humidity greater than or equal to that specified below.

(continued)

5.5 Programs and Manuals

5.5.9 Ventilation Filter Testing Program (VFTP) (continued)

ESF Filter System	Efficiency	Temp.	RH
Control Room Emergency	≥ 99%	≥ 190°F	≥ 70%
Fuel Handling Building Exhaust and Emergency Core Cooling System and Containment Spray Cubicle Exhaust and Pipe Tunnel Exhaust	≥ 95%	≥ 125°F	≥ 95%

d. Demonstrate for each of the ESF filter systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is ≤ 6 inches of water adjusted to the maximum design flowrate when tested in general conformance with ANSI N510-1975 at the flowrate specified below. Verification of the specified flowrates may be accomplished during the performance of SR 3.7.9.4, SR 3.7.12.3, and SR 3.7.13.5 as applicable. The PTEFS and the FHBEFS filters are shared and as such, will be tested in the FHBEFS mode.

ESF Filter System	Flowrate
Control Room Emergency	≥ 1600, ≤ 2200 cfm
Fuel Handling Building Exhaust	≤ 24,000 cfm
Emergency Core Cooling System and Containment Spray Cubicle Exhaust	≤ 48,000 cfm

5.5.10 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the waste gas system, the quantity of radioactivity contained in gas decay tanks or fed into the offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be determined in accordance with the ODCM.

(continued)

5.5 Programs and Manuals

5.5.10 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the waste gas system and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., the system is not designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas decay tank and fed into the offgas treatment system is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.11 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

(continued)

5.5 Programs and Manuals

5.5.11 Diesel Fuel Oil Testing Program (continued)

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. an API gravity or an absolute specific gravity within limits,

2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

3. a clear and bright visual appearance with proper color;

b. Other properties of the new fuel oil are within limits for ASTM 2D fuel oil within 31 days of addition to the storage tanks; and

c. Total particulate concentration of the fuel oil in the storage tanks is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D2276, Method A-2 or A-3.

5.5.12 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

i. A change in the TS incorporated in the license; or

ii. A change to the UFSAR or Bases that involves an unreviewed safety question as defined in 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e) as modified by approved exemptions.

(continued)

5.5 Programs and Manuals

5.5.13 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding

5.5.13 Safety Function Determination Program (SFDP) (continued)

exception to entering supported system Conditions and Required Actions. This program implements the requirements of LCO 3.0.6.

The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists; and

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities;

and may contain additional appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

(continued)

5.5 Programs and Manuals

5.5.13 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.14 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995, with the following exception: Section 10.2.1 of NEI 94-01 Revision 0, is not applicable to Unit 1 penetration P-16. This exception for penetration P-16 shall expire at the completion of Unit 1 outage Z1R15.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 46.79 psig. The containment design pressure is 47 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be $\leq$ 0.1% of containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $< 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage is $< 0.6\ L_a$ when combined with all Type B and C test results at a test pressure of $\geq P_a$.

   2) For each door, leakage rate is $\leq$ 1.0 SCFH when tested at $\geq$ 2.5 psig and < 10 psig, or $\leq$ 4.75 SCFH when tested at $\geq$ 10 psig.

(continued)

5.5 Programs and Manuals

5.5.14 Containment Leakage Rate Testing Program (continued)

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.6 Reporting Requirements (continued)

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing, as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits";
LCO 3.4.10, "Pressurizer Safety Valves"; and
LCO 3.4.12, "Low Temperature Overpressure Protection."

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those identified in the following documents: License Amendment number 177 for Unit 1 and 164 for Unit 2.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluency period and for any revision or supplement thereto.

5.6.7 Post Accident Monitor Report

When a report is required by LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.8 Containment Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Containment Tendon Surveillance Program shall be reported to the NRC within 90 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.1 Containment

LCO 3.6.1 Containment shall be OPERABLE.

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	-------------------NOTE-------------------- SR 3.0.2 is not applicable. --- Perform required visual examinations and leakage rate testing, except for containment air lock testing, in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.1.2	Verify containment structural integrity in accordance with the Containment Tendon Surveillance Program.	In accordance with the Containment Tendon Surveillance Program

3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Air Locks

LCO 3.6.2 Two containment air locks shall be OPERABLE.

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

ACTIONS

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

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment air locks with one containment air lock door inoperable.	------------NOTES------------ 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls if both air locks are inoperable. ----------------------------	(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.1	Verify the OPERABLE door is closed in the affected air lock.	1 hour
	<u>AND</u>		
	A.2	Lock the OPERABLE door closed in the affected air lock.	24 hours
	<u>AND</u>		
	A.3	Verify the OPERABLE door is locked closed in the affected air lock.	Once per 31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more containment air locks with containment air lock interlock mechanism inoperable.	------------NOTES------------ 1. Required Actions B.1, B.2, and B.3 are not applicable if both doors in the same air lock are inoperable and Condition C is entered. 2. Entry and exit of containment is permissible under the control of a dedicated individual. -----------------------------	
	B.1 Verify an OPERABLE door is closed in the affected air lock.	1 hour
	<u>AND</u>	
	B.2 Lock an OPERABLE door closed in the affected air lock.	24 hours
	<u>AND</u>	
	B.3 Verify an OPERABLE door is locked closed in the affected air lock.	Once per 31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more containment air locks inoperable for reasons other than Condition A or B.	C.1	Initiate action to evaluate overall containment leakage rate per LCO 3.6.1.	Immediately
	AND		
	C.2	Verify a dcor is closed in the affected air lock.	1 hour
	AND		
	C.3	Restore air lock to OPERABLE status.	24 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	-------------------NOTES------------------ 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria applicable to SR 3.6.1.1. --- Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program
SR 3.6.2.2	-------------------NOTE------------------- Only required to be performed upon entry or exit through the containment air lock. --- Verify only one door in the air lock can be opened at a time.	184 days

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment consists of the concrete reactor building, its steel liner, and the penetrations through this structure which encloses the entire reactor and Reactor Coolant System. The structure is designed to contain radioactive material that may be released from the reactor core following a design basis Loss of Coolant Accident . In addition, this structure provides shielding from the fission products that may be present in the containment atmosphere from both normal and post accident conditions (Ref. 1).

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

The cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions, and the dome roof is prestressed utilizing a three way post tensioning system.

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

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

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

   1. capable of being closed by an OPERABLE automatic containment isolation signal in MODES 1, 2 and 3,

(continued)

BACKGROUND (continued)

2. capable of being closed by an OPERABLE manual containment isolation signal in MODE 4, or

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

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks."

c. The equipment hatch is installed.

APPLICABLE SAFETY ANALYSES

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

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a Loss of Coolant Accident (LOCA) and a main steam line break (Ref. 2). In addition, the release of significant fission product radioactivity within containment can occur from a LOCA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 1). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B, as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA . The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.1% per day in the safety analysis at P_a = 46.79 psig (Ref. 2).

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time the leakage limits specified in the Containment Leakage Rate Testing Program must be met, i.e., the combined Type B and C leakage must be $< 0.6\ L_a$, and the overall Type A leakage must be $< 0.75\ L_a$.

Compliance with this LCO will ensure a containment configuration, including both air locks and the equipment hatch, that is structurally sound and will limit leakage to those leakage rates assumed in the safety analysis.

Individual leakage rates for the containment air lock door seals are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J, Option B. Therefore, exceeding the individual air lock door seal leakage rate limits specified in the Containment Leakage Rate Testing Program only results in the containment being inoperable when the combined Type B and C leakage rate exceeds the overall acceptance criteria specified in that program.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

A.1

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

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program.

Failure to meet individual air lock door seal leakage limits specified in the Containment Leakage Rate Testing Program does not invalidate the acceptability of these overall leakage determinations unless their contribution to the overall Type A, B, and C leakage causes that limit to be exceeded.

As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be < 0.6 L_a for combined Type B and Type C leakage.

As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be < 0.75 L_a for overall Type A leakage .

At all other times between required leakage rate tests, the acceptance criteria for Type B and Type C leakage tests are 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1 (continued)

SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.2

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of Specification 5.5.5, "Containment Tendon Surveillance Program." Testing and Frequency are in general conformance with the recommendations of Regulatory Guide 1.35 (Ref. 3).

REFERENCES

1. UFSAR, Section 3.8.

2. UFSAR, Chapter 15.

3. Regulatory Guide 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures," Revision 2, January 1976.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.2 Containment Air Locks

BASES

BACKGROUND

Containment air locks form part of the containment pressure boundary and provide a means for personnel access during all MODES of operation.

Two air locks are provided. The personnel air lock, used for normal ingress and egress of containment, and the emergency air lock typically reserved for entering and exiting containment in emergency situations. The personnel air lock penetrates the equipment hatch. The emergency air lock is a right circular cylinder located on the opposite side of containment at ground level. The doors in each air lock are interlocked to prevent simultaneous opening. During periods when containment is not required to be OPERABLE, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. Each of the air lock doors contains double gasketed seals and local leakage rate testing capability to ensure seal integrity. To effect a leak tight seal, the air lock design uses pressure seated doors (i.e., an increase in containment internal pressure results in increased sealing force on each door). 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) in containment. As such, closure of a single air lock door supports containment OPERABILITY.

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

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident and a rod ejection accident (Ref. 1). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 1). **This leakage rate is defined in 10 CFR 50, Appendix J, Option B, as the maximum allowable containment leakage rate at the calculated peak containment internal pressure, P_a, 46.79 psig, following a design basis LOCA.** This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of the NRC Policy Statement.

LCO

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

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

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

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

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock.

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

<u>A.1, A.2 and A.3</u>

With one air lock door in one or more containment air locks inoperable, the OPERABLE door must be verified closed (Required Action A.1) in each affected containment air lock. This ensures that a leak tight containment barrier is maintained by the use of an OPERABLE air lock door. This action must be completed within 1 hour. This specified time period is consistent with the ACTIONS of LCO 3.6.1, which

(continued)

BASES

ACTIONS

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

requires 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 reasonable for locking the OPERABLE air lock door, considering the OPERABLE door of the affected air lock is being maintained closed.

Required Action A.3 verifies that an air lock with an inoperable door has been isolated by the use of a locked and closed OPERABLE air lock door. This ensures that an acceptable containment leakage boundary is maintained. The Completion Time of once per 31 days is based on engineering judgment and is considered adequate in view of the low likelihood of a locked door being mispositioned and other administrative controls.

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 same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1, C.2 and C.3 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. The 7 day restriction begins when the second air lock is discovered inoperable. Containment entry may be required on a periodic basis to perform Technical Specifications (TS) Surveillances and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment.

This Note is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment is 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

(continued)

BASES

ACTIONS

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

pressurize the containment during the short time that the OPERABLE door is expected to be open.

B.1, B.2 and B.3

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

The Required Actions have been modified by two Notes. Note 1 ensures that only the Required Actions and associated Completion Times of Condition C are required if both doors in the same air lock are inoperable. With both doors in the same air lock inoperable, an OPERABLE door is not available to be closed. Required Actions C.1, C.2 and C.3 are the appropriate remedial actions. Note 2 allows entry into and exit from containment under the control of a dedicated individual stationed at the air lock to ensure that only one door is opened at a time (i.e., the individual performs the function of the interlock).

C.1, C.2 and C.3

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

(continued)

ACTIONS

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

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

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

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 within 6 hours and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The periodic testing requirements verify that the air lock leakage, when combined with other Type B and C test results, does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by the Containment Leakage Rate Testing Program. The testing requirements for the individual doors provide early indication of potential degradation that may affect the ability to meet overall Type B leakage criteria.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1 (continued)

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. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate.

SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and given that the interlock mechanism is only challenged when the containment air lock door is opened, this test is only required to be performed upon entering or exiting a containment air lock but is not required more frequently than every 184 days. Therefore, performance of the SR may be extended beyond 184 days (plus the allowance specified in SR 3.0.2) without violating the requirements of SR 3.0.1. The 184 day Frequency is based on engineering judgment and is considered adequate in view of other indications of door and interlock mechanism status available to operations personnel.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Chapter 15.

CTS MARKUPS

3.1.8/3.4.1 LIMITING CONDITION FOR OPERATION	3.1.8/3.4.1 SURVEILLANCE REQUIREMENT
3.1.8 3.2.3.D.1 Applicability: Modes 1 and 2	3.1.8 4.2.3.D.1.a shall be checked indirectly by excore detectors and/or thermocouples and/or moveable incore detectors every shift; or after any rod motion of the non-indicating rod, exceeding 12 steps, whichever occurs first. 3.1-48 3.1-49 RA B.1 3.1-50
R.A C.1 Required Action: If the conditions of Section 3.2.3.D.1 cannot be met the reactor shall be brought to at least the Hot Shutdown condition within four hours. 3.1-33	R.A. C.1 b. During operation below 50% of rated power, no special monitoring is required.
4. DNB Parameters	
LCO 3.4.1 A. The following DNB related parameters shall be maintained within the limits shown during operation.	3.4.1 4.A.1. Each of the parameters listed in Specification 3.2.4.A shall be verified to be within its limit at least once per 12 hours. SR 3.4.1.1 SR 3.4.1.2 SR 3.4.1.3
LCO 3.4.1 1. Reactor Coolant System Tavg FOUR LOOP: ≤ 567.7°F 2. Pressurizer Pressure FOUR LOOP: ≥ 2210 psia (2195 psig)* 3. Reactor Coolant System Total Flow Rate FOUR LOOP: ≥ 362,300 GPM# 3.4-4	SR 3.4.1.4 2. The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.
LCO 3.4.1 RA A.1 RA B.1 B. With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce thermal power to less than 5% of rated thermal power within the next 4 hours. 3.4-5	3.4-6 Add SR 3.4.1.4 NOTE

LCO 3.4.1 NOTE + SR 3.4.1.1 NOTE

* Limit not applicable during either a thermal power ramp increase in excess of 5% rated thermal power per minute or a thermal power step increase in excess of 10% rated thermal power.

\# Includes a 3.5% flow measurement uncertainty. 3.4-4

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

10. Repaired Tube means a tube that has undergone a process that re-establishes its serviceability. The sleeving process utilized shall be one of the following:

    a. The Combustion Engineering welded sleeve process as described in Report CEN-331-P, Revision 1-P.**

    b. The Westinghouse Electric Corporation mechanical sleeve process as described in Report WCAP-11669.

    c. The Bechtel-KWU Alliance welded sleeve process as described in Report BKAT-01-P, Revision 1.

5.5.7

11. F^* Distance is the length of undegraded tube required to resist pullout. This distance is measured from the bottom of the upper hard roll transition toward the bottom of the tubesheet and has been conservatively determined to be 1.05 inches. The determination of the F* Distance is satisfied by the mechanical tool (rolling pin) used to install the joint having an effective length of greater than or equal to 1.05 inches. The F* Distance is applicable only to tubes that are rerolled below the mid-plane of the tubesheet and have no degradation in the rerolled expansion joints.

12. F^* Tube is a tube with indications below the F^* Distance, equal to or greater than 40%, and no indications within the F^* Distance.

5.5.7

**The following sections of ABB/CE Report CEN-629-P Revision 00 shall be implemented in conjunction with Report CEN-331-P, Revision 1-P:
  Section 4.5.3 - Tube Brushing-Cleaning Equipment
  Section 4.5.7 - Nondestructive Examination
  Section 5.0 - Sleeve Examination Program
  Section 9.3.1 - Cleaning Qualification

Bases:

3.3.1 & 4.3.1 **Steam Generators**

5.5.7

The F* Distance will be controlled by process control. For a new additional roll expansion, the requirement will be at least 1.05 inches of new hard roll. This is controlled by the length of the rolling pin used during the mechanical installation of the new hard roll joint. The distance from the original roll transition zone is also controlled by the process in that the lower end of the new roll expansion is located approximately one inch above the original factory hard roll expansion. In the case of the new roll, eddy current examination will confirm that there are no indications in the new roll region and that there is a new roll region with well defined upper and lower expansion transitions. The F* Distance is not applicable to the original factory hard roll.

3.6.1 LIMITING CONDITION FOR OPERATION

3.10 CONTAINMENT STRUCTURAL INTEGRITY (per unit)

OBJECTIVE:

To insure that the containment structure meets its design requirements throughout plant life.

SPECIFICATION:

1. Containment Leakage Rate Testing

A. Containment Leakage Rate shall be limited to:

3.6-63

3.6-92

1. An overall integrated leakage rate less than or equal to that specified by the Containment Leakage Rate Testing Program.

3.6-92

2. A combined leakage rate less than or equal to that specified by the Containment Leakage Rate Testing Program, for all penetrations and valves subject to Type B and C tests, when pressurized to P_a.

APPLICABILITY: Modes 1, 2, 3, 4 and 7 3.6-2

SURVEILLANCE REQUIREMENT

4.10 CONTAINMENT STRUCTURAL INTEGRITY (per unit)

OBJECTIVE:

To establish the testing requirements to assure containment structural integrity.

1. Containment Leakage Rate Testing

A. Surveillance and testing of the containment shall be performed as follows: 3.6-92

3.6-92

SR 3.6.1.1 w/note + 5.5.14

1. The containment Type A leakage rate shall be determined in accordance with the Containment Leakage Rate Testing Program.

3.6.1 LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.10.1.A (Continued)

ACTION: 3.6-92

With either (a) the measured overall integrated containment leakage rate, or (b) the measured combined leakage rate for all penetrations and valves subject to Type B and C tests exceeding that specified by the Containment Leakage Rate Testing Program, restore the applicable leakage rate to less than or equal to that specified by the Containment Leakage Rate Testing Program prior to entering MODE 4.

4.10.1.A.2. Type B and C tests (except air lock tests) shall be performed in accordance with that specified by the Containment Leakage Rate Testing Program. 3.6-92

3. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.10.2. 3.6-66

4. Type A, B, and C leakage rate tests shall be considered to be satisfactory if the acceptance criteria delineated by the Containment Leakage Rate Testing Program are met. 3.6-63

5. Leakage from containment isolation valves sealed by the Isolation Valve Seal Water system may be excluded from the combined Type B and C leakage rate. 3.6-

3.6.1 LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.10.1.A (Continued)	4.10.1.A.6. The retest schedules for Type A, B, and C tests shall be in accordance with the Containment Leakage Rate Testing Program. 36-63 7. Inspection and reporting of tests shall be in accordance with the Containment Leakage Rate Testing Program. 3.6

3.6.2 LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.10.2. Containment Air Locks

A. Each containment air lock shall be OPERABLE with:

3.6-92

3.6-68 1. Both doors closed except when the air lock is being used for normal entry and exit through the containment, then at least one air lock door shall be closed, and

3.6-69 2. The air lock leakage rate shall be less than or equal to that specified by the Containment Leakage Rate Testing Program.

APPLICABILITY: Modes 1, 2, 3, 4 and 7 3.6-2

3.6-57 3.6-70

ACTION:

CONDA a.) With one containment air lock door inoperable: 3.6-62

R.A. A.1 A.2 1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

3.6-72

A.3 2. Operation may continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified locked closed at least once per 31 days,

COND D R.A. D.1 D.2 3. Otherwise be in at least MODE 3 within the next 6 hours and MODE 5 within the following 30 hours.

4.10.2. Containment Air Locks

A. Each air lock shall be demonstrated OPERABLE:

3.6-92 ~~3.6-93~~ Verify that the overall air lock leakage rate, and the leakage rate for each door, are less than or equal to that specified by the Containment Leakage Rate Testing Program. Verifications shall be conducted at the frequency specified by the Containment Leakage Rate Testing Program.

3.6.2 LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.10.2 (Continued) 3.6-75 COND B 3.6-76 4. The provisions of Technical Specification 3.0.4 are not applicable. 3.6-77 COND C b.) With the containment air lock inoperable, except as a result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least MODE 3 within the next 6 hours and in MODE 5 within the following 30 hours. COND D R.A. D.1 D.2	3.6-81 SR 3.6.2.2 At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

5.5.14

6.10 Containment Leakage Rate Testing Program

5.0.45

1. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leakage Test Program" dated September 1995, with the following exception:

   Section 10.2.1 of NEI 94-01, Revision 0, is not applicable to Unit 1 penetration P-16. This exception for penetration P-16 shall expire at the completion of Unit 1 outage Z1R15.

2. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 46.79 psig. The containment design pressure is 47 psig.

3. The maximum allowable containment leakage rate, L_a, at P_a, shall be $\leq$0.1% of containment air weight per day.

4. Leakage rate acceptance criteria are:

   a. Containment leakage rate acceptance criterion is $\leq$1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are <0.60 L_a for the Type B and C tests and <0.75 L_a for the Type A tests;

   b. Air lock testing acceptance criteria are:

      1) Overall air lock leakage is $\leq$0.6 L_a when combined with all Type B and C test results at a test pressure of $\geq P_a$.

      2) For each door, leakage rate is $\leq$1.0 SCFH when tested at $\geq$2.5 psig and <10 psig, or $\leq$4.75 SCFH when tested at $\geq$10 psig.

5. The provisions of Surveillance Requirement 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

6. The provisions of Surveillance Requirement 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

MARK UP OF DOC CHANGE

NSHC NO. DISCUSSION

L-7. 32. Appropriate Required Actions and Completion Times are provided for the Condition of more than one rod position indicator channel per control rod group inoperable, and for "More than one RPI per group inoperable for one or more groups." The latter condition addresses the unlikely situation in which control rod position indication is lost in its entirety. This change is consistent with Amendment 61 for Callaway Plant (requested 9-7-90, approved 2-1-91). The NRC Staff's SER for this amendment indicates that "the requested amendments correct an oversight in the licensee's existing TS that would require operation of a system (control rods) important to safety without availability of the preferred monitoring capability" (rod position indication). This same oversight exists in the old STS and in the new ITS. Both currently require a shutdown (control rod manipulation) without complete control rod position indication. This new Condition will provide an increased time to attempt repair while limiting the likelihood of control rod misalignment during this period. The safety function of the control rods (tripability) is unaffected by this condition, and as such, control rod misalignment is the primary concern. The proposed 30 hours will significantly increase the probability of identifying and correcting the cause of the inoperability, while allowing time to plan alternative monitoring of control rod position should a plant shutdown be required. Without the change, the plant is required to initiate a shutdown in 1 hour and be in MODE 3 within 7 hours. This is not sufficient time to notify the appropriate individuals, brief them in the situation and alternatives for control rod monitoring, or to allow them to identify and correct the cause of the inoperability. Therefore, the current requirements as written will result in a plant shutdown without adequate time to address corrective actions or monitoring capabilities. The proposed Condition provides appropriate compensatory measures to continue to assure compliance with the assumptions of the safety analysis while providing sufficient time to identify and correct minor causes of inoperability, or to preplan alternative measures relative to monitoring control rod position during a plant shutdown.

L-5. 33. The requirement to place the reactor in Hot Shutdown ~~and maintain the reactor trip breakers open~~ when the required actions associated with the Rod Position Indicator Specification cannot be met, has been modified to only require the plant be brought to a power level of less than 50% RTP. A shutdown to MODE 3 places the unit outside the MODE in which the rod position indicators are required. However, reducing power to less than 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors. Therefore, this is an appropriate action if the initial RA is not met and is consistent with the described Bases of NUREG-1431.

NSHC	NO.	DISCUSSION
A.	90.	In CTS 4.10.5, the Surveillance Frequency for the verification of containment pressure has been specified as once per "12 hours" instead of the "once per shift." At Zion Station, the normal shift is 12 hours. As a result, this change is editorial in nature.
L-26.	91.	In CTS 4.10.6, the Surveillance Frequency for the verification of containment temperature has been revised to once per "24 hours" instead of "once per shift." The 24 hour Frequency is considered acceptable based on the observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). In addition, other indications are available in the control room to alert the operator to an abnormal containment temperature condition.
A.	92.	~~Deleted~~ References to Appendix J have been changed to reference the Containment Leakage Rate Testing Program following implementation of 10 CFR 50, Appendix J, Option B.
L-28	93.	This change to the requirements of the CTS 4.9.3.A.2 exempts certain automatic containment isolation valves from the 18 month surveillance testing that would demonstrate satisfactory operation. The valves are exempted because they are locked, sealed, or otherwise secured in the required position under administrative controls. These valves do not reposition in order to fulfill their safety function, and are secured in their required position to fulfill their accident function. Therefore no automatic isolation is required. This exemption is in accordance with NUREG-1431, Rev 1.
L-29	94.	This change to the requirements of CTS 4.5.1.b.2 eliminates the 18 month surveillance for those required (Accident Inlet, Accident Outlet, and Normal Inlet) dampers that have been secured in the accident position. It would be superfluous to verify the position of such dampers, and any alteration which would allow the dampers to be repositioned would constitute a change to the facility design.
L-A	95.	~~SR 4.5.1.a.2, verification of SW flow to the RCFC Coolers, has been relocated under the Zion ITS. It has been replaced by performance of ITS SR 3.7.8.1.~~ SR 4.5.1.a.2, verification of SW flow to the RCFC coolers, has been revised under the Zion ITS. The flow verification (SR 3.6.6.2) now verifies that the SW System configuration is within the assumptions of the SW System hydraulic flow analysis that was performed to verify SW System design basis.

NSHC	NO.	DISCUSSION
L-A	43.	CTS requirement 4.22.1.A.4.a has been relocated to licensee control. This requirement specifies the performance of mechanical snubber drag testing during functional tests. In light of other required testing (e.g.; activation, release rate), this testing does not provide any added data relevant to the determination of snubber functionality. This testing will continued to be performed as a "good practice" under licensee control in plant procedures. Based on the above information, it has been concluded that moving this test to plant procedures can be done without an impact on safety. Changes to these documents will be controlled in accordance with plant processes and practices. This change is consistent with NUREG-1431, in that all snubber testing is relocated to licensee control.
L-A	44.	CTS requirement 4.22.3 has been relocated to plant procedures. This requirement specifies the performance of service life monitoring for snubber components. This requirement was placed into the Technical Specifications to address snubber components with limited life (elastomers), which have since been improved and replaced. Monitoring will continue to be performed under licensee control. Changes to these documents will be controlled in accordance with plant processes and practices. This change is consistent with NUREG-1431, in that all snubber testing is relocated to licensee control.
A.	45.	Information pertaining to the Containment Leakage Rate Testing Program implemented under 10 CFR 50, Appendix J, Option B, has been relocated to Specification 5.5.14, Containment Leakage Rate Testing Program.

CLEAN DOC

NSHC NO. DISCUSSION

L-7. 32. Appropriate Required Actions and Completion Times are provided for the Condition of more than one rod position indicator channel per control rod group inoperable, and for "More than one RPI per group inoperable for one or more groups." The latter condition addresses the unlikely situation in which control rod position indication is lost in its entirety. This change is consistent with Amendment 61 for Callaway Plant (requested 9-7-90, approved 2-1-91). The NRC Staff's SER for this amendment indicates that "the requested amendments correct an oversight in the licensee's existing TS that would require operation of a system (control rods) important to safety without availability of the preferred monitoring capability" (rod position indication). This same oversight exists in the old STS and in the new ITS. Both currently require a shutdown (control rod manipulation) without complete control rod position indication. This new Condition will provide an increased time to attempt repair while limiting the likelihood of control rod misalignment during this period. The safety function of the control rods (tripability) is unaffected by this condition, and as such, control rod misalignment is the primary concern. The proposed 30 hours will significantly increase the probability of identifying and correcting the cause of the inoperability, while allowing time to plan alternative monitoring of control rod position should a plant shutdown be required. Without the change, the plant is required to initiate a shutdown in 1 hour and be in MODE 3 within 7 hours. This is not sufficient time to notify the appropriate individuals, brief them in the situation and alternatives for control rod monitoring, or to allow them to identify and correct the cause of the inoperability. Therefore, the current requirements as written will result in a plant shutdown without adequate time to address corrective actions or monitoring capabilities. The proposed Condition provides appropriate compensatory measures to continue to assure compliance with the assumptions of the safety analysis while providing sufficient time to identify and correct minor causes of inoperability, or to preplan alternative measures relative to monitoring control rod position during a plant shutdown.

L-5. 33. The requirement to place the reactor in Hot Shutdown when the required actions associated with the Rod Position Indicator Specification cannot be met, has been modified to only require the plant be brought to a power level of less than 50% RTP. A shutdown to MODE 3 places the unit outside the MODE in which the rod position indicators are required. However, reducing power to less than 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors. Therefore, this is an appropriate action if the initial RA is not met and is consistent with the described Bases of NUREG-1431.

M. 34. An Applicability for Specification 3.2.1.H has been added. This Applicability is for MODES 1 and 2 and ensures that the limits on

NSHC	NO.	DISCUSSION
A.	90.	In CTS 4.10.5, the Surveillance Frequency for the verification of containment pressure has been specified as once per "12 hours" instead of the "once per shift." At Zion Station, the normal shift is 12 hours. As a result, this change is editorial in nature.
L-26.	91.	In CTS 4.10.6, the Surveillance Frequency for the verification of containment temperature has been revised to once per "24 hours" instead of "once per shift." The 24 hour Frequency is considered acceptable based on the observed slow rates of temperature increase within containment as a result of environmental heat sources (due to the large volume of containment). In addition, other indications are available in the control room to alert the operator to an abnormal containment temperature condition.
A.	92.	References to Appendix J have been changed to reference the Containment Leakage Rate Testing Program following implementation of 10 CFR 50, Appendix J, Option B.
L-28	93.	This change to the requirements of the CTS 4.9.3.A.2 exempts certain automatic containment isolation valves from the 18 month surveillance testing that would demonstrate satisfactory operation. The valves are exempted because they are locked, sealed, or otherwise secured in the required position under administrative controls. These valves do not reposition in order to fulfill their safety function, and are secured in their required position to fulfill their accident function. Therefore no automatic isolation is required. This exemption is in accordance with NUREG-1431, Rev 1.
L-29	94.	This change to the requirements of CTS 4.5.1.b.2 eliminates the 18 month surveillance for those required (Accident Inlet, Accident Outlet, and Normal Inlet) dampers that have been secured in the accident position. It would be superfluous to verify the position of such dampers, and any alteration which would allow the dampers to be repositioned would constitute a change to the facility design.
L-A	95.	SR 4.5.1.a.2, verification of SW flow to the RCFC coolers, has been revised under the Zion ITS. The flow verification (SR 3.6.6.2) now verifies that the SW System configuration is within the assumptions of the SW System hydraulic flow analysis that was performed to verify SW System design basis. Specifically, SW flow of 1500 gpm was verified to the RCFC coolers by performance of a flow analysis, given a minimum SW component alignment. The SW System alignment assumptions of the flow

NSHC	NO.	DISCUSSION
L-A	43.	CTS requirement 4.22.1.A.4.a has been relocated to licensee control. This requirement specifies the performance of mechanical snubber drag testing during functional tests. In light of other required testing (e.g.; activation, release rate), this testing does not provide any added data relevant to the determination of snubber functionality. This testing will continued to be performed as a "good practice" under licensee control in plant procedures. Based on the above information, it has been concluded that moving this test to plant procedures can be done without an impact on safety. Changes to these documents will be controlled in accordance with plant processes and practices. This change is consistent with NUREG-1431, in that all snubber testing is relocated to licensee control.
L-A	44.	CTS requirement 4.22.3 has been relocated to plant procedures. This requirement specifies the performance of service life monitoring for snubber components. This requirement was placed into the Technical Specifications to address snubber components with limited life (elastomers), which have since been improved and replaced. Monitoring will continue to be performed under licensee control. Changes to these documents will be controlled in accordance with plant processes and practices. This change is consistent with NUREG-1431, in that all snubber testing is relocated to licensee control.
A.	45.	Information pertaining to the Containment Leakage Rate Testing Program implemented under 10 CFR 50, Appendix J, Option B, has been relocated to Specification 5.5.14, Containment Leakage Rate Testing Program.

NUREG MARKUPS

5.5 Programs and Manuals (continued)

CTS 4.0.5

(12)

5.5.~~8~~6 Inservice Testing Program

This program provides controls, pumps and valves for inservice testing of ASME Code Class 1, 2, and 3 ~~components including applicable supports~~. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

edit

ASME ~~Boiler and Pressure Vessel Code and applicable Addenda~~ terminology for inservice testing activities	Required frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

CTS 4.3.1.B.1-4 ACTS Bases

5.5.~~9~~7 Steam Generator (SG) Tube Surveillance Program

~~Reviewer's Note: The Licensee's current licensing basis steam generator tube surveillance requirements shall be relocated from the LCO and included here. An appropriate administrative controls program format should be used.~~

INSERT 11A

(continued)

"Defect" means an imperfection of such severity that it exceeds the Plugging or Repair Limit, where applicable. A tube containing a defect is defective.

"F* distance" is the length of undegraded tube required to resist pullout. This distance is measured from the bottom of the upper hard roll transition toward the bottom of the tube sheet and has been conservatively determined to be 1.05 inches. The determination of F* distance is satisfied by the mechanical tool (rolling pin) used to install the joint having an effective length of greater than or equal to 1.05 inches. The F* distance is applicable only to tubes that are rerolled below the mid-plane of the tubesheet and have no degradation in the rerolled expansion joints. The F* Distance is not applicable to the original factory hard roll.

"F* tube" means a tube with indications below the F* distance greater than or equal to the Plugging or Repair Limit, and no indications within the F* distance.

"Imperfection" means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy current testing indications < 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

"Plugging or Repair Limit" means the imperfection depth at or beyond which the tube shall be repaired or removed from service by plugging because it may become unserviceable prior to the next inspection.

"Repaired Tube" means a tube that has undergone a process that re-establishes its serviceability. The sleeving process utilized shall be one of the following:

1. The Combustion Engineering welded sleeve process as described in Report CEN-331-P, Revision 1-P. The following section of AB/CE Report CEN-629-P, Revision 00, shall be implemented in conjunction with Report CEN-331-P, Revision 1-P:

   Section 4.5.3 - Tube Brushing-Cleaning Equipment
   Section 4.5.7 - Nondestructive Examination
   Section 5.0 - Sleeve Examination Program
   Section 9.3.1 - Cleaning Qualification

2. The Westinghouse Electric Corporation mechanical sleeve process as described in Report WCAP-11669.

3. The Bechtel-KWU Alliance welded sleeve process as described in Report BKAT-01-P, Revision 1.

"Sleeve Inspection" means an inspection of the sleeved portion of the tube. For a Combustion Engineering welded sleeve, this inspection will include the upper weld which forms the new pressure boundary and the sleeve material below the upper weld. For a Westinghouse Electric Corp. mechanical sleeve, this inspection will include the entire length of sleeve material. For a Bechtel-KWU Alliance welded sleeve, this inspection will include the upper weld which forms the new pressure boundary and the sleeve material below the upper weld.

"Tube Inspection" means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. If a tube does not permit the passage of the eddy current inspection probe the entire length and through the U-bend; this shall be recorded and an adjacent tube shall be inspected.

"Unserviceable" describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified above.

5.5 Programs and Manuals (continued)

5.5.~~15~~ 13 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

edit

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists; and

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; ~~and~~

edit

and may contain additional ~~d. Other~~ appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.0-24

INSERT 17A Table 5.5.7-1

Insert 17B

New Specification 5.5.14

INSERT 17B for new Specification 5.5.14

5.5.14 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995, with the following exception: Section 10.2.1 of NEI 94-01 Revision 0, is not applicable to Unit 1 penetration P-16. This exception for penetration P-16 shall expire at the completion of Unit 1 outage Z1R15.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 46.79 psig. The containment design pressure is 47 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be $\leq$ 0.1% of containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $< 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage is $< 0.6\ L_a$ when combined with all Type B and C test results at a test pressure of $\geq P_a$.

   2) For each door, leakage rate is $\leq$ 1.0 SCFH when tested at $\geq$ 2.5 psig and < 10 psig, or $\leq$ 4.75 SCFH when tested at $\geq$ 10 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following: [The individual specifications that address RCS pressure and temperature limits must be referenced here.]

INSERT 21A

22.

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents: [Identify the NRC staff approval document by date.]

INSERT 21B

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

Insert 21C

WCAP-14040

[SER pending]

Reviewers' Notes: The methodology for the calculation of the P-T limits for NRC approval should include the following provisions:

1. The methodology shall describe how the neutron fluence is calculated (reference new Regulatory Guide when issued).

2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.

3. Low Temperature Overpressure Protection (LTOP) System lift setting limits for the Power Operated Relief Valves (PORVs), developed using NRC-approved methodologies may be included in the PTLR.

4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for radiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.

5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits.

(continued)

INSERT 21A for Spec. 5.6.6.a:

3.4.3 RCS Pressure and Temperature (P/T) Limits
3.4.10 Pressurizer Safety Valves
3.4.12 Low Temperature Overpressure Protection (LTOP)

INSERT 21B for Spec. 5.6.6.b:

The RCS pressure and temperature limits, the Cold Overpressure Protection setpoint and arming temperature, and the RCS Safety Valve applicability for Units 1 and 2 will be those previously reviewed and approved in Amendments 177 and 164, respectively.

INSERT 21C for Spec. 5.6.6.b:

The first revision to the limits after initial implementation of the PTLR will be submitted to the NRC for review prior to approval. Subsequent revisions will be made without prior approval.

BASES

SR 3.0.1 (continued)

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current MODE or other specified conditions in the Applicability due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed.

Additional sample provided

INSERT B10A →

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per . . ." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers plant operating conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

4

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. An example of where SR 3.0.2 does not apply is a Surveillance with a Frequency of "in accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions.~~" The requirements of regulations take precedence over the TS. The TS cannot in and of themselves extend a test interval specified in the regulations.

the Containment Leakage Rate Testing Program

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	Perform required visual examinations and leakage rate testing except for containment air lock testing, in accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions.~~ the Containment Leakage Rate Testing Program (19) ~~The maximum allowable leakage rate, L_a, is [ ]% of containment air weight per day at the calculated peak containment pressure, P_a.~~ BWR 14 C1	-----NOTE------ ~~SR 3.0.2 is not applicable~~ (19) --------------- In accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~ the Containment Leakage Rate Testing Program (19)
[SR 3.6.1.2	Verify containment structural integrity in accordance with the Containment Tendon Surveillance Program.	In accordance with the Containment Tendon Surveillance Program]

BWR 14 C1

NOT USED

~~The leakage rate acceptance criterion is ≤ 1.0 La. 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 La for Type B and Type C tests, and ≤ 0.75 La for the Type A tests.~~ (19)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.1	-------------------NOTES------------------- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria ~~of SR 3.6.1.1 in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions~~ applicable to SR 3.6.1.1 (19). --- Perform required air lock leakage rate testing in accordance ~~with 10 CFR 50, Appendix J, as modified by approved exemptions~~ the Containment Leakage Rate Testing Program (19). ~~The acceptance criteria for air lock testing are:~~ ~~a. Overall air lock leakage rate is ≤ [0.05 L_a] when tested at ≥ P_a.~~ 6 when combined with other Type B and Type C test results ~~b. For each door, leakage rate is ≤ [.01 L_a] when tested at ≥ [P_a psig].~~ 1 SCFH, with test at 2.5 ... and ≤ 10 psig, or ≤ 4/75 SCFH when tested at ≥ 10 psig (19)	-----NOTE----- ~~SR 3.0.2 is not applicable~~ -------------- In accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~ the Containment Leakage Rate Testing Program (19)
SR 3.6.2.2	-------------------NOTE------------------- Only required to be performed upon entry or exit through the ~~into~~ containment air lock. --- Verify only one door in the air lock can be opened at a time.	184 days

BLLR 13 C2

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment ~~(Atmospheric)~~

BASES

BACKGROUND

which completely encloses the entire Reactor and Reactor Coolant System.

EDITORIAL

The containment consists of the concrete reactor building, its steel liner, and the penetrations through this structure. The structure is designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident ~~(DBA)~~. In addition ~~Additionally~~, this structure provides shielding from the fission products that may be present in the containment atmosphere from both normal and post ~~following~~ accident conditions. (Ref. 1)

Loss of Coolant (19)

The containment is a reinforced concrete structure with a cylindrical wall, a flat foundation mat, and a shallow dome roof. The inside surface of the containment is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions.

Plant Specific

~~For containments with ungrouted tendons,~~ The cylinder wall is prestressed with a post tensioning system in the vertical and horizontal directions, and the dome roof is prestressed utilizing a three way post tensioning system.

The concrete reactor building is required for structural integrity of the containment under Design Basis Accident ~~(DBA)~~ conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the environment. SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J ~~(Ref. 1)~~, as modified by approved exemptions. (19)

(19) Option 3

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

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

10

1. capable of being closed by an OPERABLE automatic containment isolation system in MODES 1, 2 and 3, ~~or~~

(continued)

10

BASES

2. Capable of being closed by an OPERABLE manual Containment Isolation Signal in MODE 4, or

BACKGROUND (continued)

3 ~~2~~. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves."

Bur-15 C3

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks."

c. The ~~All~~ equipment hatches are ~~closed~~ installed.

[c. ~~The pressurized sealing mechanism associated with a penetration is OPERABLE, except as provided in LCO 3.6.[ ].~~]

APPLICABLE SAFETY ANALYSES

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

Editorial

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA) and a main steam line break ~~, and a rod ejection accident (REA)~~ (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA ~~or REA~~. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of ~~[0.1]~~ 0.1% of containment air weight per day (Ref. 1 ~~3~~). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J ~~(Ref. 1)~~ option B, as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting ~~DBA~~ design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be ~~[0.1]~~ 0.1% per day in the safety analysis at P_a = ~~[44.1]~~ 46.79 psig (Ref. 1 ~~3~~).

19

19

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

BWR-14 C1 → INSERT "4"

Editorial.

Containment OPERABILITY is maintained by limiting leakage to within the acceptance criteria of 10 CFR 50, Appendix J (Ref. 1).

Compliance with this LCO both air locks and the will ensure a containment configuration, including equipment hatches, that is structurally sound and ~~that~~ will limit leakage to those leakage rates assumed in the safety analysis.

the individual Individual leakage rates ~~specified~~ for the containment air lock (19) ~~(LCO 3.6.2)~~ ~~[and purge valves with resilient seals (LCO 3.6.3)]~~ are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates (19) exceeding air lock ~~exceeding these individual~~ limits door seal (19) only result in the containment being inoperable when the leakage ~~results in~~ exceeds ~~exceeding~~ the overall (19) acceptance criteria ~~of Appendix J~~ (19) specified in the Containment Leakage Rate Testing Program (19) specified in that program.

(19) Option B

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.~~4~~3, "Containment Penetrations."

ACTIONS

A.1

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

(continued)

BASES

ACTIONS (continued)

EDITORIAL

B.1 and B.2

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

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

(19) Replace w/ Insert 3

BWR-14 C1 → INSERT "2"

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of 10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions. Failure to meet air lock ~~[and purge valve with resilient seal]~~ leakage limits specified in LCO 3.6.2 ~~[and LCO 3.6.3]~~ does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. SR Frequencies are as required by Appendix J, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

the Containment Leakage Rate Testing Program

CLARIFICATION

SR 3.6.1.2

~~For ungrouted, post tensioned tendons,~~ this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of ~~the~~ Specification 5.5.5, "Containment Tendon Surveillance Program." Testing and Frequency are consistent with the recommendations of Regulatory Guide 1.35 (Ref. ~~4~~ 3).

REFERENCES

1. UFSAR, Section 3.8.
~~1. 10 CFR 50, Appendix J.~~

~~2.~~ 2. UFSAR, Chapter ~~[15]~~.

(continued)

3.6.1 BASES

Please Note; Inserts 1 & 2 are the result of approved generic changes to NUREG-1431.

INSERT "1"

BWR-14, C1 (19) Containment Leakage Rate Testing Program

...≤ 1.0 La, 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 La, and the overall Type A leakage must be < 0.75 La.

(19) the leakage limits specified in the Containment Leakage Rate Testing Program must be met, i.e.,

INSERT "2"

BWR-14, C1

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 La for combined Type B and Type C leakage, and < 0.75 La for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 La. At ≤ 1.0 La the offsite dose consequences are bounded by the assumptions of the safety analysis.

INSERT 3

FOR BASES 3.6.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program.

Failure to meet individual air lock door seal leakage limits specified in the Containment Leakage Rate Testing Program does not invalidate the acceptability of these overall leakage determinations unless their contribution to the overall Type A, B, and C leakage causes that limit to be exceeded.

As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be $< 0.6\ L_a$ for combined Type B and Type C leakage.

As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be $< 0.75\ L_a$ for overall Type A leakage.

At all other times between required leakage rate tests, the acceptance criteria for Type B and Type C leakage tests are based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident and a rod ejection accident (Ref. 2). In the analysis of each of these accidents, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of [0.1]% of containment air weight per day (Ref. 2). ~~This leakage rate is defined in 10 CFR 50, Appendix J (Ref. 1), as L_a = [0.1]% per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a = [14.4] psig following a DBA.~~ This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

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This leakage rate is defined in 10CFR50, Option B, as the maximum allowable containment leakage rate at the calculated peak containment internal pressure, Pa, 46.7? psig, following a design basis LOCA.

The containment air locks satisfy Criterion 3 of the NRC Policy Statement.

LCO

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

pressure boundary

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

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due

(continued)

BASES

ACTIONS

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

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

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

EDITORIAL

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in ~~plant must be brought to~~ a MODE in which the LCO does not apply. This is done by placing the Unit in ~~To achieve this status, the plant must be brought to at least~~ MODE 3 within 6 hours and ~~to~~ MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of (19) the Containment Leakage rate Testing program ~~10 CFR 50, Appendix J (Ref. 1), 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 containment OPERABILITY testing.~~ The periodic testing requirements verify that the air lock leakage when combined with other Type B and C test is does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by ~~Appendix J (Ref. 1), as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply.~~ (19) the containment Leakage rate Testing Program. The testing requirements for the individual doors provide early indication of potential degradation that may affect the ability to meet overall Type B leakage criteria.

CLARIFICATION

(continued)

BWR 15

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.2.1 (continued)

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. This ensures that air lock leakage is properly accounted for in determining the ~~overall~~ containment leakage rate. (Combined Type B and C)

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SR 3.6.2.2

The air lock interlock is designed to prevent simultaneous opening of both doors in a single air lock. Since both the inner and outer doors of an air lock are designed to withstand the maximum expected post accident containment pressure, closure of either door will support containment OPERABILITY. Thus, the door interlock feature supports containment OPERABILITY while the air lock is being used for personnel transit in and out of the containment. Periodic testing of this interlock demonstrates that the interlock will function as designed and that simultaneous opening of the inner and outer doors will not inadvertently occur. Due to the purely mechanical nature of this interlock, and ~~given~~ that the interlock mechanism is only challenged when the containment air lock door is opened, ~~entered~~, this test is only required to be performed upon entering or exiting a containment air lock but is not required more frequently than every 184 days. The 184 day Frequency is based on engineering judgment and is considered adequate in view of other indications of door and interlock mechanism status available to operations personnel.

BWR 15

C2

REFERENCES

1. 10 CFR 50, Appendix J.
2. FSAR, Section [6.2].

Usage Rule Clarification

Therefore, performance of the SR may be extended beyond 184 days (plus the allowance specified in SR 3.0.2) without violating the requirement of SR 3.0.1

DOD CHANGES

3. This change eliminates the potential confusion that may arise with respect to the application of an unplanned event which satisfies the requirement of a given SR by including a discussion in the Bases of SR 3.0.1. Currently, only Section 3.8 contains the Note which states that "credit may be taken for unplanned events that satisfy this SR" In addition, the Notes also contain a restriction on the MODE of performance such that the surveillance is not to be performed in a MODE where perturbation to the electrical distribution system would cause a challenge to safety systems. The intent of the Note is applicable to any SR. The revision to the Bases for SR 3.0.1 will provide the necessary clarification so that the usage of this allowance can be applied consistently throughout the Technical Specifications. This change is being tracked generically as TSTF-08.

4. 10 CFR 50, Appendix J, Option B, has been implemented, and the correct reference is now the Containment Leakage Rate Testing Program.

CHANGE NUMBER	DISCUSSION

17. A new LCO and associated Bases have been added. LCO 3.6.9, "Isolation Valve Seal Water (IVSW) System," is provided to help ensure containment leakage following the worse case design basis accident is reduced to less than the maximum value assumed in the safety analysis. This LCO is unique to Zion Station and is contained in the CTS (3.9.1). IVSW satifies Criterion 3 of the NRC Policy Statement.

18. In CTS 3.9.6, the requirement to limit the maximum opening of the containment purge supply and purge exhaust valves to 50 degrees, and the associated surveillance requirement (CTS 4.9.6.B) which verifies the valves are limited to a maximum opening of 50 degrees every 18 months, have been removed from the Technical Specifications. Permanent plant modifications have been made to the valve actuators which prevent the valves from opening greater than 50 degrees. Removal or alteration to the valve actuators such that the valves could open greater than 50 degrees would constitute a change to the facility design. Changes to the facility design must be evaluated using the criteria set forth in 10 CFR 50.59 to determine if the change involves an unreviewed safety question. Any change deemed to involve a unreviewed safety question must first be submitted for NRC approval. In addition, specific surveillances prescribing a periodic verification of the system design are considered continually met by the design control process. As such, removing these requirements from the Technical Specifications does not result in a reduction of safety since the design of the containment purge supply and purge exhaust valves ensures the valves are not opened greater than 50 degrees. As addressed above, not including this surveillance in Zion's ITS is consistent with the NUREG reviewers note stating that this surveillance requirement is not required if permanent modifications have been installed limiting valve opening to less than 50%.

19. ~~Deleted~~References to Appendix J have been changed to the Containment Leakage Rate Testing Program following implementation of 10 CFR 50, Appendix J, Option B.

20. This change to the Zion ITS exempts certain automatic containment isolation valves from the 18 month surveillance testing that would demonstrate satisfactory operation. The valves are exempted because they are locked, sealed, or otherwise secured in the required position under administrative controls. This provision is not contained in NUREG 1431, Rev 0, but it is incorporated into NUREG 1431, Rev 1.

previously approved on the Ginna docket during conversion to the ITS on the basis that it duplicates the regulations.

24. New specification 5.5.14 implements the provisions of 10 CFR 50, Appendix J, Option B. This was incorporated into the Zion license under amendment 175/162.

ATTACHMENT 2

MARK UP OF ITS CHANGE

OI 5.1.2 – Clarification of designated Control Room SRO

ITS Section 5.1.2 requires clarification in the discussion of the designated Control Room SRO, and the NOTE deleted.

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 Station Manager

The station manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The station manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

5.1.2 Shift Supervisor (SS)

~~NOTE~~
~~A single SS may be responsible for the control room command function for both units if one or both units are in MODES 5 or 6.~~

~~The~~**An** SS shall be responsible for the control room command function. During any absence of the **designated Control Room Command** SS from the control room while ~~the~~**either** unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SS from the control room while ~~the unit is~~**both units are** in MODE 5 or 6, **or defueled,** an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

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5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

5.1.1 Station Manager

The station manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The station manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

5.1.2 Shift Supervisor (SS)

An SS shall be responsible for the control room command function. During any absence of the designated Control Room Command SS from the control room while either unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SS from the control room while both units are in MODE 5 or 6, or defueled, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

NUREG MARKUPS

5.0 ADMINISTRATIVE CONTROLS

5.1 Responsibility

CTS 6.1.1.b

5.1.1 Station Manager (1)

The ~~[Plant Superintendent]~~ station manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

The ~~[Plant Superintendent]~~ station manager or his designee shall approve, prior to implementation, each proposed test, experiment or modification to systems or equipment that affect nuclear safety.

Fig 6.1-1

5.1.2 (25)

A ~~The~~ ~~[Shift Supervisor (SS)]~~ shall be responsible for the designated Control Room Command control room command function. During any absence of the [SS] from the control room while ~~the~~ either unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the [SS] from the control room while ~~the~~ both units ~~is~~ are in MODE 5 or 6, or defueled, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

DOD CHANGES

previously approved on the Ginna docket during conversion to the ITS on the basis that it duplicates the regulations.

24. New specification 5.5.14 implements the provisions of 10 CFR 50, Appendix J, Option B. This was incorporated into the Zion license under amendment 175/162.

25. This specification is rewritten to reflect current regulatory requirements by one individual with an SRO license assuming control room command responsibility for both units. With both units in MODES 5, 6, or defueled, an individual with either an SRO or RO license may be designated to assume the control room command function.