

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>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.</p> <p>The leakage rate acceptance criterion is \leq 1.0 L_a. However, during the first unit startup following testing performed in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, the leakage rate acceptance criteria are $<$ 0.6 L_a for the Type B and Type C tests, and $<$ 0.75 L_a for the Type A test.</p>	<p>-----NOTE----- SR 3.0.2 is not applicable -----</p> <p>In accordance with, 10 CFR 50, Appendix J, as modified by approved exemptions</p>

THE CONTAINMENT LEAKAGE RATE TESTING PROGRAM.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.6.2.1</p> <p>-----NOTES-----</p> <p>1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.</p> <p>2. Results shall be <u>evaluated</u> against acceptance criteria of SR 3.6.1.1 in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.</p> <p>-----</p> <p>Perform required air lock leakage rate testing in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.</p> <p>The acceptance criteria for air lock testing are:</p> <p>a. Overall air lock leakage rate is ≤ 0.05 L, when tested at ≥ 15.0 psig.</p> <p>b. For each door, leakage rate is ≤ 0.01 L, when tested at ≥ 6 psig.</p>	<p>-----NOTE-----</p> <p>SR 3.0.2 is not applicable</p> <p>-----</p> <p>In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.</p>
<p>SR 3.6.2.2</p> <p>-----NOTE-----</p> <p>Only required to be performed upon entry or exit through the containment air lock.</p> <p>-----</p> <p>Verify only one door in the air lock can be opened at a time.</p>	<p>184 days</p>

Applicable to

THE CONTAINMENT LEAKAGE RATE TESTING PROGRAM.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
<p>SR 3.6.3.8 Verify the combined leakage rate for all shield building bypass leakage paths is $\leq 0.25 L_s$ when pressurized to ≥ 15.0 psig.</p>	<p>-----NOTE----- SR 3.0.2 is not applicable. -----</p> <p>In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions</p>

THE CONTAINMENT LEAKAGE RATE TESTING PROGRAM.

5.7 Procedures, Programs, and Manuals

5.7.2.18 Safety Function Determination Program (SFDP) (continued)

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.

Insert 5.7.2.19 (Attached)

5.7.2.19 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. Visual examination and testing, including test intervals and extensions, shall be in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a , is 15.0 psig.

The maximum allowable containment leakage rate, L_a , at P_a , is 0.25% of the primary containment air weight per day.

Leakage rate acceptance criteria are:

- a. Containment overall leakage rate acceptance criterion is $< 1.0 L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60 L_a$ for the combined Type B and Type C tests, and $\leq 0.75 L_a$ for Type A tests. At all other times, leakage rate acceptance criteria are as required by the Containment Leakage Rate Testing Program.
- b. Air lock testing acceptance criteria are:
 - 1) Overall air lock leakage rate is $\leq 0.05 L_a$ when tested at $\geq P_a$.
 - 2) For each door, leakage rate is $\leq 0.01 L_a$ when pressurized to ≥ 6 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program. Criteria for extending test intervals for scheduling purposes are addressed within the Program.

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

INSERT AFTER SECTION 5.7.2.18, page 5.0-28

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.

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).

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.~~

DELETE

in the Containment Leakage Rate Testing Program; criteria for extending test intervals for scheduling purposes are addressed within the program. (continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND The containment is a free standing steel pressure vessel surrounded by a reinforced concrete shield building. The containment vessel, including all its penetrations, is a low leakage steel shell designed to contain the radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). Additionally, the containment and shield building provide shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment vessel is a vertical cylindrical steel pressure vessel with hemispherical dome and a concrete base mat with steel membrane. It is completely enclosed by a reinforced concrete shield building. An annular space exists between the walls and domes of the steel containment vessel and the concrete shield building to provide for the collection, mixing, holdup, and controlled release of containment out leakage. Ice condenser containments utilize an outer concrete building for shielding and an inner steel containment for leak tightness.

Containment piping penetration assemblies provide for the passage of process, service, sampling, and instrumentation pipelines into the containment vessel while maintaining containment integrity. The shield building provides shielding and allows controlled filtered release of the annulus atmosphere under accident conditions, as well as environmental missile protection for the containment vessel and Nuclear Steam Supply System.

The inner steel containment 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.

OPTION B

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:

(continued)

BASES

BACKGROUND
(continued)

- a. All penetrations required to be closed during accident conditions are either:
 1. capable of being closed by an OPERABLE automatic containment isolation system, or
 2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves."
- b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks."
- c. All equipment hatches are closed.

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 rates.

The DBAs that result in a challenge to containment-OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA), a steam line break (SLB), and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% of containment air weight per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J (Ref. 1), as L_c : the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_c) ~~resulting~~ *related to* ~~from the limiting DBA~~. The allowable leakage rate represented by L_c forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_c is assumed to be 0.25% per day in the safety analysis at $P_c = 15.0$ psig which bounds the calculated peak containment

OPTION B
DESIGN BASIS
LOCA

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

internal pressure resulting from the limiting design basis LOCA (Ref. 3).

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

The containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

the leakage rate acceptance criteria described in the Containment Leakage Rate Test Program.

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

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

Individual leakage rates specified for the containment air lock (LCO 3.6.2), purge valves with resilient seals, and shield building containment bypass leakage (LCO 3.6.3) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the acceptance criteria of Appendix J.

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, "Containment Penetrations."

(continued)

BASES (continued)

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 OPERABLE 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.

B.1 and B.2

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

SURVEILLANCE
REQUIREMENTS

SR 3.6.1.1

The Containment Leakage Rate Testing Program

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, shield building containment bypass leakage path, and purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required ~~10 CFR 50, Appendix J~~, leakage test is required to be $< 0.6 L_a$ for combined Type B and C leakage, and $< 0.75 L_a$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $< 1.0 L_a$. At $\leq 1.0 L_a$, the offsite dose consequences are bounded by the assumptions of the safety analysis.

Following an outage or shutdown that included Type B and C testing only,

following an outage or shutdown that included Type A testing.

(continued)

Containment Leakage Rate Testing Program

described in the Containment Leakage Rate Testing Program assures that the overall primary leakage is limited to $\leq 1.0 L_a$

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.1.1 (continued)

the Containment Leakage Rate Testing Program, therefore,

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.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors" *(Option B)*
2. Watts Bar FSAR, Section 15.0, "Accident Analysis."
3. Watts Bar FSAR, Section 6.2, "Containment Systems."

4. Regulatory Guide 1.163,

-Performance-Based Requirements."

"Performance-Based Containment Leak-Test Program," September 1995.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The DBAs that result in a significant 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 0.25% of containment air weight per day (Ref. 2). ~~This leakage rate is defined in 10 CFR 50, Appendix J (Ref. 1), as $L_a = 0.25\%$ of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a following a DBA. A P_a value of 15.0 psig is utilized which bounds the calculated peak containment internal pressure 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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The containment air locks satisfy Criterion 3 of the NRC Policy Statement.

LCO

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

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

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BASES (continued)

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 D (2.50.1) as modified by approved~~
~~Appendix D~~ This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by: ~~Appendix D (2.50.1) as modified by approved~~
~~10 CFR 50 Appendix D (2.50.1) as modified by approved~~
~~Appendix D (2.50.1) as modified by approved~~ the Containment Leakage Rate Testing Program.

THE CONTAINMENT
LEAKAGE RATE
TESTING PROGRAM

therefore, SR 3.0.2
(which allows
Frequency extensions)
does not apply.

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

applicable to

as defined in the
containment leakage
Rate Testing Program.

COMBINED TYPE B AND C.

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.

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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.2.2 (continued)

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. The 184 day Frequency is based on engineering judgment and is considered adequate in view of other indications of door status available to operations personnel and because the interlock is only disabled in MODES 5 and 6.

REFERENCES

- (OPTION B)
1. Title 10, Code of Federal Regulations, Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors."
 2. Watts Bar FSAR, Section 15.0, "Accident Analysis."
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- Performance - Based
Requirements

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.3.4 (continued)

isolation time and Frequency of this SR are in accordance with the Inservice Testing Program or 92 days.

SR 3.6.3.5

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, is required to ensure OPERABILITY.

OPTION B, (Ref. 4)
Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 3).

Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that occurring to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.

SR 3.6.3.6

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative control. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.3.6 (continued)

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

SR 3.6.3.7

Verifying that each 24 inch containment lower compartment purge valve is blocked to restrict opening to $\leq 50^\circ$ is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 1 and 2. If a LOCA occurs, the purge valves must close to maintain containment leakage within the values assumed in the accident analysis. At other times when purge valves are required to be capable of closing (e.g., during movement of irradiated fuel assemblies), pressurization concerns are not present, thus the purge valves can be fully open. The 18 month Frequency is appropriate because the blocking devices are typically removed only during a refueling outage.

SR 3.6.3.8

The as-left bypass leakage rate prior to the first startup after performing a required Containment Leakage Rate Test program leakage requires calculation using maximum pathway leakage.

This SR ensures that the combined leakage rate of all shield building bypass leakage paths is less than or equal to the specified leakage rate. This provides assurance that the assumptions in the safety analysis are met. ~~The leakage rate of each bypass leakage path is assumed to be the maximum pathway leakage (leakage through the worse of the two isolation valves), unless the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange. In this case, the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves. This method of quantifying maximum pathway leakage is only to be used for this SR (i.e., Appendix J maximum pathway leakage limits are to be quantified in accordance with Appendix J).~~

At all other times the leakage rate will be calculated using minimum pathway leakage.

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(continued)

Therefore, SR 3.0.2 (which allows Frequency extensions) does not apply.

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.3.8 (continued)

the Containment Leakage Rate Testing Program

~~The frequency is required by 10 CFR 50, Appendix J, as modified by approved exemptions and therefore, the frequency extensions of SR 3.0.2 may not be applied since the testing is an Appendix J Type C test. This SR simply imposes additional acceptance criteria. Although not a part of L_a, the Shield Building Bypass leakage path combined leakage rate is determined using the 10 CFR 50, Appendix J, Type B and C leakage rates for the applicable barriers.~~

(OPTION B)

REFERENCES

1. Watts Bar FSAR, Section 15.0, "Accident Analysis."
2. Watts Bar FSAR, Section 6.2.4.2, "Containment Isolation System Design," and Table 6.2.4-1, "Containment Penetrations and Barriers."
3. Generic Issue B-20, "Containment Leakage Due to Seal Deterioration."

(Option B)

4. Title 10, Code of Federal Regulations, Part 50, Appendix - J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors - Performance-Based Requirements."

D. The following exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. Therefore, these exemptions are granted pursuant to 10 CFR 50.12.

(1) Deleted

(2) The facility was previously granted an exemption from the criticality monitoring requirements of 10 CFR 70.24 (see Special Nuclear Material License No. SNM-1861 dated September 5, 1979). The technical justification is contained in Section 9.1 of Supplement 5 to the Safety Evaluation Report, and the staff's environmental assessment was published on April 18, 1985 (50 FR 15516). The facility is hereby exempted from the criticality alarm system provisions of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license.

(3) The facility requires an exemption from 10 CFR 73.55(c)(10). The justification for this exemption is contained in Section 13.6.9 of Supplement 15 and 20 to the Safety Evaluation Report. The staff's environmental assessment was published on April 25, 1995 (60 FR 20291). Pursuant to 10 CFR 73.5, the facility is exempted from the stated implementation schedule of the surface vehicle bomb rule, and may implement the same as late as February 17, 1996.

(4) The facility was previously granted an exemption from certain requirements of 10 CFR 73.55(d)(5) relating to the returning of picture badges upon exit from the protected areas, such that individuals not employed by TVA who are authorized unescorted access into protected areas can take their badges offsite (see 59 FR 66061, December 22, 1994). The granting of this exemption is hereby affirmed.

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 the Containment Leakage Rate Testing Program.	-----NOTE----- SR 3.0.2 is not applicable ----- In accordance with the Containment Leakage Rate Testing Program.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.6.2.1</p> <p>-----NOTES-----</p> <ol style="list-style-type: none"> 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. <p>-----</p> <p>Perform required air lock leakage rate testing in accordance with the Containment Leakage Rate Testing Program.</p>	<p>-----NOTE-----</p> <p>SR 3.0.2 is not applicable</p> <p>-----</p> <p>In accordance with the Containment Leakage Rate Testing Program.</p>
<p>SR 3.6.2.2</p> <p>-----NOTE-----</p> <p>Only required to be performed upon entry or exit through the containment air lock.</p> <p>-----</p> <p>Verify only one door in the air lock can be opened at a time.</p>	<p>184 days</p>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.6.3.8 Verify the combined leakage rate for all shield building bypass leakage paths is $\leq 0.25 L_0$ when pressurized to ≥ 15.0 psig.	-----NOTE----- SR 3.0.2 is not applicable. ----- In accordance with the Containment Leakage Rate Testing Program

5.7 Procedures, Programs, and Manuals

5.7.2.18 Safety Function Determination Program (SFDP) (continued)

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.

5.7.2.19 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. Visual examination and testing, including test intervals and extensions, shall be in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a , is 15.0 psig.

The maximum allowable containment leakage rate, L_a , at P_a , is 0.25% of the primary containment air weight per day.

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.19 Containment Leakage Rate Testing Program (continued)

Leakage rate acceptance criteria are:

- a. Containment overall leakage rate acceptance criterion is $< 1.0 L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60 L_a$ for the combined Type B and Type C tests, and $\leq 0.75 L_a$ for Type A tests. At all other times, leakage rate acceptance criteria are as required by the Containment Leakage Rate Testing Program.
- b. Air lock testing acceptance criteria are:
 - 1) Overall air lock leakage rate is $\leq 0.05 L_a$ when tested at $\geq P_a$.
 - 2) For each door, leakage rate is $\leq 0.01 L_a$ when pressurized to ≥ 6 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program. Criteria for extending test intervals for scheduling purposes are addressed within the Program.

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

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.

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).

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 in the Containment Leakage Rate Testing Program; criteria for extending test intervals for scheduling purposes are addressed within the Program.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1 Containment

BASES

BACKGROUND

The containment is a free standing steel pressure vessel surrounded by a reinforced concrete shield building. The containment vessel, including all its penetrations, is a low leakage steel shell designed to contain the radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). Additionally, the containment and shield building provide shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment vessel is a vertical cylindrical steel pressure vessel with hemispherical dome and a concrete base mat with steel membrane. It is completely enclosed by a reinforced concrete shield building. An annular space exists between the walls and domes of the steel containment vessel and the concrete shield building to provide for the collection, mixing, holdup, and controlled release of containment out leakage. Ice condenser containments utilize an outer concrete building for shielding and an inner steel containment for leak tightness.

Containment piping penetration assemblies provide for the passage of process, service, sampling, and instrumentation pipelines into the containment vessel while maintaining containment integrity. The shield building provides shielding and allows controlled filtered release of the annulus atmosphere under accident conditions, as well as environmental missile protection for the containment vessel and Nuclear Steam Supply System.

The inner steel containment 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 (Ref. 1), as modified by approved exemptions.

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

(continued)

BASES

BACKGROUND
(continued)

- a. All penetrations required to be closed during accident conditions are either:
 1. capable of being closed by an OPERABLE automatic containment isolation system, or
 2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves."
 - b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks."
 - c. All equipment hatches are closed.
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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 rates.

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA), a steam line break (SLB), and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.25% of containment air weight per day (Ref. 3). This leakage rate, used in the evaluation of offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a : the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) related to the design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.25% per day in the safety analysis at $P_a = 15.0$ psig which bounds the calculated peak containment internal pressure resulting from the limiting design basis LOCA (Ref. 3).

(continued)

BASES

APPLICABLE
SAFETY ANALYSES
(continued)

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

The containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

Containment OPERABILITY is maintained by limiting leakage to the leakage rate acceptance criteria described in the Containment Leakage Rate Testing Program.

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

Individual leakage rates specified for the containment air lock (LCO 3.6.2), purge valves with resilient seals, and shield building containment bypass leakage (LCO 3.6.3) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J, Option B. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the acceptance criteria of Appendix J, 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, "Containment Penetrations."

(continued)

BASES (continued)

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 OPERABLE 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.

B.1 and B.2

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

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 air lock, shield building containment bypass leakage path, and purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be $< 0.6 L_a$ for combined Type B and C leakage following an outage or shutdown that included Type B and C testing only, and $\leq 0.75 L_a$ for overall Type A leakage following an outage or shutdown that included Type A testing. At all other times, the acceptance criteria described in the Containment Leakage Rate Testing Program assures that the overall primary

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.1.1 (continued)

containment leakage rate is limited to $\leq 1.0 L_a$. 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, therefore, 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.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix J (Option B), "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors - Performance-Based Requirements."
 2. Watts Bar FSAR, Section 15.0, "Accident Analysis."
 3. Watts Bar FSAR, Section 6.2, "Containment Systems."
 4. Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995.
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BASES (continued)

APPLICABLE
SAFETY ANALYSES

The DBAs that result in a significant 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 (L_a) of 0.25% of containment air weight per day (Ref. 2). 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 exit from containment.

(continued)

BASES (continued)

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 acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate. The Frequency is required by the Containment Leakage Rate Testing Program, therefore, SR 3.0.2 (which allows Frequency extensions) does not apply.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria applicable to SR 3.6.1.1, as defined in the Containment Leakage Rate Testing Program. 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.

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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.2.2 (continued)

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. The 184 day Frequency is based on engineering judgment and is considered adequate in view of other indications of door status available to operations personnel and because the interlock is only disabled in MODES 5 and 6.

REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix J (Option B), "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors - Performance-Based Requirements."
 2. Watts Bar FSAR, Section 15.0, "Accident Analysis."
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BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.3.4 (continued)

isolation time and Frequency of this SR are in accordance with the Inservice Testing Program or 92 days.

SR 3.6.3.5

For containment purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B (Ref. 4), is required to ensure OPERABILITY.

Operating experience has demonstrated that this type of seal has the potential to degrade in a shorter time period than do other seal types. Based on this observation and the importance of maintaining this penetration leak tight (due to the direct path between containment and the environment), a Frequency of 184 days was established as part of the NRC resolution of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration" (Ref. 3).

Additionally, this SR must be performed within 92 days after opening the valve. The 92 day Frequency was chosen recognizing that cycling the valve could introduce additional seal degradation (beyond that occurring to a valve that has not been opened). Thus, decreasing the interval (from 184 days) is a prudent measure after a valve has been opened.

SR 3.6.3.6

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative control. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.3.6 (continued)

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

SR 3.6.3.7

Verifying that each 24 inch containment lower compartment purge valve is blocked to restrict opening to $\leq 50^\circ$ is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 1 and 2. If a LOCA occurs, the purge valves must close to maintain containment leakage within the values assumed in the accident analysis. At other times when purge valves are required to be capable of closing (e.g., during movement of irradiated fuel assemblies), pressurization concerns are not present, thus the purge valves can be fully open. The 18 month Frequency is appropriate because the blocking devices are typically removed only during a refueling outage.

SR 3.6.3.8

This SR ensures that the combined leakage rate of all shield building bypass leakage paths is less than or equal to the specified leakage rate. This provides assurance that the assumptions in the safety analysis are met. The as-left bypass leakage rate prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test requires calculation using maximum pathway leakage (leakage through the worse of the two isolation valves). At all other times the leakage rate will be calculated using minimum pathway leakage. If the penetration is isolated by use of one closed and de-activated automatic valve, closed manual valve, or blind flange, then the leakage rate of the isolated bypass leakage path is assumed to be the actual pathway leakage through the isolation device. If both isolation valves in the penetration are closed, the actual leakage rate is the lesser leakage rate of the two valves.

(continued)

BASES

SURVEILLANCE
REQUIREMENTS

SR 3.6.3.8 (continued)

The frequency is required by the Containment Leakage Rate Testing Program, therefore, SR 3.0.2 (which allows frequency extensions) does not apply. This SR simply imposes additional acceptance criteria. Although not a part of L_a , the Shield Building Bypass leakage path combined leakage rate is determined using the 10 CFR 50, Appendix J (Option B), Type B and C leakage rates for the applicable barriers.

REFERENCES

1. Watts Bar FSAR, Section 15.0, "Accident Analysis."
 2. Watts Bar FSAR, Section 6.2.4.2, "Containment Isolation System Design," and Table 6.2.4-1, "Containment Penetrations and Barriers."
 3. Generic Issue B-20, "Containment Leakage Due to Seal Deterioration."
 4. Title 10, Code of Federal Regulations, Part 50, Appendix J (Option B), "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors - Performance-Based Requirements."
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