Attachment II

Marked Up Copy of R.E. Ginna Nuclear Power Plant
Technical Specifications and License

Included Pages:

License, page 5
4.4-1
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(a) Provisions establishing preventive maintenance and periodic visual inspection requirements; and

(b) Leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

(6) Iodine Monitoring

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

(a) Training of personnel;

(b) Procedures for monitoring; and

(c) Provisions for maintenance of sampling and analysis equipment.

Change 1

A.S

D. The facility requires exemptions from certain requirements of 10 CFR 50.46(a)(1), 50.48(c)(4), and Appendix J to 10 CFR Part 50. These include: (1) an exemption from 50.46(a)(1), that ECCS performance be calculated in accordance with an acceptable calculational model which conforms to the provisions in Appendix K (SER dated April 18, 1978). The exemption will expire upon receipt and approval of revised ECCS calculations; (2) certain exemptions from Appendix J to 10 CFR Part 50 section III.A.4.(a) maximum allowable leakage rate for reduced pressure tests, section III.B.1 acceptable technique for performing local (Type B) leakage rate tests, section III.D.1 scheduling of containment integrated leakage rate tests, and section III.D.2 testing interval for containment airlocks (SER dated March 28, 1978); and (3) an exemption to the schedular requirements for the alternative shutdown system as set forth in 10 CFR 50.48(c)(4) (NRC letter dated May 10, 1984). The exemption is effective until startup from the 1986 refueling outage. The aforementioned exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, the exemptions are hereby granted pursuant to 10 CFR 50.12.

E. Physical Protection - The licensee shall maintain in effect and fully implement all provisions of the following Commission-approved documents, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p), which are being withheld from public disclosure pursuant to 10 CFR 73.21:

Change 2

4.4 Containment Tests

Applicability

Applies to containment leakage and structural integrity.

Objective

To verify that potential leakage from the containment and the pre-stressing tendon loads are maintained within specified values.

Specification

~~4.4.1 Integrated Leakage Rate Test~~

~~4.4.1.1 Definitions~~

The peak calculated containment internal pressure for the design basis loss of coolant accident,

A.1 A.2

Pa ~~(psig)~~ is ~~the containment vessel design pressure of~~ 60 psig.

M.1

~~Pt (psig) is the containment vessel reduced test pressure for periodic testing.~~

~~Lt (weight percent/24 hours) is the maximum allowable leakage rates of the containment vessel test atmosphere at pressure Pt.~~

The maximum allowable primary containment leakage rate,

A.1 A.2

La ~~(weight percent/24 hours) is the maximum allowable leakage rate of the containment vessel test atmosphere~~ at ~~pressure~~ shall be Pa, 0.2% ~~24 hrs~~ of primary containment air weight per day.

Change 2

M.1

~~Lam and Ltm (weight percent/24 hours) are the total measured containment leakage rates of the containment vessel test atmosphere at pressures Pa and Pt respectively.~~

4.4.1.2 Pretest Requirements

Change 3

R.1

Retained in NEI 94-01, Section 9.2.1 and RG 1.163, Section C.3

a. A visual examination of the accessible interior and exterior surfaces of the containment structure shall be performed to uncover any evidence of structural deterioration which may affect either the containment structure integrity or leak-tightness. If there is evidence of structural deterioration, integrated leak rate testing shall not be performed until appropriate corrective action has been taken. Except for repairs to correct structural deterioration, however, no repairs or adjustments shall be made during the period between the initiation of the inspection and the performance of the test.

Change 3

R.1

Retained in ANSI/ANS 56.8-1994

b. Closure of containment isolation valves shall be accomplished by normal operation and without any preliminary exercising or adjustments.

guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

4.4.1.3 Conduct of Tests

Change 4

A.3

a. All integrated leak rate tests shall be conducted in accordance with the ~~provisions of American National Standard N45.4-1972, Leakage Rate Testing of Containment Structures for Nuclear Reactors, March 16, 1972.~~

R.1

Retained in ANSI/ANS 56.8-1994 and NEI 94-01, Section 9.2.3

L.1

b. The accuracy of each integrated leak rate test shall be verified by a supplemental test which confirms the accuracy of the test instrumentation and calculational methods provided in ANSI/ANS 56.8-1994 methods by ~~determining a leak rate which is within 0.25Lt of the test result.~~ If results are not within these methods tolerances, ~~0.25Lt~~ the reason shall be determined, corrective action taken and a successful supplemental test performed.

M.1

c. ~~Integrated leak rate tests shall be conducted at an initial pressure (beginning of test) Pt $\geq$ 35 psig.~~

R.1

Retained in ANSI/ANS 56.8-1994 and NEI 94-01, Section 9.2.3

d. If during the test, including the supplemental test, potentially excessive leakage paths are identified which will interfere with satisfactory completion of the test, or which result in the test not meeting the acceptance criteria, the test shall be terminated and the leakage through such paths shall be measured using local leakage testing methods. Repairs and/or adjustments to equipment shall be made and an integrated leak rate test performed.

4.4.1.4 Acceptance Criteria

M.1

a. ~~The leakage rate Ltm shall be <0.75 Lt at Pt. Pt is defined as the containment vessel reduced test pressure which is greater than or equal to 35 psig. Ltm is defined as the total measured containment leakage rate at pressure Pt. Lt is defined as the maximum allowable leakage rate at pressure Pt.~~

Change 5

b. ~~Lt shall be determined as Lt = La$\left(\frac{Pt}{Pa}\right)^{1/2}$ which equals .1528 percent weight per day at 35 psig. Pa is defined as the calculated peak containment internal pressure related to design basis accidents which is greater than or equal to 60 psig. La is defined as the maximum allowable leakage rate at Pa which equals .2 percent weight per day.~~

L.2

c. The leakage rate at Pa ~~(Lam)~~ shall be <0.75 La. ~~Lam is defined as the total measured containment leakage rate at pressure Pa.~~

4.4.1.5 Test Frequency

for Type A tests. Following plant startup, primary containment leakage rate acceptance criterion is ≤ 1.0 La

Change 6

a. ~~A set of three integrated leak rate tests shall be performed at approximately equal intervals during each 10-year service period. The third test of each set shall be conducted in the final year of the 10-year service period or one year before or after the final year of the 10-year service period provided:~~ Based on acceptable performance-based tests:

L.3

i. the interval between any two Type A tests shall ~~does~~ not exceed ~~four~~ 10 years, (with an additional 15 month extension allowed)

L.4

ii. following each in-service inspection, the containment airlocks, ~~the steam generator inspection/maintenance penetration,~~ and the equipment hatch are leak tested prior to returning the plant to operation, and

Retained in NEI 94-01, Sections 9.2.3, 9.2.4, and 10.2.2.1

R.1

iii. any repair, replacement, or modification of a containment barrier resulting from the inservice inspections shall be followed by the appropriate leakage test.

Change 6

L.5

Retained in NEI 94-01, Section 9.2.6

R.1

b. If any test fails to meet the acceptance criteria of 4.4.1.4.a, ~~the test schedule for subsequent regularly scheduled inservice tests shall be submitted to the Commission for review and approval.~~

~~c. If two consecutive tests fail to meet the acceptance criteria of 4.4.1.4.a,~~ a retest shall be performed ~~at each refueling shutdown or approximately every~~ within 48 ~~18~~ months~~, whichever comes first,~~ until two consecutive tests (conducted between 24 and 48 months apart) meet the acceptance criteria of 4.4.1.4.a, after which time the retest schedule of 4.4.1.5.a may be resumed.

L.6

Change 7

Retained in NEI 94-01, Section 12.1 and ANSI/ANS 56.8-1994

R.1

4.4.1.6 Additional Requirements

a. A summary technical report shall be prepared ~~submitted to the Commission~~ after the conduct of each integrated leak rate test. Information on any valve closure malfunction or valve leakage that requires corrective action before the test shall be included in the report.

Change 8

Retained in ANSI/ANS 56.8-1994

R.1

4.4.2 Local Leak Detection Tests

4.4.2.1 Test

a. Local leakage rate tests shall be performed at intervals specified in 4.4.2.4 below and at a pressure of not less than 57.6 ~~60~~ psig and not more than 60 psig

L.7

Change 8

Retained in NEI 94-01, Section 6.0 and ANSI/ANS 56.8-1994, Option B

R.1

b. The local leakage rate shall be measured for each of the following components:

i. Containment penetrations that employ resilient seals, gaskets, or sealant compounds, piping penetrations with expansion bellows and electrical penetrations with flexible metal seal assemblies.

ii. Air lock and equipment door seals.

iii. Fuel transfer tube.

iv. Isolation valves on the testable fluid systems lines penetrating the containment.

v. Other containment components, which require leak repair in order to meet the acceptance criterion for any integrated leakage rate test.

4.4.2.2 Acceptance Criterion

Change 9

Containment isolation boundaries are inoperable from a leakage standpoint when the demonstrated leakage of a single boundary or cumulative total leakage of all boundaries is greater than 0.60 La.

4.4.2.3 Corrective Action

Change 10

Retained in NEI 94-01, ANSI/ANS 56.8-1994 and LCO 3.6.1

a. If at any time it is determined that the total leakage from all penetrations and isolation boundaries exceeds 0.60 La, repairs shall be initiated immediately.

LCO 3.6.1 A.4 Change 10

b. If repairs are not completed and conformance to the acceptance criterion of 4.4.2.2 is not demonstrated within ~~48~~ 1 ~~hours,~~ hour the reactor shall be shutdown and depressurized until repairs are effected and the local leakage meets the acceptance criterion.

LCO 3.6.3. & Admin Controls A.4 Insert 1

c. If it is determined that the leakage through a mini-purge supply and exhaust line is greater than 0.05 La, ~~an engineering evaluation shall be performed and plans for corrective action developed.~~ declare the penetration inoperable and isolate within 24 hours

4.4.2.4 Test Frequency

L.4 Change 11

a. Except as specified in b. and c. below, individual penetrations and containment isolation valves shall be tested in accordance with RG 1.163 ~~10 CFR 50, Appendix J,~~ as modified by approved exemptions.

Retained in NEI 94-01, Section 10.2.1.2 R.1

b. Based on acceptable performance-based testing, the containment equipment hatch, fuel transfer tube, steam generator inspection/maintenance penetration, and shutdown purge system flanges shall be tested every 120 months (with an additional 24 months allowed) ~~at each refueling shutdown or after each use, if that be sooner.~~

M.2

INSERT 1

Air lock acceptance criteria are:

1) For each air lock, overall leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$, and

2) For each door, leakage rate is $\leq 0.01L_a$ when tested at $\geq P_a$.

Change 11

L.8

c. The containment air locks shall be tested at intervals of no more than ~~six~~ 30 months by pressurizing the space between the air lock doors. In addition, following opening of the air lock door during the interval, a test shall be performed by pressurizing between the dual seals of each door opened, within 7 days 48 hours of the opening (or 30 days if opened more frequently), unless the reactor was in the cold shutdown condition at the time of the opening or has been subsequently brought to the cold shutdown condition. A test shall also be performed by pressurizing between the dual seals of each door prior within 48 hours/of leaving the cold shutdown condition, ~~unless the doors have not been open since the last test performed either by pressurizing the space between the air lock doors or by pressurizing between the dual door seals.~~

4.4.7 Containment Hydrogen Monitors

4.4.7.1 Demonstrate that two hydrogen monitors are operable at least daily by verifying that the unit is on or in standby.

4.4.7.2 At least once per quarter perform a channel calibration using two sample gases containing known concentrations of hydrogen.

Basis:

~~The containment is designed for an accident pressure of 60 psig.[1] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure. The maximum temperature of the steam-air mixture at the peak accident pressure of 60 psig is calculated to be 286°F.~~

~~Prior to initial operation, the containment was strength tested at 69 psig and then was leak tested. The acceptance criterion for this pre-operational leakage rate test was established as 0.1% per 24 hours at 60 psig. This leakage rate was believed consistent with the construction of the containment,(2) which is equipped with independent leak-testable penetrations and contains channels over all containment liner welds, which were independently leak tested during construction.~~

~~Safety analyses have been performed on the basis of a leakage rate of 0.20% per 24 hours at 60 psig. With this leakage rate and with minimum containment engineered safeguards operating (i.e., either 2 filter units and no spray, or 1 filter unit and 1 spray, or no filter units and 2 sprays) the public exposure would be well below 10 CFR 100 values in the event of the design basis accident.(3)~~

~~Performance of the integrated leakage rate test provides an over-all assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions, the test is to be performed without preliminary leak detection surveys or leak repairs, and containment isolation valves are to be closed in the normal manner. The test pressure of 35 psig for the integrated leakage rate test is sufficiently high to provide an accurate measurement of the leakage rate and it duplicates the preoperational leakage rate test at 35 psig.~~

The Specification also allows for possible deterioration of the leakage rate between tests, by requiring that the total measured leakage rate be only 75% of the maximum allowable leakage rate.

The duration and methods for the integrated leakage rate test established by ANSI N45.4-1972 provide a minimum level of accuracy and allow for daily cyclic variation in temperature and thermal radiation. The frequency of the integrated leakage rate test is keyed to the refueling schedule for the reactor, because these tests can best be performed during refueling shutdowns. Refueling shutdowns are scheduled at approximately one year intervals.

The specified frequency of integrated leakage rate tests is based on three major considerations. First is the low probability of leaks in the liner, because of (a) the use of weld channels to test the leaktightness of the welds during erection, (b) conformance of the complete containment to a 0.1% per day leak rate at 60 psig during preoperational testing, and (c) absence of any significant stresses in the liner during reactor operation. Second is the more frequent testing, at the full accident pressure, of those portions of the containment envelope that are most likely to develop leaks during reactor operation (penetrations and isolation valves) and the low value (0.60 La) of the total leakage that is specified as acceptable. Third is the tendon stress surveillance program, which provides assurance than an important part of the structural integrity of the containment is maintained.

~~The basis for specification of a total leakage of 0.60 La from penetrations and isolation boundaries is that only a portion of the allowable integrated leakage rate should be from those sources in order to provide assurance that the integrated leakage rate would remain within the specified limits during the intervals between integrated leakage rate tests. Because most leakage during an integrated leak rate test occurs though penetrations and isolation valves, and because for most penetrations and isolation valves a smaller leakage rate would result from an integrated leak test than from a local test, adequate assurance of maintaining the integrated leakage rate within the specified limits is provided.~~ The limiting leakage rates from the Recirculation Heat Removal Systems are judgement values based primarily on assuring that the components could operate without mechanical failure for a period on the order of 200 days after a design basis accident. The test

The pre-stress confirmation test provides a direct measure of the load-carrying capability of the tendon.

If the surveillance program indicates by extensive wire breakage or tendon stress relation that the pre-stressing tendons are not behaving as expected, the situation will be evaluated immediately. The specified acceptance criteria are such as to alert attention to the situation well before the tendon load-carrying capability would deteriorate to a point that failure during a design basis accident might be possible. Thus the cause of the incipient deterioration could be evaluated and corrective action studied without need to shut down the reactor. The containment is provided with two readily removable tendons that might be useful to such a study. In addition, there are 40 tendons, each containing a removable wire which will be used to monitor for possible corrosion effects.

Operability of the containment isolation boundaries ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Performance of cycling tests and verification of isolation times associated with automatic containment isolation valves is covered by the Pump and Valve Test Program. Compliance with Appendix J to 10 CFR 50 is addressed under local leak testing requirements.

References:

(1) ~~UFSAR Section 3.1.2.2.7~~ Not used.

(2) ~~UFSAR Section 6.2.6.1~~ Not used.

(3) ~~UFSAR Section 15.6.4.3~~ Not used.

(4) UFSAR Section 6.3.3.8

(5) UFSAR Table 15.6-9

(6) FSAR Page 5.1.2-28

(7) North-American-Rockwell Report 550-x-32, Autonetics Reliability Handbook, February 1963.

(8) FSAR Page 5.1.2-28

Attachment III

Marked Up Copy of Improved Technical Specifications Submitted in LAR
Dated May 26, 1995 to Reflect NRC Implementation Guidance

Included Pages:

Attachment C

1.1 Definitions (continued)

$\bar{E}$ — AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies (in MeV) per disintegration for non-iodine isotopes, with half lives > 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.
~~L_a~~	~~The maximum allowable primary containment leakage rate, L_a, shall be 0.2% of primary containment air weight per day at the calculated peak containment pressure (P_a).~~
LEAKAGE	LEAKAGE from the RCS shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or return), that is captured and conducted to collection systems or a sump or collecting tank; 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or 3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System; b. <u>Unidentified LEAKAGE</u> All LEAKAGE (except RCP seal water injection or return) that is not identified LEAKAGE;

(continued)

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 and containment ~~mini-purge valve testing, in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions~~ the Containment Leakage Rate Testing Program	In accordance ~~with 10 CFR 50, Appendix J, as modified by approved exemptions~~
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

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 applicable to evaluated against acceptance criteria ~~of~~ SR 3.6.1.1 ~~in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.~~ ~~3. SR 3.0.2 is not applicable.~~ --- Perform required air lock leakage rate testing in accordance with the Containment Leakage Testing Program ~~10 CFR 50, Appendix J, as modified by approved exemptions.~~ ~~The acceptance criteria for air lock testing are:~~ ~~a. Leakage rate for each air lock is $\leq 0.05\ L_a$ when tested at $\geq P_a$.~~ ~~b. Leakage rate for each door is $\leq 0.01\ L_a$ when tested at $\geq P_a$.~~	In accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~
SR 3.6.2.2	Verify only one door in each air lock can be opened at a time.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.4	Perform required leakage rate testing of containment mini-purge valves with resilient seals in accordance with the Containment Leakage Rate Testing Program ~~10 CFR 50, Appendix J, as modified by approved exemptions.~~ ~~The acceptance criteria for each mini-purge valve with resilient seals is $\leq 0.05\ L_a$ when tested at $\geq P_a$.~~	In accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions.~~
SR 3.6.3.5	Verify each automatic containment isolation valve that is not locked, sealed, or otherwise secured in the required position actuates to the isolation position on an actual or simulated actuation signal.	24 months

5.5 Programs and Manuals

5.5.14 SFDP (continued)

c. A required system or trains redundant to the inoperable support system(s) or train 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.

5.5.15 Secondary Water Chemistry Program

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

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;

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.

Insert 5.5.16, Containment Leakage Rate Testing Program

5.5.16 Containment Leakage Rate Testing Program

Revised CTS 4.4.1.3

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.

Revised CTS 4.4.1.1

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 60 psig.

Revised CTS 4.4.1.1

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.2% of containment air weight per day.

Leakage Rate acceptance criteria are:

Revised CTS 4.4.1.4.c and CTS 4.4.2.2

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

NEW

b. Air lock testing acceptance criteria are:

   1) For each air lock, overall leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$, and

   2) For each door, leakage rate is $\leq 0.01 L_a$ when tested at $\geq P_a$.

CTS 4.4.2.3.c

c. Mini-purge valve acceptance criteria is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

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 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

BASES

SR 3.0.2 (continued)

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.~~

~~Therefore, there is a Note in the Frequency stating, "SR 3.0.2 is not applicable."~~

Therefore, when a test interval is specified in the regulations, the test interval cannot be exceeded by TS, and the SR includes a Note in the Frequency stating, "SR 3.0.2 is not applicable." An example of an exception when the test interval is not specified in the regulations is the Note in the Containment Leakage Rate Testing Program, "SR 3.0.2 is not applicable." This exception is provided because the program already includes extension of test intervals.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that 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 intervals) or periodic Completion Time intervals beyond those specified.

(continued)

BASES

BACKGROUND (continued)

The cylinder wall is connected to sandstone rock located beneath the containment by use of 160 post-tensioned rock anchors that are coupled with tendons located in the cylinder wall. This design ensures that the rock acts as an integral part of the containment structure.

The concrete containment structure is required for structural integrity of the containment under DBA conditions. The steel liner and its penetrations establish the leakage limiting boundary of the containment. Maintaining the containment OPERABLE limits the leakage of fission product radioactivity from the containment to the outside environment to within the limits of 10 CFR 100 (Ref. 3). SR 3.6.1.1 leakage rate requirements comply with 10 CFR 50, Appendix J, Option B (Ref. 4), as modified by approved exemptions.

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

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

   1. Capable of being closed by an OPERABLE automatic containment isolation system, or

   2. Closed by OPERABLE containment isolation barriers, except as provided in LCO 3.6.3, "Containment Isolation Barriers."

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

c. All equipment and personnel hatches or doors are closed when the air lock is not being used for entry into and exit from containment.

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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA), a steam line break, and a rod ejection accident (REA) (Ref. 5). 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 originally strength tested at 69 psig (115% of design). The acceptance criteria for this test was 0.1% of the containment air weight per day at 60 psig which was based on the construction techniques that were used (Ref. 5). Following successful completion of this test, the accident analyses were performed assuming a leakage rate of 0.2% of the containment air weight per day. This leakage rate, in combination with the minimum containment engineered safeguards operating (i.e., either 2 post-accident charcoal filter trains and no containment spray, 1 post-accident charcoal filter train and 1 containment spray train, or no post-accident charcoal filter trains and 2 containment spray trains) results in offsite doses well within the limits of 10 CFR 100 (Ref. 3) in the event of a DBA.

The leakage rate of 0.2% of the containment air weight per day is defined in 10 CFR 50, Appendix J, Option B (Ref. 5), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the ~~limiting DBA~~ 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 0.2% per day in the safety analysis at P_a = ~~59.8~~ 60 psig ~~(Ref. 5)~~.

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 entering MODE 4 for the first time following performance of periodic testing performed in accordance with 10 CFR 50, Appendix J, Option B. At that time, the combined Type B and C leakage must be $< 0.6\ L_a$ on a maximum pathway leakage rate (MXPLR) basis, and the overall Type A leakage must be $< 0.75\ L_a$. At all other times prior to performing as found testing, the acceptance criteria for Type B and C testing is $< 0.6\ L_a$ on a minimum pathway leakage rate (MNPLR) basis. Containment OPERABILITY is also defined by acceptable structural integrity following a DBA.

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

Individual leakage rates specified for the containment air lock (LCO 3.6.2) and mini-purge valves with resilient seals (LCO 3.6.3) and administrative limits for individual isolation barriers 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 for Type A, B, and C tests.

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 MODES 5 and 6 to prevent leakage of radioactive material from containment.

(continued)

BASES (continued)

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. 4), as modified by approved exemptions (Refs. 6 and 7)~~. Failure to meet air lock and mini-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 these limits to be exceeded. As left leakage prior to entering MODE 4 for the first time following performance of required 10 CFR 50, Appendix J periodic testing, is required to be < 0.6 L_a for combined Type B and C leakage on a MXPLR basis, and < 0.75 L_a for overall Type A leakage (Ref. ~~8~~ 6). At all other times between the required leakage tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. This is maintained by limiting combined Type B and C leakage to < 0.6 L_a on a MXPLR basis until performance of as found testing. At ≤ 1.0 L_a, the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by ~~Appendix J, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply.~~ These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.2

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are generally consistent with the recommendations of Regulatory Guide 1.35 (Ref. ~~9~~ 7) except that tendon material tests and inspections are not required (Ref. ~~10~~ 8).

BASES (continued)

REFERENCES

1. Atomic Industry Forum, GDC 10 and 49, issued for comment July 10, 1967.
2. UFSAR, Section 3.8.1.
3. 10 CFR 100.
4. 10 CFR 50, Appendix J, Option B
5. UFSAR, Section 6.2.
6. ~~Letter from D. L. Ziemann, NRC, to L. D. White, RG&E, Subject: "Amendment No. 17 to Provisional Operating License," dated March 28, 1978.~~
7. ~~Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "Completion of Appendix J Review," dated May 6, 1981.~~
8. ~~Regulatory Guide DG-1037.~~

7 ~~9.~~ Regulatory Guide 1.35, Revision 2.

8 ~~10.~~ Letter from J. A. Zwolinski, NRC, to R. W. Kober, RG&E, Subject: "Safety Evaluation Containment Vessel Tendon Surveillance Program," dated August 19, 1985.

6. NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 0.

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 0.2% of containment air weight per day (Ref. 1). This leakage rate is defined in 10 CFR 50, Appendix J , Option B (Ref. 2), as L_a = 0.2% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a = ~~59.8~~ 60 psig following ~~a~~ ~~DBA~~ the 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

The equipment hatch and personnel hatch containment air locks form part of the containment pressure boundary. As part of containment, the air lock safety function is related to control of the containment leakage rate following 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 10 CFR 50, Appendix J Type B air lock leakage test (i.e., SR 3.6.2.1), and both air lock doors must be OPERABLE such that they can remain closed with leakage within acceptable limits following a DBA. The interlock allows only one door of an air lock to be opened at a 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

ACTIONS (continued)

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 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 the Containment Leakage Rate Testing Program ~~10 CFR 50, Appendix J (Ref. 2), as modified by approved exemptions (Ref. 3)~~. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria were established based on industry experience. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate.

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 requires that the results of this SR be evaluated against the acceptance criteria of ~~SR 3.6.1.1~~ the Containment Leakage Rate Testing Program. This ensures that air lock leakage is properly accounted for in determining the overall containment leakage rate. ~~Note 3 states that SR 3.0.2 (which allows Frequency extensions) does not apply since the Frequency is required by Appendix J (Ref. 2), as modified by approved exemptions. (Ref. 3)~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 airlock door is opened, this test is only required to be performed once every 24 months. The 24 month Frequency is based on engineering judgment and is considered adequate in view of other indications of door and interlock mechanism status available to operations personnel.

REFERENCES

1. UFSAR, Section 6.2.1.1.

2. 10 CFR 50, Appendix J~~.~~, Option B

3. ~~Letter from D. L. Ziemann, NRC, to L. D. White, RG&E, Subject: "Amendment No. 17 to Provisional Operating License," dated March 28, 1978.~~

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.2 (continued)

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

SR 3.6.3.3

Verifying that the isolation time of each automatic containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program.

SR 3.6.3.4

For containment mini-purge valves with resilient seals, additional leakage rate testing beyond the test requirements of 10 CFR 50, Appendix J, Option B, 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 outside environment), a leakage acceptance criteria of $\leq 0.05\ L_a$ when tested at $\geq P_a$ is specified for each mini-purge isolation valve with resilient seals. The Frequency of testing is specified in ~~10 CFR 50, Appendix J, as modified by approved exemptions (Refs. 6 and 7)~~ the Containment Leakage Rate Testing Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.5

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

REFERENCES

1. Atomic Industry Forum GDC 53 and 57, issued for comment July 10, 1967.
2. Branch Technical Position CSB 6-4, "Containment Purging During Normal Operation."
3. UFSAR, Section 6.2.4 and Table 6.2-15.
4. 10 CFR 50, Appendix A, GDC 55, 56, and 57.
5. Ginna Station Procedure A-3.3.
6. ~~Letter from D. L. Ziemann, NRC, to L. D. White, RG&E, Subject: "Amendment No. 17 to Provisional Operating License," dated March 28, 1978.~~
7. ~~Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: "Completion of Appendix J Review," dated May 6, 1981.~~