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DEFINITIONS

SECTION

 {1.20 L_a .... 1-3}

DEFINITIONS

SECTION

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIST OF TABLES (Continued)

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME (Continued)

breaker trip coil from when the monitored parameter exceeds its trip setpoint at the channel sensor of the associated:

a. Turbine stop valves, and
b. Turbine control valves.

The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

FRACTION OF LIMITING POWER DENSITY

1.14 The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.

FRACTION OF RATED THERMAL POWER

1.15 The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

FREQUENCY NOTATION

1.16 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

1.17 A GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.18 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

1.19 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

LIMITING CONTROL ROD PATTERN

1.20 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

Add "INSERT A"

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INSERT A

$\underline{L_a}$

1.20 The maximum allowable primary containment leakage rate, L_a, shall be 0.635 % of primary containment air weight per day at the calculated peak containment pressure (P_a = 39.6 psig).

LINEAR HEAT GENERATION RATE

1.2~~1~~2 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.2~~2~~3 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e., all relays and contacts, all trip units, solid state logic elements, etc. of a logic circuit, from sensor through and including the actuated device to verify OPERABILITY. THE LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.2~~3~~4 The MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be the highest value of the FLPD which exists in the core.

MEMBER(S) OF THE PUBLIC

1.2~~4~~5 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

MINIMUM CRITICAL POWER RATIO

1.2~~5~~6 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

OFFSITE DOSE CALCULATION MANUAL

1.2~~6~~7 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Technical Specification Section 6.2.F.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-Annual Radioactive Effluent Release Reports required by Technical Specification Sections 6.6.A.3 and 6.6.A.4.

OPERABLE - OPERABILITY

1.2~~7~~8 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, a normal and an emergency electrical power source, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION - CONDITION

1.2~~8~~9 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

1.~~29~~30 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.3~~0~~1 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.3~~1~~2 PRIMARY CONTAINMENT INTEGRITY shall exist when:

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

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

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except ~~as provided in Table 3.6.3-1 of~~ for valves that are open under administrative control as permitted by Specification~~n~~ 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is OPERABLE pursuant to Specification 3.6.1.3.

d. The primary containment leakage rates are maintained within the limits ~~of Specification 3.6.1.2.~~ per Surveillance Requirement 4.6.1.1.b.

e. The suppression chamber is OPERABLE pursuant to Specification 3.6.2.1.

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

g. Primary containment structural integrity has been verified in accordance with Surveillance Requirement 4.6.1.1.e.

PROCESS CONTROL PROGRAM

1.3~~2~~3 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.3~~3~~4 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.3~~4~~5 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3323 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.3~~5~~6 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE EVENT

1.3~~6~~7 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.3~~7~~8 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

1.3~~8~~9 SECONDARY CONTAINMENT INTEGRITY shall exist when:

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

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

2. Closed by at least one manual valve, blind flange, or deactivated automatic damper secured in its closed position, except as provided in Table 3.6.5.2-1 of Specification 3.6.5.2.

b. All secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is OPERABLE pursuant to Specification 3.6.5.3.

d. At least one door in each access to the secondary containment is closed.

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

f. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.5.1.a.

SHUTDOWN MARGIN

1.~~39~~40 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

SITE BOUNDARY

1.4~~0~~1 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SOURCE CHECK

1.4~~1~~2 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

STAGGERED TEST BASIS

1.4~~2~~3 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

1.4~~3~~4 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TURBINE BYPASS SYSTEM RESPONSE TIME

1.4~~4~~5 The TURBINE BYPASS SYSTEM RESPONSE TIME shall be time interval from when the turbine bypass control unit generates a turbine bypass valve flow signal until the turbine bypass valves travel to their required positions. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

UNIDENTIFIED LEAKAGE

1.4~~5~~6 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

VENTILATION EXHAUST TREATMENT SYSTEM

1.4~~6~~7 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.4~~7~~8 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
A. AUTOMATIC INITIATION				
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
(1) Low, Level 3	7	2	1, 2, 3	20
(2) Low Low, Level 2	2, 3	2	1, 2, 3	20
(3) Low Low Low, Level 1	1, 10	2	1, 2, 3	20
b. Drywell Pressure - High	2, 7, 10	2	1, 2, 3	20
c. Main Steam Line				
1) Radiation - High	1	2	1, 2, 3	21
	3	2	1, 2, 3	22
2) Pressure - Low	1	2	1	23
3) Flow - High	1	2/line[(d)]	1, 2, 3	21
d. Main Steam Line Tunnel Temperature - High	1	2	1[(i)(j)], 2[(i)(j)], 3[(i)(j)]	21
e. Main Steam Line Tunnel ΔTemperature - High	1	2	1[(i)(j)], 2[(i)(j)], 3[(i)(j)]	21
f. Condenser Vacuum - Low	1	2	1, 2*, 3*	21
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	4[(c)(e)]	2	1, 2, 3 and **	24
b. Drywell Pressure - High	4[(c)(e)]	2	1, 2, 3	24
c. Reactor Vessel Water Level - Low Low, Level 2	4[(c)(e)]	2	1, 2, 3, and #	24
d. Fuel Pool Vent Exhaust Radiation - High	4[(c)(e)]	2	1, 2, 3, and **	24

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	5	1	1, 2, 3	22
b. Heat Exchanger Area Temperature - High	5	1/heat exchanger	1, 2, 3	22
c. Heat Exchanger Area Ventilation ΔT - High	5	1/heat exchanger	1, 2, 3	22
d. SLCS Initiation	5(f)	NA	1, 2, 3	22
e. Reactor Vessel Water Level - Low Low, Level 2	5	2	1, 2, 3	22
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	8	1	1, 2, 3	22
b. RCIC Steam Supply Pressure - Low	8, 9(g)	2	1, 2, 3	22
c. RCIC Turbine Exhaust Diaphragm Pressure - High	8	2	1, 2, 3	22
d. RCIC Equipment Room Temperature - High	8	1	1, 2, 3	22
e. RCIC Steam Line Tunnel Temperature - High	8	1	1, 2, 3	22
f. RCIC Steam Line Tunnel Δ Temperature - High	8	1	1, 2, 3	22
g. Drywell Pressure - High	9(g)	2	1, 2, 3	22
h. RCIC Equipment Room Δ Temperature - High	8	1	1, 2, 3	22

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION				
a. RHR Equipment Area Δ Temperature - High	8	1/RHR area	1, 2, 3	22
b. RHR Area Temperature - High	8	1/RHR area	1, 2, 3	22
c. RHR Heat Exchanger Steam Supply Flow - High	8	1	1, 2, 3	22
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	6	2	1, 2, 3	25
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	6	1	1, 2, 3	25
c. RHR Pump Suction Flow - High	6	1	1, 2, 3	25
d. RHR Area Temperature - High	6	1/RHR area	1, 2, 3	25
e. RHR Equipment Area ΔT - High	6	1/RHR area	1, 2, 3	25
B. MANUAL INITIATION				
1. Inboard Valves	1, 2, 5, 6, 7	1/group	1, 2, 3	26
2. Outboard Valves	1, 2, 5, 6, 7	1/group	1, 2, 3	26
3. Inboard Valves	4 (c) (e)	1/group	1, 2, 3 and **,#	26
4. Outboard Valves	4(c) (e)	1/group	1, 2, 3 and **,#	26
5. Inboard Valves	3, 8, 9	1/valve	1, 2, 3	26
6. Outboard Valves	3, 8, 9	1/valve	1, 2, 3	26
7. Outboard Valve	8(h)	1/group	1, 2, 3	26

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN with the next 24 hours.

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 22 - Close the affected system isolation valves within 1 hour and declare the affected system inoperable.

ACTION 23 - Be in at least STARTUP within 6 hours.

ACTION 24 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

ACTION 25 - Lock the affected system isolation valves closed within 1 hour and declare the affected system inoperable.

ACTION 26 - Provided that the manual initiation function is OPERABLE for each other group valve, inboard or outboard, as applicable, in each line, restore the manual initiation function to OPERABLE status within 24 hours; otherwise, restore the manual initiation function to OPERABLE status within 8 hours; otherwise:

a. Be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, or

b. Close the affected system isolation valves within the next hour and declare the affected system in operable.

Deleted

NOTES

* May be bypassed with reactor steam pressure ≤ 1043 psig and all turbine stop valves closed.

** When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

\# During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

(a) See Specification 3.6.3, Table 3.6.3-1 for valves in each valve group.

(b) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the channel in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter. In addition for those trip systems with a design providing only one channel per trip system, the channel may be placed in an inoperable status for up to 8 hours for required surveillance testing without placing the channel in the tripped condition provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is operable and all required actuation instrumentation for that redundant valve is OPERABLE, or place the trip system in the tripped condition.

(c) Also actuates the standby gas treatment system.

(d) A channel is OPERABLE if 2 of 4 instruments in that channel are OPERABLE.

(e) Also actuates secondary containment ventilation isolation dampers per Table 3.6.5.2-1.

(f) Closes only RWCU system inlet outboard valve.

TABLE 3.3.2-3 (Continued)

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION	N/A
a. Reactor Vessel Water Level - Low, Level 3	
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	
c. RHR Pump Suction Flow - High	
d. RHR Area Cooler Temperature High	
e. RHR Equipment Area ΔT High	
B. MANUAL INITIATION	N/A
1. Inboard Valves	
2. Outboard Valves	
3. Inboard Valves	
4. Outboard Valves	
5. Inboard Valves	
6. Outboard Valves	
7. Outboard Valve	

TABLE NOTATIONS

* Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed.

** Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

\# Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to isolation time shown in Table 3.6.3-1 and 3.6.5.2-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

MSIV

the MSIVs

N/A Not Applicable.

for valves that are open under administrative control as permitted by

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2,* and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. e After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seal with gas at Pa, 39.6 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 La. e

~~b.~~ a. At least once per 31 days by verifying that all primary containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Table 3.6.3-1 of Specification 3.6.3. e e

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b.

c. By verifying each primary containment air lock OPERABLE per Specification 3.6.1.3.

d. By verifying the suppression chamber OPERABLE per Specification 3.6.2.1.

e. Verify primary containment structural integrity in accordance with the Inservice Inspection Program for Post Tensioning Tendons. The frequency shall be in accordance with the Inservice Inspection Program for Post Tensioning Tendons.

Add "INSERT C"

*See Special Test Exception 3.10.1

**Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been deinerted since the last verification or more often than once per 92 days.

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4.6.1.1.b. Perform required visual examinations and leakage rate testing except for primary containment air lock testing and main steam lines through the isolation valves, in accordance with and at the frequency# specified by 10 CFR 50, Appendix J, as modified by approved exemptions.

The overall integrated leakage rate acceptance criterion is $\leq 1.0\ L_a$. The Type B and C combined leakage rate acceptance criterion is $\leq 0.60\ L_a$. 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 $< 0.60\ L_a$ for the combined Type B and Type C tests, and $< 0.75\ L_a$ for the Type A test.

INSERT C1

\# The provisions of Specification 4.0.2 are not applicable to the frequencies specified by 10 CFR 50, Appendix J.

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LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, 0.635 percent by weight of the containment air per 24 hours at P_a, 39.6 psig.

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and all valves listed in Table 3.6.3-1, except for main steam isolation valves and valves which are hydrostatically leak tested per Table 3.6.3-1, subject to Type B and C tests when pressurized to P_a, 39.6 psig.

c. *Less than or equal to 100 scf per hour for all four main steam lines through the isolation valves when tested at 25.0 psig.

d. A combined leakage rate of less than or equal to 1 gpm times the total number of ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 P_a, 43.6 psig.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:

With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 L_a, or

b. The measured combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam isolation valves and valves which are hydrostatically leak tested per Table 3.6.3-1, subject to Type B and C tests exceeding 0.60 L_a, or

c. The measured leakage rate exceeding 100 scf per hour for all four main steam lines through the isolation valves, or

d. The measured combined leakage rate for all ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 1 gpm times the total number of such valves,

*Exemption to Appendix "J" of 10 CFR 50.

(Next page is 3/4 6-5)

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

restore:

a. The overall integrated leakage rate(s) to less than or equal to 0.75 L_a, and

b. The combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam isolation valves and valves which are hydrostatically leak tested per Table 3.6.3-1, subject to Type B and C tests to less than or equal to 0.60 L_a, and

c. The leakage rate to less than or equal to 100 scf per hour for all four main steam lines through the isolation valves, and

d. The combined leakage rate for all ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 1 gpm times the total number of such valves,

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972:

a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10 month intervals during shutdown at P_a, 39.6 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

b. If any periodic Type A test fails to meet 0.75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a, at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 L_a.

2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25% of the total measured leakage at P_a, 39.6 psig.

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SURVEILLANCE REQUIREMENTS (Continued)

d. Type B and C tests shall be conducted with gas at P_a, 39.6 psig*, at intervals no greater than 24 months except for tests involving:

1. Air locks,
2. Main steam line isolation valves,
3. Valves pressurized with fluid from a seal system, and
4. ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. Main steam line isolation valves shall be leak tested at least once per 18 months.

g. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a, 43.6 psig, and the seal system capacity is adequate to maintain system pressure for at least 30 days.

h. ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at least once per 18 months.

i. The provisions of Specification 4.0.2 are not applicable to 24 month or 40 ± 10 month surveillance intervals.

*Unless a hydraulic test is required per Table 3.6.3-1.

SURVEILLANCE REQUIREMENTS (Continued)

c. By verifying at least two suppression chamber water level instrumentation channels and at least 14 suppression pool water temperature instrumentation channels, 7 in each of two divisions, OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,

2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

3. CHANNEL CALIBRATION at least once per 18 months.

The suppression chamber water level and suppression pool temperature alarm setpoint shall be:

a) High water level $\leq$ +2 inches*

b) Low water level $\geq$ -3 inches*

c) High temperature $\leq$ 105°F

d. By conducting drywell-to-suppression chamber bypass leak tests and verifying that the $A/\sqrt{k}$ calculated from the measured leakage is within the specified limit when drywell-to-suppression chamber bypass leak tests are conducted:

1. At least once per 18 months at an initial differential pressure of 1.5 psi, and

2. At the first refueling outage and then on the schedule required for Type A Overall Integrated Containment Leakage Rate tests by Specification 4.6.1.2.a; at an initial differential pressure of 5 psi,

except that, if the first two 5 psi leak tests performed up to that time result in:

1. A calculated $A/\sqrt{k}$ within the specified limit, and

2. The $A/\sqrt{k}$ calculated from the leak tests at 1.5 psi is $\leq$ 20% of the specified limit,

then the leak tests at 5 psi may be discontinued.

Add "INSERT C2"

*Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

ATTACHMENT B
PROPOSED AMENDMENTS TO THE
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INSERT C2

d. By conducting drywell-to-suppression chamber bypass leak tests at least once per 18 months at an initial differential pressure of 1.5 psi and verifying that the $A/\sqrt{k}$ calculated from the measured leakage is within the specified limit.

If any 1.5 psi leak test results in a calculated $A/\sqrt{k} > 20\%$ of the specified limit, then the test schedule for subsequent tests shall be reviewed by the Commission.

If two consecutive 1.5 psi leak tests result in a calculated $A/\sqrt{k}$ greater than the specified limit, then:

1. A 1.5 psi leak test shall be performed at least once per 9 months until two consecutive 1.5 psi leak tests result in the calculated $A/\sqrt{k}$ within the specified limits, and

2. A 5 psi leak test, performed with the second consecutive successful 1.5 psi leak test, results in a calculated $A/\sqrt{k}$ within the specified limit, after which the above schedule of once per 18 months for only 1.5 psi leak tests may be resumed.

If any required 5 psi leak test results in a calculated $A/\sqrt{k}$ greater than the specified limit, then the test schedule for subsequent tests shall be reviewed by the Commission.

If two consecutive 5 psi leak tests result in a calculated $A/\sqrt{k}$ greater than the specified limit, then a 5 psi leak test shall be performed at least once per 9 months until two consecutive 5 psi leak tests result in a calculated $A/\sqrt{k}$ within the specified limit, after which the above schedule of once per 18 months for only 1.5 psi leak tests may be resumed.

SURVEILLANCE REQUIREMENTS (Continued)

If any 1.5 psi or 5 psi leak test results in:

1. A calculated $A/\sqrt{k}$ greater than the specified limit, or
2. A calculated $A/\sqrt{k}$ from a 1.5 psi leak test > 20% of the specified limit,

then the test schedule for subsequent tests shall be reviewed by the Commission.

If two consecutive 1.5 psi leak tests result in a calculated $A/\sqrt{k}$ greater than the specified limit, then:

1. A 1.5 psi leak test shall be performed at least once per 9 months until two consecutive 1.5 psi leak tests result in the calculated $A/\sqrt{k}$ within the specified limits, and
2. A 5 psi leak test, performed with the second consecutive successful 1.5 psi leak test, results in a calculated $A/\sqrt{k}$ within the specified limit, after which the above schedule for only 1.5 psi leak tests may be resumed.

If two consecutive 5 psi leak tests result in a calculated $A/\sqrt{k}$ greater than the specified limit, then a 5 psi leak test shall be performed at least once per 9 months until two consecutive 5 psi leak tests result in a calculated $A/\sqrt{k}$ within the specified limit, after which the above schedule for only 1.5 psi leak tests may be resumed.

CONTAINMENT SYSTEMS

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

Replace with "INSERT D"

LIMITING CONDITION FOR OPERATION

3.6.3 The primary containment isolation valves and the reactor instrumentation line excess flow check valves shown in Table 3.6.3-1 shall be OPERABLE with isolation times less than or equal to those shown in Table 3.6.3-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more of the primary containment isolation valves shown in Table 3.6.3-1 inoperable:

1. Maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either;

a) Restore the inoperable valve(s) to OPERABLE status, or

b) Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position,* or

c) Isolate each affected penetration by use of at least one closed manual valve or blind flange.*

2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one or more of the reactor instrumentation line excess flow check valves shown in Table 3.6.3-1 inoperable:

1. Operation may continue and the provisions of Specification 3.0.3 are not applicable provided that within 4 hours either:

a) The inoperable valve is returned to OPERABLE status, or

b) The instrument line is isolated and the associated instrument is declared inoperable.

2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Add "INSERT E"

*Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control.

ATTACHMENT B
PROPOSED AMENDMENTS TO THE
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INSERT D

3.6.3 Each primary containment isolation valve and reactor instrumentation line excess flow check valve shall be OPERABLE**.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more of the primary containment isolation valves, except the reactor instrumentation line excess flow check valves, inoperable:

INSERT E

** Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifing the specified isolation time.

4.6.3.2 Each primary containment automatic isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

isolation

4.6.3.3 The isolation time of each primary containment power operated or automatic valve shown in Table 3.6.3-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 Each reactor instrumentation line excess flow check valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow.

4.6.3.5 Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing the explosive squib from at least one explosive valve such that the explosive squib in each explosive valve will be tested at least once per 90 months, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No explosive squib shall remain in use beyond the expiration of its shelf-life and operating-life.

Add "INSERT F"

Add "INSERT G"

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page B-7

INSERT F

4.6.3.6 At least once per 18 months:

a. Verify leakage rate through all four main steam lines through the isolation valves is $\leq$ 100 scfh when tested at $\geq$ 25.0 psig.*

b. Verify combined leakage rate of $\leq$ 1 gpm times the total number of primary containment isolation valves through hydrostatically tested lines that penetrate the primary containment is not exceeded when these isolation valves are tested at 1.1 P_a, $\geq$ 43.6 psig.*

INSERT G (Footnote)

* Results shall be excluded from the combined leakage for all penetrations and seals subject to Type B and C tests.

TABLE 3.6.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP[(a)]	MAXIMUM ISOLATION TIME (Seconds)
a. **Automatic Isolation Valves**		
1. Main Steam Isolation Valves	1	5*
1B21-F022A, B, C, D[(b)]		
1B21-F028A, B, C, D[(b)]		
2. Main Steam Line Drain Valves	1	
1B21-F016		≤ 15
1B21-F019		≤ 15
1B21-F067A, B, C, D[(b)]		≤ 23
3. Reactor Coolant System Sample Line Valves[(c)]	3	≤ 5
1B33-F019		
1B33-F020		
4. Drywell Equipment Drain Valves	2	
1RE024		≤ 20
1RE025		≤ 20
1RE026		≤ 15
1RE029		≤ 15
5. Drywell Floor Drain Valves	2	≤ 20
1RF012		
1RF013		
6. Reactor Water Cleanup Suction Valves	5	≤ 30
1G33-F001[(d)]		
1G33-F004		
7. RCIC Steam Line Valves	8	
1E51-F008[(e)]		≤ 20
1E51-F063		≤ 15
1E51-F064[(f)]		≤ 15
1E51-F076		≤ 15

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP[(a)]	MAXIMUM ISOLATION TIME (Seconds)
Automatic Isolation Valves (Continued)		
8. Containment Vent and Purge Valves	4	
1VQ026		≤ 10
1VQ027		≤ 10
1VQ029		≤ 10
1VQ030		≤ 10
1VQ031		≤ 10
1VQ032		≤ 5
1VQ034		≤ 10
1VQ035		≤ 5
1VQ036		≤ 10
1VQ040		≤ 10
1VQ042		≤ 10
1VQ043		≤ 10
1VQ047		≤ 5
1VQ048		≤ 5
1VQ050		≤ 5
1VQ051		≤ 5
1VQ068		≤ 5
9. RCIC Turbine Exhaust Vacuum Breaker Line Valves	9	N.A.
1E51-F080		
1E51-F086		
10. LPCS, HPCS, RCIC, RHR Injection Testable Check Bypass Valves	N.A.	N.A.

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)
Automatic Isolation Valves (Continued)		
11. Containment Monitoring Valves	2	≤ 5
1CM017A,B		
1CM018A,B		
1CM019A,B		
1CM020A,B		
1CM021B(h)		
1CM022A(h)		
1CM025A(h)		
1CM026B(h)		
1CM027		
1CM028		
1CM029		
1CM030		
1CM031		
1CM032		
1CM033		
1CM034		
12. Drywell Pneumatic Valves		
1IN001A and B	10	≤ 30
1IN017	10	≤ 22
1IN074	10	≤ 22
1IN075	10	≤ 22
1IN031	2	≤ 5
13. RHR Shutdown Cooling Mode Valves	6	
1E12-F008		≤ 40
1E12-F009		≤ 40
1E12-F023		≤ 90
1E12-F053 A and B		≤ 29
1E12-F099A and B(g)(i)		≤ 30

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TABLE 3.6.3-1 (Continued)
PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)
Automatic Isolation Valves (Continued)		
14. Tip Guide Tube Ball Valves (Five Valves)	7	N.A.
1C51-J004		
15. Reactor Building Closed Cooling Water System Valves	2	≤ 30
1WR029		
1WR040		
1WR179		
1WR180		
16. Primary Containment Chilled Water Inlet Valves	2	
1VP113 A and B		≤ 90
1VP063 A and B		≤ 40
17. Primary Containment Chilled Water Outlet Valves	2	
1VP053 A and B		≤ 40
1VP114 A and B		≤ 90
18. Recirc. Hydraulic Flow Control Line Valves(g)	2	≤ 5
1B33-F338 A and B		
1B33-F339 A and B		
1B33-F340 A and B		
1B33-F341 A and B		
1B33-F342 A and B		
1B33-F343 A and B		
1B33-F344 A and B		
1B33-F345 A and B		
19. Feedwater Testable Check Valves	2	N.A.
1B21-F032 A and B		

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP[(a)]	MAXIMUM ISOLATION TIME (Seconds)
b. Manual Isolation Valves		
1. 1FC086		N.A.
2. 1FC113		N.A.
3. 1FC114		N.A.
4. 1FC115		N.A.
5. 1MC027[(1)]		N.A.
6. 1MC033[(1)]		N.A.
7. 1SA042[(1)]		N.A.
8. 1SA046[(1)]		N.A.

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

c. Excess Flow Check Valves[(g)]

1. 1B21-F374
2. 1B21-F376
3. 1B21-F359
4. 1B21-F355
5. 1B21-F361
6. 1B21-F378
7. 1B21-F372
8. 1B21-F370
9. 1B21-F363
10. 1B21-F353
11. 1B21-F415A, B
12. 1B21-F357
13. 1B21-F382
14. 1B21-F328A, B, C, D
15. 1B21-F327A, B, C, D
16. 1B21-F413A, B
17. 1B21-F344
18. 1B21-F365
19. 1B21-F443
20. 1B21-F439
21. 1B21-F437
22. 1B21-F441

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

Excess Flow Check Valves(g) (Continued)

23. 1B21-F445A, B
24. 1B21-F453
25. 1B21-F447
26. 1B21-F455A, B
27. 1B21-F451
28. 1B21-F449
29. 1B21-F367
30. 1B21-F326A, B, C, D
31. 1B21-F325A, B, C, D
32. 1B21-F350
33. 1B21-F346
34. 1B21-F348
35. 1B21-F471
36. 1B21-F473
37. 1B21-F469
38. 1B21-F475A, B
39. 1B21-F465A, B
40. 1B21-F467
41. 1B21-F463
42. 1B21-F380
43. 1G33-F312A, B
44. 1G33-F309
45. 1E12-F315

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

Excess Flow Check Valves[g] (Continued)

46. 1E12-F359A, B
47. 1E12-F319
48. 1E12-F317
49. 1E12-F360A, B
50. 1E21-F304
51. 1E22-F304
52. 1E22-F341
53. 1E22-F342
54. 1B33-F319A, B
55. 1B33-F317A, B
56. 1B33-F313A, B, C, D
57. 1B33-F311A, B, C, D
58. 1B33-F315A, B, C, D
59. 1B33-F301A, B
60. 1B33-F307A, B, C, D
61. 1B33-F305A, B, C, D
62. 1CM004
63. 1CM002
64. 1CM012
65. 1CM010
66. 1VQ061
67. 1B21-F457
68. 1B21-F459

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

Excess Flow Check Valves[(g)] (Continued)

69. 1B21-F461
70. 1CM102
71. 1B21-F570
72. 1B21-F571

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

d. Other Isolation Valves

1. MSIV Leakage Control System

   1E32-F001A, E, J, N[b]

2. Reactor Feedwater and RWCU System Return

   1B21-F010A, B
   1B21-F065A, B
   1G33-F040

3. Residual Heat Removal/Low Pressure Coolant Injection System

   1E12-F042A, B, C
   1E12-F016A, B
   1E12-F017A, B
   1E12-F004A, B, C[j]
   1E12-F027A, B[j]
   1E12-F024A, B[j]
   1E12-F021[j]
   1E12-F302[j]
   1E12-F064A, B, C[j]
   1E12-F011A, B[j]
   1E12-F088A, B, C[j]
   1E12-F025A, B, C[j]
   1E12-F030[j]
   1E12-F005[j]
   1E12-F073A, B[j]
   1E12-F074A, B[j]
   1E12-F055A, B[j]
   1E12-F036A, B[j]
   1E12-F311A, B[j]
   1E12-F041A, B, C[k]
   1E12-F050A, B[k]

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

Other Isolation Valves (Continued)

4. Low Pressure Core Spray System

1E21-F005
1E21-F001(j)
1E21-F012(j)
1E21-F011(j)
1E21-F018(j)
1E21-F031(j)
1E21-F006(k)

5. High Pressure Core Spray System

1E22-F004
1E22-F015(j)
1E22-F023(j)
1E22-F012(j)
1E22-F014(j)
1E22-F005(k)

6. Reactor Core Isolation Cooling System

1E51-F013
1E51-F069
1E51-F028
1E51-F068
1E51-F040
1E51-F031(j)
1E51-F019(j)
1E51-F065(k)
1E51-F066(k)
1E51-F059(m)
1E51-F022(m)
1E51-F362(n)
1E51-F363(n)

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TABL. .6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

Other Isolation Valves (Continued)

7. Post LOCA Hydrogen Control

1HG001A, B
1HG002A, B
1HG005A, B
1HG006A, B

8. Standby Liquid Control System

1C41-F004A, B
1C41-F007

9. Reactor Recirculation Seal Injection

1B33-F013A, B[j]
1B33-F017A, B[j]

10. Drywell Pneumatic System

1IN018

11. Reference Leg Backfill

1C11-F422B
1C11-F422D
1C11-F422F
1C11-F422G
1C11-F423B
1C11-F423D
1C11-F423F
1C11-F423G

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

* But ≥ 3 seconds.

(a) See Specification 3.3.2, Table 3.3.2-1, for isolation signal(s) that operates each valve group.

(b) Not included in total sum of Type B and C tests.

(c) May be opened on an intermittent basis under administrative control.

(d) Not closed by SLCS actuation.

(e) Not closed by Trip Functions 5a, b or c, Specification 3.3.2, Table 3.3.2-1.

(f) Not closed by Trip Functions 4a, c, d, e or f of Specification 3.3.2, Table 3.3.2-1.

(g) Not subject to Type C leakage test.

(h) Opens on an isolation signal. Valves will be open during Type A test. No Type C test required.

(i) Also closed by drywell pressure-high signal.

(j) Hydraulic leak test at 43.6 psig.

(k) Not subject to Type C leakage test - leakage rate tested per Specification 4.4.3.2.2.

(l) These penetrations are provided with removable spools outboard of the outboard isolation valve. During operation, these lines will be blind flanged using a double O-ring and a type B leak test. In addition, the packing of these isolation valves will be soap-bubble tested to ensure insignificant or no leakage at the containment test pressure each refueling outage.

(m) If valves 1E51-F362 and 1E51-F363 are locked closed and acceptably leak rate tested, then valves 1E51-F059 and 1E51-F022 are not considered to be primary containment isolation valves and are not required to be leak rate tested.

(n) Either the 1E51-F362 or the 1E51-F363 valve may be open when the RCIC system is in the standby mode of operation, and both valves may be open during operation of the RCIC system in the full flow test mode, providing that:

1) valve 1E51-F022 is acceptably leak rate tested, and

2) valve 1E51-F059 is deactivated, locked closed and acceptably leak rate tested, and

3) the spectacle flange, installed immediately downstream of the 1E51-F059 valve, is closed and acceptably leak rate tested.

Information Only
No Changes

3/4.6.4 VACUUM RELIEF

LIMITING CONDITION FOR OPERATION

3.6.4 All suppression chamber - drywell vacuum breakers shall be OPERABLE and closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one suppression chamber - drywell vacuum breaker inoperable and/or open, within 4 hours close the manual isolation valves on both sides of the inoperable and/or open vacuum breaker. Restore the inoperable and/or open vacuum breaker to OPERABLE and closed status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one position indicator of any OPERABLE suppression chamber - drywell vacuum breaker inoperable, restore the inoperable position indicator to OPERABLE status within 14 days or visually verify the vacuum breaker to be closed at least once per 24 hours. Otherwise, declare the vacuum breaker inoperable.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each suppression chamber - drywell vacuum breaker shall be:

a. Verified closed at least once per 7 days.

b. Demonstrated OPERABLE:

1. At least once per 31 days and within 12 hours after any discharge of steam to the suppression chamber from the safety-relief valves, by cycling each vacuum breaker through at least one complete cycle of full travel.

2. At least once per 31 days by verifying both position indicators OPERABLE by performance of a CHANNEL FUNCTIONAL TEST.

3. At least once per 18 months by;

a) Verifying the force required to open the vacuum breaker, from the closed position, to be less than or equal to 0.5 psid, and

b) Verifying both position indicators OPERABLE by performance of a CHANNEL CALIBRATION.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.4.2 The manual isolation valves on both sides of an inoperable and/or open suppression chamber-drywell vacuum breaker shall be verified to be closed at least once per 7 days.

4.6.4.3 Vacuum breaker header flanges which have been broken shall be leak tested after re-making by Type B test at 39.6 psig per Specification 4.6.1.2.d.

6

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

DRYWELL AND SUPPRESSION CHAMBER HYDROGEN RECOMBINER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent drywell and suppression chamber hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one drywell and/or suppression chamber hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each drywell and suppression chamber hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 92 days by cycling each flow control valve and recirculation valve through at least one complete cycle of full travel.

b. At least once per ~~6~~ 18 months by verifying, during a recombiner system functional test:

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

   2. That the reaction chamber gas temperature increases to 1200 ± 25°F within 2 hours.

c. At least once per 18 months by:

   1. Performing a CHANNEL CALIBRATION of all recombiner operating instrumentation and control circuits.

   2. Verifying the integrity of all heater electrical circuits by performing a resistance to ground test within 30 minutes following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 100,000 ohms.

d. By measuring the leakage rate:

   1. As a part of the overall integrated leakage rate test required by Specification 3.6.1.2, or

   2. By measuring the leakage rate of the system outside of the containment isolation valves at P_a, 39.6 psig, on the schedule required by Specification 4.6.1.2 and including the measured leakage as a part of the leakage determined in accordance with Specification 4.6.1.2.

e

Add "INSERT H"

3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

The structural integrity of the primary containment is ensured by the successful completion of the Inservice Inspection Program for Post Tensioning Tendons and by associated visual inspections of the steel liner and penetrations for evidence of deterioration or breach of integrity. This ensures that the structural integrity of the primary containment will be maintained in accordance with the provisions of the Primary Containment Tendon Surveillance Program. Testing and Frequency are consistent with the recommendations of Regulatory Guide 1.35, Revision 3, except that the Unit 1 and 2 primary containments shall be treated as twin containments even though the Initial Structural Integrity Tests were not within 2 years of each other.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 39.6 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J to 10 CFR 50 with the exception of exemption(s) granted for main steam isolation valve leak testing and testing the airlocks after each opening.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

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The limitation on closure and leak rate for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

e e

3/4.6.1.1

3/4.6.1.2 DELETED

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INSERT H

PRIMARY CONTAINMENT INTEGRITY is maintained by limiting overall integrated leakage to $\leq 1.0\ L_a$ and the Type B and C combined leakage rate acceptance criterion is $\leq 0.60\ L_a$. Prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test, the combined Type B and C leakage must be $< 0.60\ L_a$, and the overall Type A leakage must be $< 0.75\ L_a$ when a Type A test is scheduled. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

The maximum allowable leakage rate for the primary containment (L_a) is 0.635% by weight of the containment atmosphere per day at the calculated maximum peak containment pressure (P_a) of 39.6 psig.

Individual leakage rates specified for the primary containment air lock, main steam lines through the isolation valves, and valves in hydrostatically tested lines are addressed in LCO 3.6.1.3, and Surveillance Requirement 4.6.3.6.

Surveillance Requirement 4.6.1.1.b maintains PRIMARY CONTAINMENT INTEGRITY by requiring compliance with the visual examinations and leakage rate test requirements of 10 CFR 50, Appendix J, as modified by approved exemptions. Failure to meet air lock leakage testing (4.6.1.3) or main steam isolation valve leakage (4.6.3.6.a) does not necessarily result in a failure of this Surveillance Requirement, 4.6.1.1.b. The impact of the failure to meet these Surveillance Requirements 4.6.1.3 and 4.6.1.1.b must be evaluated against the Type A, B, and C acceptance criteria of 10 CFR 50, Appendix J, as modified by approved exemptions. The leakage limits for main steam lines through the isolation valves and leakage test results of Surveillance Requirement 4.6.3.6.a are not included in the total sum of Type B and C tests (approved exemption). As-left leakage prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test is required to be $< 0.60\ L_a$ for combined Type B and C leakage, and $< 0.75\ L_a$ for overall Type A leakage. At all other times between required Type A tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. The combined Type B and C leakage remains as $\leq 0.60\ L_a$ between scheduled tests, in accordance with Appendix J.

The Frequency is required by 10 CFR 50, Appendix J, as modified by approved exemptions. Thus, 4.0.2 (which allows Frequency extensions) does not apply to Surveillance Requirement 4.6.1.1.b.

DEPRESSURIZATION SYSTEMS (Continued)

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety-relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety-relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety relief valve to assure mixing and uniformity of energy insertion to the pool.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

The OPERABILITY of the primary containment isolation valves 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. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

Add "INSERT J"

3/4.6.4 VACUUM RELIEF

Vacuum relief breakers are provided to equalize the pressure between the suppression chamber and drywell. This system will maintain the structural integrity of the primary containment under conditions of large differential pressures.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There are four valves to provide redundancy so that operation may continue for up to 72 hours with one vacuum breaker inoperable provided that the manual isolation valves on each side are in the closed position.

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INSERT J

Primary Containment Isolation Valves (PCIVs) form a part of the primary containment boundary. The PCIV safety function is related to control of primary containment leakage rates during accidents or other conditions to limit the untreated release of radioactive materials from the containment in excess of the design limits.

The automatic isolation valves are required to have isolation times within limits and actuate on an automatic isolation signal. The valves covered by this specification are listed with their associated stroke times, and other design information for lines penetrating the Primary Containment, in UFSAR Section 6.2.

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact.

Main steam lines through the isolation valves and hydrostatically tested valves must meet alternative leakage rate requirements. Other PCIV leakage rates are addressed by specification 3/4.6.1.1, "PRIMARY CONTAINMENT INTEGRITY". UFSAR Section 6.2 also describes special leakage test requirements and exemptions.

This specification provides assurance that the PCIVs will perform their designed safety functions to control leakage from the primary containment during accidents.

The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the primary containment.

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INSERT J (Continued)

Surveillance Requirement 4.6.3.6.a verifies leakage through all four main steam lines is $\leq$ 100 scfh when tested at $\geq P_t$ (25.0 psig). The transient and accident analyses are based on leakage at the specified leakage rate. The leakage rate for main steam lines through the isolation valves must be verified to be in accordance with the leakage test requirements of 10 CFR 50, Appendix J, as modified by approved exemptions. A Note has been added to this Surveillance Requirement requiring the results to be excluded from the total of Type B and Type C tests. This ensures that leakage rate for main steam lines through the isolation valves is properly accounted for in accordance with an approved exemption. The frequency is "at least once per 18 months" in accordance with an approved exemption.

Surveillance Requirement 4.6.3.6.b test of hydrostatically tested lines provides assurance that the assumptions of UFSAR Section 6.2 are met. The combined leakage rates must be demonstrated in accordance with the leakage rate test at a frequency of "at least once per 18 months". A Note has been added to this Surveillance Requirement requiring the results to be excluded the total of Type B and Type C tests. This is in accordance with 10 CFR 50, Appendix J, and approved exemptions.

(d) The maximum average planar linear heat generation (MAPLHGR) limit will be reduced by 0.85.

(e) Technical Specification Setpoints shall read as follows:

T.S.2.2.1 S 0.66W + 45.7 (Trip Setpoint)
S 0.66W + 48.7 (Allowable)

T.S.3.2.2 S (0.66W + 45.7) T*
S_{RB} (0.66W + 36.7) T*
T* as defined in T.S.3.2.2

T.S.3.3.6 APRM Upscale 0.66W + 36.7 (Trip Setpoint)
APRM Upscale 0.66W + 39.7 (Allowable)
RBM Upscale 0.66W + 34.7 (Trip Setpoint)
RBM Upscale 0.66W + 37.7 (Allowable)

(f) The average power range monitor (APRM) flux noise will be measured once per shift; and the recirculation loop flow will be reduced if the flux noise averaged over 1/2 hour exceeds 5 percent peak to peak, as measured by the APRM chart recorder.

Add "INSERT K" Attached

(g) The core plate delta P noise will be measured once per shift, and the recirculation loop flow will be reduced if the noise exceeds one (1) psi peak-to-peak.

Am. 12 12/20/82

D. Exemptions from certain requirements (b) of Appendices G, H and J and 10 CFR Part 73 are described in the Safety Evaluation Report and Supplement No. 1, No. 2 and No. 3 to the Safety Evaluation Report. ~~In addition,~~ An exemption was requested until the completion of the first refueling from the requirements of 10 CFR 70.24 (c) and an exemption from 10 CFR Part 50, Appendix E from performing a full scale exercise within one year before issuance of an operating license, both exemptions (b) and (c) are described in Supplement No. 2 of the Safety Evaluation Report. (d) ~~Finally,~~ An exemption was requested from the requirements of 10 CFR 50.44 until either the required 100 percent rated thermal power trip startup test has been completed or the reactor has operated for 120 effective full power days as specified by the Technical Specifications. d ~~This latter~~ e Exemption is described in the safety evaluation of License Amendment No. 12. These 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, these exemptions are hereby granted. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, and the rules and regulations of the Commission (except as hereinafter exempted therefrom), and the provisions of the Act.

0011k

The facility requires exemptions from certain requirements of 10 CFR Part 50, 10 CFR Part 70, and 10 CFR Part 73. These include: (a)

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INSERT K (Unit 1, NPF-11)

(e) An exemption from the requirement of paragraph III.D of Appendix J to conduct the third Type A test of each ten-year service period when the plant is shutdown for the 10-year plant inservice inspections. Exemption (e) is described in the safety evaluation accompanying Amendment No. ____ to this license.

DEFINITIONS

SECTION

1.0 DEFINITIONS

DEFINITIONS

SECTION

DEFINITIONS (Continued) PAGE

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIST OF TABLES (Continued)

e

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME (Continued)

FRACTION OF LIMITING POWER DENSITY

1.14 The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.

FRACTION OF RATED THERMAL POWER

1.15 The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

FREQUENCY NOTATION

1.16 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

1.17 A GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.18 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

1.19 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

LIMITING CONTROL ROD PATTERN

1.2~~0~~ 1 A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

Add "INSERT A"

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INSERT A

$\underline{L_a}$

1.20 The maximum allowable primary containment leakage rate, L_a, shall be 0.635 % of primary containment air weight per day at the calculated peak containment pressure (P_a = 39.6 psig).

LINEAR HEAT GENERATION RATE

1.2~~1~~2 LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST

1.2~~2~~3 A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components, i.e., all relays and contacts, all trip units, solid state logic elements, etc. of a logic circuit, from sensor through and including the actuated device to verify OPERABILITY. THE LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps such that the entire logic system is tested.

MAXIMUM FRACTION OF LIMITING POWER DENSITY

1.2~~3~~4 The MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be the highest value of the FLPD which exists in the core.

MEMBER(S) OF THE PUBLIC

1.2~~4~~5 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

MINIMUM CRITICAL POWER RATIO

1.2~~5~~6 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

OFFSITE DOSE CALCULATION MANUAL

1.2~~6~~7 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Technical Specification Section 6.2.F.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-Annual Radioactive Effluent Release Reports required by Technical Specification Sections 6.6.A.3 and 6.6.A.4.

OPERABLE - OPERABILITY

1.2~~7~~8 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, a normal and an emergency electrical power source, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION - CONDITION

1.2~~8~~ 9 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

1.~~29~~ 30 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.3~~0~~ 1 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

1.3~~1~~ 2 PRIMARY CONTAINMENT INTEGRITY shall exist when:

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

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

2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 of Specification, 3.6.3. e

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is OPERABLE pursuant to Specification 3.6.1.3.

d. d The primary containment leakage rates are maintained within the limits of per e Specification 3.6.1.2. Surveillance Requirement 4.6.1.1.b.

e. The suppression chamber is OPERABLE pursuant to Specification 3.6.2.1.

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

g. Primary containment structural integrity has been verified in accordance with Surveillance Requirement 4.6.1.1.e.

for valves that are open under administrative control as permitted by

PROCESS CONTROL PROGRAM

1.3~~7~~3 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.3~~8~~4 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.3~~4~~ 5 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3323 MWT.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.3~~5~~ 6 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE EVENT

1.3~~6~~ 7 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.3~~7~~ 8 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

1.3~~8~~ 9 SECONDARY CONTAINMENT INTEGRITY shall exist when:

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

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

   2. Closed by at least one manual valve, blind flange, or deactivated automatic damper secured in its closed position, except as provided in Table 3.6.5.2-1 of Specification 3.6.5.2.

b. All secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is OPERABLE pursuant to Specification 3.6.5.3.

d. At least one door in each access to the secondary containment is closed.

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

f. The pressure within the secondary containment is less than or equal to the value required by Specification 4.6.5.1.a.

SHUTDOWN MARGIN

1.~~39~~ 40 SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming all control rods are fully inserted except for the single control rod of highest reactivity worth which is assumed to be fully withdrawn and the reactor is in the shutdown condition; cold, i.e. 68°F; and xenon free.

SITE BOUNDARY

1.4~~0~~1 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SOURCE CHECK

1.4~~1~~2 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

STAGGERED TEST BASIS

1.4~~2~~3 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals.

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

THERMAL POWER

1.4~~3~~4 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TURBINE BYPASS SYSTEM RESPONSE TIME

1.4~~4~~5 The TURBINE BYPASS SYSTEM RESPONSE TIME shall be time interval from when the turbine bypass control unit generates a turbine bypass valve flow signal until the turbine bypass valves travel to their required positions. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

UNIDENTIFIED LEAKAGE

1.4~~5~~6 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

VENTILATION EXHAUST TREATMENT SYSTEM

1.4~~6~~7 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.4~~7~~8 VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
A. AUTOMATIC INITIATION				
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
(1) Low, Level 3	7	2	1, 2, 3	20
(2) Low Low, Level 2	2, 3	2	1, 2, 3	20
(3) Low Low Low, Level 1	1, 10	2	1, 2, 3	20
b. Drywell Pressure - High	2, 7, 10	2	1, 2, 3	20
c. Main Steam Line				
1) Radiation - High	1 3	2 2	1, 2, 3 1, 2, 3	21 22
2) Pressure - Low	1	2	1	23
3) Flow - High	1	2/line$^{(d)}$	1, 2, 3	21
d. Main Steam Line Tunnel Temperature - High	1	2	$1^{(i)(j)}$, $2^{(i)(j)}$, $3^{(i)(j)}$	21
e. Main Steam Line Tunnel ΔTemperature - High	1	2	$1^{(i)(j)}$, $2^{(i)(j)}$, $3^{(i)(j)}$	21
f. Condenser Vacuum - Low	1	2	1, 2*, 3*	21
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	$4^{(c)(e)}$	2	1, 2, 3 and **	24
b. Drywell Pressure - High	$4^{(c)(e)}$	2	1, 2, 3	24
c. Reactor Vessel Water Level - Low Low, Level 2	$4^{(c)(e)}$	2	1, 2, 3, and #	24
d. Fuel Pool Vent Exhaust Radiation - High	$4^{(c)(e)}$	2	1, 2, 3, and **	24

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	5	1	1, 2, 3	22
b. Heat Exchanger Area Temperature - High	5	1/heat exchanger	1, 2, 3	22
c. Heat Exchanger Area Ventilation ΔT - High	5	1/heat exchanger	1, 2, 3	22
d. SLCS Initiation	$5^{(f)}$	NA	1, 2, 3	22
e. Reactor Vessel Water Level - Low Low, Level 2	5	2	1, 2, 3	22
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	8	1	1, 2, 3	22
b. RCIC Steam Supply Pressure - Low	8, $9^{(g)}$	2	1, 2, 3	22
c. RCIC Turbine Exhaust Diaphragm Pressure - High	8	2	1, 2, 3	22
d. RCIC Equipment Room Temperature - High	8	1	1, 2, 3	22
e. RCIC Steam Line Tunnel Temperature - High	8	1	1, 2, 3	22
f. RCIC Steam Line Tunnel Δ Temperature - High	8	1	1, 2, 3	22
g. Drywell Pressure - High	$9^{(g)}$	2	1, 2, 3	22
h. RCIC Equipment Room Δ Temperature - High	8	1	1, 2, 3	22

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION				
a. RHR Equipment Area Δ Temperature - High	8	1/RHR area	1, 2, 3	22
b. RHR Area Temperature - High	8	1/RHR area	1, 2, 3	22
c. RHR Heat Exchanger Steam Supply Flow - High	8	1	1, 2, 3	22
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	6	2	1, 2, 3	25
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	6	1	1, 2, 3	25
c. RHR Pump Suction Flow - High	6	1	1, 2, 3	25
d. RHR Area Temperature - High	6	1/RHR area	1, 2, 3	25
e. RHR Equipment Area ΔT - High	6	1/RHR area	1, 2, 3	25
B. MANUAL INITIATION				
1. Inboard Valves	1, 2, 5, 6, 7	1/group	1, 2, 3	26
2. Outboard Valves	1, 2, 5, 6, 7	1/group	1, 2, 3	26
3. Inboard Valves	4 (c) (e)	1/group	1, 2, 3 and **,#	26
4. Outboard Valves	4(c) (e)	1/group	1, 2, 3 and **,#	26
5. Inboard Valves	3, 8, 9	1/valve	1, 2, 3	26
6. Outboard Valves	3, 8, 9	1/valve	1, 2, 3	26
7. Outboard Valve	8(h)	1/group	1, 2, 3	26

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION STATEMENTS

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN with the next 24 hours.

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 22 - Close the affected system isolation valves within 1 hour and declare the affected system inoperable.

ACTION 23 - Be in at least STARTUP within 6 hours.

ACTION 24 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

ACTION 25 - Lock the affected system isolation valves closed within 1 hour and declare the affected system inoperable.

ACTION 26 - Provided that the manual initiation function is OPERABLE for each other group valve, inboard or outboard, as applicable, in each line, restore the manual initiation function to OPERABLE status within 24 hours; otherwise, restore the manual initiation function to OPERABLE status within 8 hours; otherwise:

a. Be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, or

b. Close the affected system isolation valves within the next hour and declare the affected system in operable.

Deleted

TABLE NOTATIONS

* May be bypassed with reactor steam pressure < 1043 psig and all turbine stop valves closed.

** When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

\# During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

(a) See Specification 3.6.3, Table 3.6.3-1 for valves in each valve group. e

(b) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the channel in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter. In addition for those trip systems with a design providing only one channel per trip system, the channel may be placed in an inoperable status for up to 8 hours for required surveillance testing without placing the channel in the tripped condition provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is operable and all required actuation instrumentation for that redundant valve is OPERABLE, ~~or~~ place the trip system in the tripped condition.

(c) Also actuates the standby gas treatment system.

(d) A channel is OPERABLE if 2 of 4 instruments in that channel are OPERABLE.

(e) Also actuates secondary containment ventilation isolation dampers per Table 3.6.5.2-1.

(f) Closes only RWCU system inlet outboard valve.

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION	N/A
a. Reactor Vessel Water Level - Low, Level 3	
b. Reactor Vessel (RHR Cut-In Permissive) Pressure - High	
c. RHR Pump Suction Flow - High	
d. RHR Area Cooler Temperature High	
e. RHR Equipment Area ΔT High	
B. MANUAL INITIATION	N/A
1. Inboard Valves	
2. Outboard Valves	
3. Inboard Valves	
4. Outboard Valves	
5. Inboard Valves	
6. Outboard Valves	
7. Outboard Valve	

TABLE NOTATIONS

* Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed.

** Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

\# Isolation system instrumentation response time specified for the Trip Function actuating each valve group shall be added to isolation time shown in Table 3.6.3-1 and 3.6.5.2-1 for valves in each valve group to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

the MSIVs

MSIV

N/A Not Applicable.

for valves that are open under administrative control as permitted

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2,* and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated: a

a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seal with gas at Pa, 39.6 psig, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 La.

~~b.~~ a At least once per 31 days by verifying that all primary containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Table 3.6.3-1 of Specification 3.6.3. e

Add New b. from "INSERT B"

c. By verifying each primary containment air lock OPERABLE per Specification 3.6.1.3.

d. By verifying the suppression chamber OPERABLE per Specification 3.6.2.1.

e. Verify primary containment structural integrity in accordance with the Inservice Inspection Program for Post Tensioning Tendons. The frequency shall be in accordance with the Inservice Inspection Program for Post Tensioning Tendons.

Add "INSERT C1"

*See Special Test Exception 3.10.1

**Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been deinerted since the last verification or more often than once per 92 days.

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INSERT B

4.6.1.1.b. Perform required visual examinations and leakage rate testing except for primary containment air lock testing and main steam lines through the isolation valves, in accordance with and at the frequency# specified by 10 CFR 50, Appendix J, as modified by approved exemptions.

The overall integrated leakage rate acceptance criterion is $\leq 1.0\ L_a$. The Type B and C combined leakage rate acceptance criterion is $\leq 0.60\ L_a$. 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 $< 0.60\ L_a$ for the combined Type B and Type C tests, and $< 0.75\ L_a$ for the Type A test.

INSERT C1

\# The provisions of Specification 4.0.2 are not applicable to the frequencies specified by 10 CFR 50, Appendix J.

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PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, 0.635 percent by weight of the containment air per 24 hours at P_a, 39.6 psig.

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and all valves listed in Table 3.6.3-1, except for main steam isolation valves and valves which are hydrostatically leak tested per Table 3.6.3-1, subject to Type B and C tests when pressurized to P_a, 39.6 psig.

c. *Less than or equal to 100 scf per hour for all four main steam lines through the isolation valves when tested at 25.0 psig.

d. A combined leakage rate of less than or equal to 1 gpm times the total number of ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 P_a, 43.6 psig.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:

With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 L_a, or

b. The measured combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam isolation valves and valves which are hydrostatically leak tested per Table 3.6.3-1, subject to Type B and C tests exceeding 0.60 L_a, or

c. The measured leakage rate exceeding 100 scf per hour for all four main steam lines through the isolation valves, or

d. The measured combined leakage rate for all ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 1 gpm times the total number of such valves,

*Exemption to Appendix J of 10 CFR Part 50.

 (Next page is 3/4 6-5)

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CONTAINMENT SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

restore:

a. The overall integrated leakage rate(s) to less than or equal to 0.75 L_a, and

b. The combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam isolation valves and valves which are hydrostatically leak tested per Table 3.6.3-1, subject to Type B and C tests to less than or equal to 0.60 L_a, and

c. The leakage rate to less than or equal to 100 scf per hour for all four main steam lines through the isolation valves, and

d. The combined leakage rate for all ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 1 gpm times the total number of such valves,

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972:

a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10 month intervals during shutdown at P_a, 39.6 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

b. If any periodic Type A test fails to meet 0.75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a, at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 L_a.

2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25% of the total measured leakage at P_a, 39.6 psig.

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CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

d. Type B and C tests shall be conducted with gas at P_a, 39.6 psig*, at intervals no greater than 24 months except for tests involving:

   1. Air locks,

   2. Main steam line isolation valves,

   3. Valves pressurized with fluid from a seal system, and

   4. ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. Main steam line isolation valves shall be leak tested at least once per 18 months.

g. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a, 43.6 psig, and the seal system capacity is adequate to maintain system pressure for at least 30 days.

h. ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at least once per 18 months.

i. The provisions of Specification 4.0.2 are not applicable to 24 month or 40 ± 10 month surveillance intervals.

*Unless a hydraulic test is required per Table 3.6.3-1.

SURVEILLANCE REQUIREMENTS (Continued)

c. By verifying at least 2 suppression chamber water level instrumentation channels and at least 14 suppression pool water temperature instrumentation channels, 7 in each of two divisions, OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,
2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
3. CHANNEL CALIBRATION at least once per 18 months.

The suppression chamber water level and suppression pool temperature alarm setpoint shall be:

a) High water level $\leq$ +2 inches*

b) Low water level $\geq$ -3 inches*

c) High temperature $\leq$ 105°F

d. By conducting drywell-to-suppression chamber bypass leak tests and verifying that the $A/\sqrt{k}$ calculated from the measured leakage is within the specified limit when drywell-to-suppression chamber bypass leak tests are conducted:

1. At least once per 18 months at an initial differential pressure of 1.5 psi, and
2. At the first refueling outage and then on the schedule required for Type A Overall Integrated Containment Leakage Rate tests by Specification 4.6.1.2.a., at an initial differential pressure of 5 psi,

except that, if the first two 5 psi leak tests performed up to that time result in:

1. A calculated $A/\sqrt{k}$ within the specified limit, and
2. The $A/\sqrt{k}$ calculated from the leak tests at 1.5 psi is $\leq$ 20% of the specified limit,

then the leak tests at 5 psi may be discontinued.

Add "INSERT C2"

*Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).

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INSERT C2

d. By conducting drywell-to-suppression chamber bypass leak tests at least once per 18 months at an initial differential pressure of 1.5 psi and verifying that the A/√k calculated from the measured leakage is within the specified limit.

If any 1.5 psi leak test results in a calculated A/√k > 20% of the specified limit, then the test schedule for subsequent tests shall be reviewed by the Commission.

If two consecutive 1.5 psi leak tests result in a calculated A/√k greater than the specified limit, then:

1. A 1.5 psi leak test shall be performed at least once per 9 months until two consecutive 1.5 psi leak tests result in the calculated A/√k within the specified limits, and

2. A 5 psi leak test, performed with the second consecutive successful 1.5 psi leak test, results in a calculated A/√k within the specified limit, after which the above schedule of once per 18 months for only 1.5 psi leak tests may be resumed.

If any required 5 psi leak test results in a calculated A/√k greater than the specified limit, then the test schedule for subsequent tests shall be reviewed by the Commission.

If two consecutive 5 psi leak tests result in a calculated A/√k greater than the specified limit, then a 5 psi leak test shall be performed at least once per 9 months until two consecutive 5 psi leak tests result in a calculated A/√k within the specified limit, after which the above schedule of once per 18 months for only 1.5 psi leak tests may be resumed.

SURVEILLANCE REQUIREMENTS (Continued)

If any 1.5 psi or 5 psi leak test results in:

1. A calculated $A/\sqrt{K}$ greater than the specified limit, or

2. A calculated $A/\sqrt{K}$ from a 1.5 psi leak test > 20% of the specified limit,

then the test schedule for subsequent tests shall be reviewed by the Commission.

If two consecutive 1.5 psi leak tests result in a calculated $A/\sqrt{K}$ greater than the specified limit, then:

1. A 1.5 psi leak test shall be performed at least once per 9 months until two consecutive 1.5 psi leak tests result in the calculated $A/\sqrt{K}$ within the specified limits, and

2. A 5 psi leak test, performed with the second consecutive successful 1.5 psi leak test, results in a calculated $A/\sqrt{K}$ within the specified limit, after which the above schedule for only 1.5 psi leak tests may be resumed.

If two consecutive 5 psi leak tests result in a calculated $A/\sqrt{K}$ greater than the specified limit, then a 5 psi leak test shall be performed at least once per 9 months until two consecutive 5 psi leak tests result in a calculated $A/\sqrt{K}$ within the specified limit, after which the above schedule for only 1.5 psi leak tests may be resumed.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

Replace with "INSERT D"

LIMITING CONDITION FOR OPERATION

3.6.3 The primary containment isolation valves and the reactor instrumentation line excess flow check valves shown in Table 3.6.3-1 shall be OPERABLE with isolation times less than or equal to those shown in Table 3.6.3-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more of the primary containment isolation valves shown in Table 3.6.3-1 inoperable:

   1. Maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either;

      a) Restore the inoperable valve(s) to OPERABLE status, or

      b) Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position,* or

      c) Isolate each affected penetration by use of at least one closed manual valve or blind flange.*

   2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one or more of the reactor instrumentation line excess flow check valves shown in Table 3.6.3-1 inoperable:

   1. Operation may continue and the provisions of Specification 3.0.3 are not applicable provided that within 4 hours either:

      a) The inoperable valve is returned to OPERABLE status, or

      b) The instrument line is isolated and the associated instrument is declared inoperable.

   2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Add "INSERT E"

*Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control.

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INSERT D

3.6.3 Each primary containment isolation valve and reactor instrumentation line excess flow check valve shall be OPERABLE**.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more of the primary containment isolation valves, except the reactor instrumentation line excess flow check valves, inoperable:

INSERT E

** Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each primary containment isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifing the specified isolation time.

4.6.3.2 Each primary containment automatic isolation valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

isolation

4.6.3.3 The isolation time of each primary containment power operated or automatic valve shown in Table 3.6.3-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 Each reactor instrumentation line excess flow check valve shown in Table 3.6.3-1 shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow.

4.6.3.5 Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing the explosive squib from at least one explosive valve such that the explosive squib in each explosive valve will be tested at least once per 90 months, and initiating the explosive squib. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No explosive squib shall remain in use beyond the expiration of its shelf-life and operating-life.

Add "INSERT F"

Add "INSERT G"

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INSERT F

4.6.3.6 At least once per 18 months:

a. Verify leakage rate through all four main steam lines through the isolation valves is $\leq$ 100 scfh when tested at $\geq$ 25.0 psig.*

b. Verify combined leakage rate of $\leq$ 1 gpm times the total number of primary containment isolation valves through hydrostatically tested lines that penetrate the primary containment is not exceeded when these isolation valves are tested at 1.1 P_a, $\geq$ 43.6 psig.*

INSERT G (Footnote)

* Results shall be excluded from the combined leakage for all penetrations and seals subject to Type B and C tests.

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TABLE 3.6.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)
a. Automatic Isolation Valves		
1. Main Steam Isolation Valves	1	5*
2B21-F022A, B, C, D(b)		
2B21-F028A, B, C, D(b)		
2. Main Steam Line Drain Valves	1	
2B21-F016		≤ 15
2B21-F019		≤ 15
2B21-F067A, B, C, D(b)		≤ 23
3. Reactor Coolant System Sample Line Valves(c)	3	
2B33-F019		≤ 5
2B33-F020		
4. Drywell Equipment Drain Valves	2	
2RE024		≤ 20
2RE025		≤ 20
2RE026		≤ 15
2RE029		≤ 15
5. Drywell Floor Drain Valves	2	≤ 20
2RF012		
2RF013		
6. Reactor Water Cleanup Suction Valves	5	≤ 30
2G33-F001(d)		
2G33-F004		
7. RCIC Steam Line Valves	8	
2E51-F008(e)		≤ 20
2E51-F063		≤ 15
2E51-F064(f)		≤ 15
2E51-F076(f)		≤ 15
2E51-F091(f)		≤ 15

(Next page is 3/4 6-38)

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

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VALVE FUNCTION AND NUMBER	VALVE GROUP[a]	MAXIMUM ISOLATION TIME (Seconds)
Automatic Isolation Valves (Continued)		
8. Containment Vent and Purge Valves	4	
2VQ026		$<$ 10
2VQ027		$\le$ 10
2VQ029		$\le$ 10
2VQ030		$\le$ 10
2VQ031		$\le$ 10
2VQ032		$\le$ 5
2VQ034		$\le$ 10
2VQ035		$\le$ 5
2VQ036		$\le$ 10
2VQ040		$\le$ 10
2VQ042		$\le$ 10
2VQ043		$\le$ 10
2VQ047		$\le$ 5
2VQ048		$\le$ 5
2VQ050		$\le$ 5
2VQ051		$\le$ 5
2VQ068		$\le$ 5
9. RCIC Turbine Exhaust Vacuum Breaker Line Valves	9	N.A.
2E51-F080		
2E51-F086		
10. LPCS, HPCS, RCIC, RHR Injection Testable Check Bypass Valves[g]	N.A.	N.A.

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP[a]	MAXIMUM ISOLATION TIME (Seconds)
Automatic Isolation Valves (Continued)		
11. Containment Monitoring Valves	2	$\leq$ 5
2CM017A,B		
2CM018A,B		
2CM019A,B		
2CM020A,B		
2CM021B[h]		
2CM022A[h]		
2CM025A[h]		
2CM026B[h]		
2CM027		
2CM028		
2CM029		
2CM030		
2CM031		
2CM032		
2CM033		
2CM034		
12. Drywell Pneumatic Valves		
2IN001A and B	10	$\leq$ 30
2IN017	10	$\leq$ 22
2IN074	10	$\leq$ 22
2IN075	10	$\leq$ 22
2IN031	2	$\leq$ 5
13. RHR Shutdown Cooling Mode Valves	6	
2E12-F008		$\leq$ 40
2E12-F009		$\leq$ 40
2E12-F023		$\leq$ 90
2E12-F053 A and B		$\leq$ 29
2E12-F099A and B[g][i]		$\leq$ 30

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP[a]	MAXIMUM ISOLATION TIME (Seconds)
Automatic Isolation Valves (Continued)		
14. Tip Guide Tube Ball Valves (Five Valves) 2C51-J004	7	N.A.
15. Reactor Building Closed Cooling Water System Valves	2	≤ 30
2WR029		
2WR040		
2WR179		
2WR180		
16. Primary Containment Chilled Water Inlet Valves	2	
2VP113 A and B		≤ 90
2VP063 A and B		≤ 40
17. Primary Containment Chilled Water Outlet Valves	2	
2VP053 A and B		≤ 40
2VP114 A and B		≤ 90
18. Recirc. Hydraulic Flow Control Line Valves[g]	2	≤ 5
2B33-F338 A and B		
2B33-F339 A and B		
2B33-F340 A and B		
2B33-F341 A and B		
2B33-F342 A and B		
2B33-F343 A and B		
2B33-F344 A and B		
2B33-F345 A and B		
19. Feedwater Testable Check Valves 2B21-F032 A and B	2	N.A.

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP[a]	MAXIMUM ISOLATION TIME (Seconds)
b. Manual Isolation Valves		
1. 2FC086		N.A.
2. 2FC113		N.A.
3. 2FC114		N.A.
4. 2FC115		N.A.
5. 2MC027[1]		N.A.
6. 2MC033[1]		N.A.
7. 2SA042[1]		N.A.
8. 2SA046[1]		N.A.

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

c. Excess Flow Check Valves[g]

1. 2B21-F374
2. 2B21-F376
3. 2B21-F359
4. 2B21-F355

12 5. 2B21-F361

13 6. 2B21-F378

14 7. 2B21-F372

15 8. 2B21-F370

9. 2B21-F363
10. 2B21-F353
11. 2B21-F415A, B
12. 2B21-F357
13. 2B21-F382
14. 2B21-F328A, B, C, D
15. 2B21-F327A, B, C, D
16. 2B21-F413A, B
17. 2B21-F344
18. 2B21-F365
19. 2B21-F443
20. 2B21-F439
21. 2B21-F437
22. 2B21-F441
23. 2B21-F445A, B
24. 2B21-F453

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

Excess Flow Check Valves(g) (Continued)

25. 2B21-F447
26. 2B21-F455A, B
27. 2B21-F451
28. 2B21-F449
29. 2B21-F367
30. 2B21-F326A, B, C, D
31. 2B21-F325A, B, C, D
32. 2B21-F350
33. 2B21-F346
34. 2B21-F348
35. 2B21-F471
36. 2B21-F473
37. 2B21-F469
38. 2B21-F475A, B
39. 2B21-F465A, B
40. 2B21-F467
41. 2B21-F463
42. 2B21-F380
43. 2G33-F312A, B
44. 2G33-F309
45. 2E12-F315
46. 2E12-F359A, B
47. 2E12-F319

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

Excess Flow Check Valves(g) (Continued)

48. 2E12-F317
49. 2E12-F360A, B
50. 2E21-F304
51. 2E22-F304
52. 2E22-F341
53. 2E22-F342
54. 2B33-F319A, B
55. 2B33-F317A, B
56. 2B33-F313A, B, C, D
57. 2B33-F311A, B, C, D
58. 2B33-F315A, B, C, D
59. 2B33-F301A, B
60. 2B33-F307A, B, C, D
61. 2B33-F305A, B, C, D
62. 2CM004
63. 2CM002
64. 2CM012
65. 2CM010
66. 2VQ061
67. 2B21-F457
68. 2B21-F459
69. 2B21-F461
70. 2CM102
71. 2B21-F570
72. 2B21-F571

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

d. Other Isolation Valves

1. MSIV Leakage Control System

2E32-F001A, E, J, N[b]

2. Reactor Feedwater and RWCU System Return

2B21-F010A, B
2B21-F065A, B
2G33-F040**

3. Residual Heat Removal/Low Pressure Coolant Injection System

2E12-F042A, B, C
2E12-F016A, B
2E12-F017A, B
2E12-F004A, B, C[j]
2E12-F027A, B[j]
2E12-F024A, B[j]
2E12-F021[j]
2E12-F302[j]
2E12-F064A, B, C[j]
2E12-F011A, B[j]
2E12-F088A, B, C[j]
2E12-F025A, B, C[j]
2E12-F030[j]
2E12-F005[j]
2E12-F073A, B[j]
2E12-F074A, B[j]
2E12-F055A, B[j]
2E12-F036A, B[j]
2E12-F311A, B[j]
2E12-F041A, B, C[k]
2E12-F050A, B[k]

**For the remainder of Cycle 5, or until the first outage in which the unit is in Cold Shutdown for two weeks or greater duration, the Type C test is not required to be current for the 2G33-F040 valve and its leakage is not required to be included in the total Type B & C leakage specified by Specification 3.6.1.2.b.

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

Other Isolation Valves (Continued)

4. Low Pressure Core Spray System

2E21-F005
2E21-F001(j)
2E21-F012(j)
2E21-F011(j)
2E21-F018(j)
2E21-F031(j)
2E21-F006(k)

5. High Pressure Core Spray System

2E22-F004
2E22-F015(j)
2E22-F023(j)
2E22-F012(j)
2E22-F014(j)
2E22-F005(k)

6. Reactor Core Isolation Cooling System

2E51-F013
2E51-F069
2E51-F028
2E51-F068
2E51-F040
2E51-F031(j)
2E51-F019(j)
2E51-F065(k)
2E51-F066(k)
2E51-F059(m)
2E51-F022(m)
2E51-F362(n)
2E51-F363(n)

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER

Other Isolation Valves (Continued)

7. Post LOCA Hydrogen Control

2HG001A, B
2HG002A, B
2HG005A, B
2HG006A, B

8. Standby Liquid Control System

2C41-F004A, B
2C41-F007

9. Reactor Recirculation Seal Injection

2B33-F013A, B[j]
2B33-F017A, B[j]

10. Drywell Pneumatic System

2IN018

11. Reference Leg Backfill

2C11-F422B
2C11-F422D
2C11-F422F
2C11-F422G
2C11-F423B
2C11-F423D
2C11-F423F
2C11-F423G

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TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

TABLE NOTATIONS

*But ≥ 3 seconds.

(a) See Specification 3.3.2, Table 3.3.2-1, for isolation signal(s) that operates each valve group.
(b) Not included in total sum of Type B and C tests.
(c) May be opened on an intermittent basis under administrative control.
(d) Not closed by SLCS actuation.
(e) Not closed by Trip Functions 5a, b, or c, Specification 3.3.2, Table 3.3.2-1.
(f) Not closed by Trip Functions 4a, c, d, e, or f of Specification 3.3.2, Table 3.3.2-1.
(g) Not subject to Type C leakage test.
(h) Opens on an isolation signal. Valves will be open during Type A test. No Type C test required.
(i) Also closed by drywell pressure-high signal.
(j) Hydraulic leak test at 43.6 psig.
(k) Not subject to Type C leakage test - leakage rate tested per Specification 4.4.3.2.2.
(l) These penetrations are provided with removable spools outboard of the outboard isolation valve. During operation, these lines will be blind flanged using a double O-ring and a type B leak test. In addition, the packing of these isolation valves will be soap-bubble tested to ensure insignificant or no leakage at the containment test pressure each refueling outage.
(m) If valves 2E51-F362 and 2E51-F363 are locked closed and acceptably leak rate tested, then valves 2E51-F059 and 2E51-F022 are not considered to be primary containment isolation valves and are not required to be leak rate tested.
(n) Either the 2E51-F362 or the 2E51-F363 valve may be open when the RCIC system is in the standby mode of operation, and both valves may be open during operation of the RCIC system in the full flow test mode, providing that:

    1) valve 2E51-F022 is acceptably leak rate tested, and

    2) valve 2E51-F059 is deactivated, locked closed and acceptably leak rate tested, and

    3) the spectacle flange, installed immediately downstream of the 2E51-F059 valve, is closed and acceptably leak rate tested.

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Information Only
No Changes

3/4.6.4 VACUUM RELIEF

LIMITING CONDITION FOR OPERATION

3.6.4 All suppression chamber - drywell vacuum breakers shall be OPERABLE and closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one suppression chamber - drywell vacuum breaker inoperable and/or open, within 4 hours close the manual isolation valves on both sides of the inoperable and/or open vacuum breaker. Restore the inoperable and/or open vacuum breaker to OPERABLE and closed status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one position indicator of any OPERABLE suppression chamber - drywell vacuum breaker inoperable, restore the inoperable position indicator to OPERABLE status within 14 days or visually verify the vacuum breaker to be closed at least once per 24 hours. Otherwise, declare the vacuum breaker inoperable.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each suppression chamber - drywell vacuum breaker shall be:

a. Verified closed at least once per 7 days.

b. Demonstrated OPERABLE:

1. At least once per 31 days and within 12 hours after any discharge of steam to the suppression chamber from the safety-relief valves, by cycling each vacuum breaker through at least one complete cycle of full travel.

2. At least once per 31 days by verifying both position indicators OPERABLE by performance of a CHANNEL FUNCTIONAL TEST.

3. At least once per 18 months by;

a) Verifying the force required to open the vacuum breaker, from the closed position, to be less than or equal to 0.5 psid, and

b) Verifying both position indicators OPERABLE by performance of a CHANNEL CALIBRATION.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.4.2 The manual isolation valves on both sides of an inoperable and/or open suppression chamber-drywell vacuum breaker shall be verified to be closed at least once per 7 days.

4.6.4.3 Vacuum breaker header flanges which have been broken shall be leak tested after remaking by Type B test at 39.6 psig per Specification 4.6.1.2.d.

e

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

DRYWELL AND SUPPRESSION CHAMBER HYDROGEN RECOMBINER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent drywell and suppression chamber hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one drywell and/or suppression chamber hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each drywell and suppression chamber hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 92 days by cycling each flow control valve and recirculation valve through at least one complete cycle of full travel.

b. At least once per ~~6~~ 18 months by verifying, during a recombiner system functional test:

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

   2. That the reaction chamber gas temperature increases to 1200 ± 25°F within 2 hours.

c. At least once per 18 months by:

   1. Performing a CHANNEL CALIBRATION of all recombiner operating instrumentation and control circuits.

   2. Verifying the integrity of all heater electrical circuits by performing a resistance to ground test within 30 minutes following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 100,000 ohms.

d. By measuring the leakage rate:

   1. As a part of the overall integrated leakage rate test required by Specification 3.6.1.2, or

   2. By measuring the leakage rate of the system outside of the containment isolation valves at P_a, 39.6 psig, on the schedule required by Specification 4.6.1.2 and including the measured leakage as a part of the leakage determined in accordance with Specification 4.6.1.2.

e

Add "INSERT H"

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

The structural integrity of the primary containment is ensured by the successful completion of the Inservice Inspection Program for Post Tensioning Tendons and by associated visual inspections of the steel liner and penetrations for evidence of deterioration or breach of integrity. This ensures that the structural integrity of the primary containment will be maintained in accordance with the provisions of the Primary Containment Tendon Surveillance Program. Testing and Frequency are consistent with the recommendations of Regulatory Guide 1.35, Revision 3, except that the Unit 1 and 2 primary containments shall be treated as twin containments even though the Initial Structural Integrity Tests were not within 2 years of each other.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 39.6 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J to 10 CFR 50 with the exception of exemption(s) granted for main steam isolation valve leak testing and testing the airlocks after each opening.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitation on closure and leak rate for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

3/4.6.1.1

3/4.6.1.2 DELETED

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INSERT H

PRIMARY CONTAINMENT INTEGRITY is maintained by limiting overall integrated leakage to $\leq 1.0\ L_a$ and the Type B and C combined leakage rate acceptance criterion is $\leq 0.60\ L_a$. Prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test, the combined Type B and C leakage must be $< 0.60\ L_a$, and the overall Type A leakage must be $< 0.75\ L_a$ when a Type A test is scheduled. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

The maximum allowable leakage rate for the primary containment (L_a) is 0.635% by weight of the containment atmosphere per day at the calculated maximum peak containment pressure (P_a) of 39.6 psig.

Individual leakage rates specified for the primary containment air lock, main steam lines through the isolation valves, and valves in hydrostatically tested lines are addressed in LCO 3.6.1.3, and Surveillance Requirement 4.6.3.6.

Surveillance Requirement 4.6.1.1.b maintains PRIMARY CONTAINMENT INTEGRITY by requiring compliance with the visual examinations and leakage rate test requirements of 10 CFR 50, Appendix J, as modified by approved exemptions. Failure to meet air lock leakage testing (4.6.1.3) or main steam isolation valve leakage (4.6.3.6.a) does not necessarily result in a failure of this Surveillance Requirement, 4.6.1.1.b. The impact of the failure to meet these Surveillance Requirements 4.6.1.3 and 4.6.1.1.b must be evaluated against the Type A, B, and C acceptance criteria of 10 CFR 50, Appendix J, as modified by approved exemptions. The leakage limits for main steam lines through the isolation valves and leakage test results of Surveillance Requirement 4.6.3.6.a are not included in the total sum of Type B and C tests (approved exemption). As-left leakage prior to the first startup after performing a required 10 CFR 50, Appendix J, leakage test is required to be $< 0.60\ L_a$ for combined Type B and C leakage, and $< 0.75\ L_a$ for overall Type A leakage. At all other times between required Type A tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. The combined Type B and C leakage remains as $\leq 0.60\ L_a$ between scheduled tests, in accordance with Appendix J.

The Frequency is required by 10 CFR 50, Appendix J, as modified by approved exemptions. Thus, 4.0.2 (which allows Frequency extensions) does not apply to Surveillance Requirement 4.6.1.1.b.

DEPRESSURIZATION SYSTEMS (Continued)

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety-relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety-relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety relief valve to assure mixing and uniformity of energy insertion to the pool.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

9

The OPERABILITY of the primary containment isolation valves 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. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

Add "INSERT J"

3/4.6.4 VACUUM RELIEF

Vacuum relief breakers are provided to equalize the pressure between the suppression chamber and drywell. This system will maintain the structural integrity of the primary containment under conditions of large differential pressures.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There are four valves to provide redundancy so that operation may continue for up to 72 hours with one vacuum breaker inoperable provided that the manual isolation valves on each side are in the closed position.

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INSERT J

Primary Containment Isolation Valves (PCIVs) form a part of the primary containment boundary. The PCIV safety function is related to control of primary containment leakage rates during accidents or other conditions to limit the untreated release of radioactive materials from the containment in excess of the design limits.

The automatic isolation valves are required to have isolation times within limits and actuate on an automatic isolation signal. The valves covered by this specification are listed with their associated stroke times, and other design information for lines penetrating the Primary Containment, in UFSAR Section 6.2.

The normally closed isolation valves are considered OPERABLE when manual valves are closed, automatic valves are de-activated and secured in their closed position, blind flanges are in place, and closed systems are intact.

Main steam lines through the isolation valves and hydrostatically tested valves must meet alternative leakage rate requirements. Other PCIV leakage rates are addressed by specification 3/4.6.1.1, "PRIMARY CONTAINMENT INTEGRITY". UFSAR Section 6.2 also describes special leakage test requirements and exemptions.

This specification provides assurance that the PCIVs will perform their designed safety functions to control leakage from the primary containment during accidents.

The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the primary containment.

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INSERT J (Continued)

Surveillance Requirement 4.6.3.6.a verifies leakage through all four main steam lines is ≤ 100 scfh when tested at ≥ P_t (25.0 psig). The transient and accident analyses are based on leakage at the specified leakage rate. The leakage rate for main steam lines through the isolation valves must be verified to be in accordance with the leakage test requirements of 10 CFR 50, Appendix J, as modified by approved exemptions. A Note has been added to this Surveillance Requirement requiring the results to be excluded from the total of Type B and Type C tests. This ensures that leakage rate for main steam lines through the isolation valves is properly accounted for in accordance with an approved exemption. The frequency is "at least once per 18 months" in accordance with an approved exemption.

Surveillance Requirement 4.6.3.6.b test of hydrostatically tested lines provides assurance that the assumptions of UFSAR Section 6.2 are met. The combined leakage rates must be demonstrated in accordance with the leakage rate test at a frequency of "at least once per 18 months". A Note has been added to this Surveillance Requirement requiring the results to be excluded the total of Type B and Type C tests. This is in accordance with 10 CFR 50, Appendix J, and approved exemptions.

the BWR Owners Group Report SLI-8211 and SLI-8218 and the recommendations of the BWR Owners Group reports. Any required modifications shall be completed on a scheduled acceptable to the NRC staff.

d. <u>Modification of Automatic Depressurization System Logic - Feasibility for Increased Diversity for Some Event Sequences (II.K.3.18, SER, SER #1, SSER #3, SSER #5)</u>

Prior to startup after the first refueling outage, the licensee shall:

(i) Install modifications to the Automatic Depressurization system described in the licensee's letter dated July 1, 1983. The final circuit diagrams and an analysis of the bypass timer time delay shall be submitted for NRC staff review and approval prior to installation.

Add "INSERT L" attached

(ii) Incorporate into the Plant Abnormal Procedures the usuage of the inhibit switch; and

(iii) Modify the Technical Specifications to provide the bypass timer ~~and~~ manual inhibit switch.

D. The facility requires exemptions from certain requirements of 10CFR Part 50, 10 CFR Part 70, and 10CFR Part 73. These include: (a) Exemptions from certain requirements of Appendices G, H and J to 10 CFR Part 50, and to 10 CFR Part 73 (b) are described in the Safety Evaluation Report and Supplement ~~Numbers~~ 1, 2, 3, and 5 to the Safety Evaluation Report. ~~In addition,~~ ~~an~~ exemption was requested until completion of the first refueling from the requirements of 10 CFR 70.24. These 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, these exemptions are hereby granted pursuant to 10 CFR 50.12. With the granting of ~~this~~ these exemptions the facility will operate to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. Before engaging in additional construction or operational activities which may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement and its Addendum, the licensee shall provide a written notification to the Director of the Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

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INSERT L (Unit 2, NPF-18)

(c) An exemption from the requirement of paragraph III.D of Appendix J to conduct the third Type A test of each ten-year service period when the plant is shutdown for the 10-year plant inservice inspections. (d) A one-time exemption from the requirement of paragraph III.A.6(b) of Appendix J to resume a Type A test schedule of three times in ten years. Exemptions (c) and (d) are described in the safety evaluation accompanying Amendment No. ____ to this license.

ATTACHMENT C
SIGNIFICANT HAZARDS CONSIDERATION

Commonwealth Edison has evaluated the proposed Technical Specification Amendment and determined that it does not represent a significant hazards consideration. Based on the criteria for defining a significant hazards consideration established in 10 CFR 50.92, operation of LaSalle County Station Units 1 and 2 in accordance with the proposed amendment will not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated because of the following:

   a. The relocation of Technical Specification 3/4.6.1.2, Primary Containment Leakage, and Surveillance Requirements 4.6.1.1.a, 4.6.4.3, and 4.6.6.1.d to specification 3/4.6.1.1, Primary Containment Integrity, as Surveillance Requirement 4.6.1.1.b continues to assure that Primary Containment leakage is maintained within the analyzed limit assumed for accident analysis by testing in accordance with 10 CFR part 50, Appendix J as modified by approved exemptions.

      The requirement to be less than 0.75 L_a for as-left Type A test and less than 0.60 L_a for Type B and C tests prior to first unit startup following testing performed in accordance with 10 CFR part 50, Appendix J, as modified by approved exemptions, provides margin for degradation between tests and thus primary containment integrity is maintained during the time period between required leakage testing. The current Limiting Condition for Operation 3.6.1.2 in conjunction with Surveillance Requirements 4.6.1.2 basically require the same leakage limits as proposed Surveillance Requirement 4.6.1.1.b. The Limiting Condition for Operation (LCO) is required to be less than 1.0 L_a and is applicable during a fuel cycle for the Type A test. The LCO for Type B and C combined leakage total is currently required to be less than 0.60 L_a. The proposed Surveillance Requirement maintains the following:

      1. The current LCO for Overall Containment leakage (as determined by a Type A test) and for the Type B and C combined leakage during the cycle by requiring overall containment leakage to be less than 1.0 L_a and Type B and C leakage total less than 0.60 L_a.

2. The associated limits specified in the current Action Statements are maintained by verifying Overall Containment leakage to be less than 0.75 L_a and Type B and C leakage total less than 0.60 L_a prior to startup from an outage in which the applicable leakage testing is conducted.

Therefore, there is no change to the consequences of an accident previously evaluated, because maintaining leakage within the analyzed limit assumed for accident analysis does not change either the onsite or offsite dose consequences resulting from an accident. In addition to this, containment leakage is not an accident initiator, so there is no effect on the probability of accident initiators. Thus there is no increase in the probability of an accident previously analyzed.

b. Relocation of Technical Specification table of Primary Containment Isolation Valves, Table 3.6.3-1, to the LaSalle UFSAR is an administrative change to remove the component list of Primary Containment Isolation Valves, Table 3.6.3-1, from the Technical Specifications. The Limiting Condition for Operation (LCO), 3.6.3, is being revised to define which components the LCO applies to. The wording of the revised LCO encompasses all of the components listed in the current Technical Specification Table 3.6.3. Removal of this component list does not change the probability of any accident initiators or change any other relevant initial assumptions. Also, there is no change to the consequences of an accident previously evaluated, because removing this list from Technical Specifications does not change either the onsite or offsite dose consequences resulting from the event. The component list will be controlled by an Administrative Procedure and can only be changed by the 10 CFR 50.59 change process with review and approval per the Onsite Review and Investigative Function. Therefore, there is no increase in either the probability or consequences of an accident previously evaluated.

c. The change in the functional test interval for the Drywell and Suppression Chamber Hydrogen Recombiner systems from "once per 6 months" to "once per 18 months" was determined by the NRC in NUREG 1366 and Generic Letter 93-05 to be acceptable by evaluation of the industry Licensing Event Reports (LERs) to assess the reliability of hydrogen recombiners. The conclusion was that the interval should be changed, because of the redundancy and apparent high reliability. A review of LaSalle LERs has shown only one LER that involved the operability of the hydrogen recombiner system and that was due to a Part 21 issue regarding circuit breaker environmental qualification. The breakers were replaced with qualified breakers. Therefore, the LaSalle Hydrogen Recombiner reliability is consistent with or better than that found by the NRC in determining this surveillance interval extension based on all LERs. Also, redundancy is the same as that assumed by the NRC; because, LaSalle has two hydrogen recombiner subsystems that are shared by Unit 1 and Unit 2. Both hydrogen recombiners subsystems are required to be Operable for either or both units in Operational Conditions 1 and 2. Based on LaSalle operating experience, the hydrogen recombiner subsystems are expected to continue to be demonstrated operable when the functional test is performed at an 18 month frequency.

Therefore, there is minimal or no change to the consequences of an accident previously evaluated, because at least one of the hydrogen recombiner subsystems is expected to be available to meet its design function to reduce the potential for hydrogen explosion or hydrogen burn in the primary containment. By preserving the integrity of the primary containment, there is no change to either the onsite or offsite dose consequences resulting from an accident. In addition to this, control of hydrogen concentration by use of a hydrogen recombiner subsystem is not an accident initiator, so there is no effect on the probability of accident initiators. Thus there is no significant increase in the probability of an accident previously analyzed.

d. The first exemption request is from the requirements of paragraph III.A.6(b) of Appendix J to allow LaSalle County Station Unit Two to return to or resume a Type A test schedule of three times in ten years (40 ±10 months). Due to consecutive failures, 10 CFR 50 Appendix J requires that Type A tests be performed every refueling outage on Unit Two until two consecutive Type A tests are satisfactory. 10 CFR Part 50 has an exemption process and is specified in 10 CFR Part 50.12(a), which states:

> "The Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of the regulations of this part,..."

The exemption process requires showing that the granting of the exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. Also, special circumstances are required to be present for the granting of an exemption. One of the special circumstances that would apply in this instance is 10 CFR part 50.12(a)(2)(ii) which state

> "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule".

This requires that it be shown that unacceptable containment leakage will be identified and corrected, by alternative methods. The alternative method is specifically Type B and C tests, which will identify any local penetration leakage. This is acceptable, because Type C test failures have been the cause for failures of as-found Type A tests in the LaSalle Unit 2 first, third, and fourth refueling outages.

Exceeding the allowable leakage rate during the performance of the Type A test is indicative of either a passive or a structural component that is leaking or that there is an inadequacy in the Local Leak Rate Test (Type B and C tests) program. When the failure of a Type A test is due to a passive or structural component, the only test for adequate repair would be the Type A test. For a Local Leak Rate Test program inadequacy, the

Type A test would serve as a means of verification of the results of the test program. The Type A tests have not found new significant Type B or C tested local penetration leakage that has not been identified by Type B or C testing alone. Therefore, the LaSalle Local Leak Rate Test program is adequate to find and correct Type B and C containment penetration leakage.

When it is determined that Type A tests failed as a direct result of as-found Type B and C minimum path leakage penalty additions and not due to a non Type B or C tested components or structures, then performance of the Type A test more frequently as required by 10 CFR Part 50, Appendix J, due only to Type B and C test failures is redundant to the performance of Type B and C tests. Therefore, Type B or C tested penetration leakage that can be determined by Type B or C tests is evaluated and corrected, as applicable, to maintain overall containment leakage within limits, without an additional Type A test.

Primary Containment leakage which includes the minimum path Primary Containment Isolation Valve leakage is an assumption in any analyzed accident which could involve an offsite radioactive release. Because performance of Type B and C tests will find and allow correction/repair of leaking valves/penetrations, verification of as-found and as-left local leakage assures that Primary Containment leakage will be within the analyzed limit assumed for accident analysis.

Therefore, for this one-time exemption for LaSalle Unit 2, there is little or no increase in the consequences of an accident previously evaluated involving the dose previously calculated either onsite or offsite at the site boundary due to any analyzed accident. In addition to this, containment leakage is not an accident initiator, so there is no effect on the probability of accident initiators. Thus there is no significant increase in the probability of an accident previously analyzed.

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

1.0 1.1 1.25 1.4 1.6 1.8 2.0 2.2 2.5 2.8 3.2 3.6 4.0

150mm

6"

e. The request for a partial exemption from paragraph III.D of Appendix J to 10 CFR 50 involves a deletion of the requirement to perform the third Type A test for each 10-year service period during the shutdown for the 10-year plant inservice inspections. There is no significant benefit in coupling these two surveillances (i.e., the Type A test and the 10-year ISI program). Each of the two surveillances is independent of the other and provides assurance of different plant characteristics. The Type A test assures the required leak-tightness for the reactor containment building be less than Appendix J acceptance criteria. This demonstrates compliance with the guidelines of 10 CFR Part 100 based on the assumptions used in the UFSAR which conform to NRC Safety Guide 4. The 10-year ISI program provides assurance of the integrity of the plant structures, systems, and components in compliance with 10 CFR 50.55(a). There is no safety-related concern necessitating their coupling to the same refueling outage. As a result, this change cannot increase the consequences (i.e., offsite dose) of any accident previously evaluated. Furthermore, since the decoupling of the test schedules has no affect on the test's effectiveness, decoupling their schedules will not increase the probability of an accident.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated because:

a. Technical Specification 3/4.6.1.2, Primary Containment Leakage, and Surveillance Requirements 4.6.1.1.a, 4.6.4.3, and 4.6.[illegible].d are being relocated to specification 3.4.6.1.1, Primary Containment Integrity, as Surveillance Requirement 4.6.1.1.b. The proposed Surveillance Requirement 4.6.1.1.b assures that Primary Containment leakage is maintained within the analyzed limit assumed for accident analysis by testing in accordance with 10 CFR part 50, Appendix J as modified by approved exemptions. Primary containment leakage is an assumption in accident analyses, and is maintained by both the current specifications and the proposed specification. The leakage does not cause an accident and no new failure modes are created. Therefore this request for exemption does not create the possibility of a new or different kind of accident from any accident previously evaluated.

b. This is an administrative change to control the list of Primary Containment Isolation Valves outside the LaSalle Unit 1 and Unit 2 Technical Specifications. The administrative controls provided to control this component list assure that the design and operation of the plant will continue to be in accordance with the UFSAR, Facility License and the associated Technical Specifications. Therefore, the possibility of a new or different kind of accident from any previously evaluated is not created.

c. The change in the functional test interval for the Drywell and Suppression Chamber Hydrogen Recombiner systems from "once per 6 months" to "once per 18 months" is based on good equipment performance on a 6 month frequency. The expected outcome of the 18 month surveillances, based on the low failure rate at a six month frequency, is to show the hydrogen recombiner subsystems Operable. This system is for mitigating the consequences of an accident that causes generation of hydrogen and oxygen in the primary containment. No new failure modes are created by this change in surveillance frequency. Therefore, the possibility of a new or different kind of accident from any previously evaluated is not created.

d. The first exemption is from the requirements of paragraph III.A.6(b) of Appendix J to allow LaSalle County Station Unit Two to return to or resume a Type A test schedule of three times in ten years (40 ±10 months). Containment leakage testing, including both Type B and C testing and Type A testing as specified in the LaSalle County Station Safety Analysis Report were evaluated in Section 6.2.6 of Safety Evaluation Report, NUREG-0519, and found to be acceptable. Since Type B and C testing will find and verify correction of penetration leakage when Type B and C test as-found penalties are specifically what caused the failure of the as-found Type A tests, then Type B and C testing will provide adequate assurance of the continued integrity of the Primary Containment without increasing the frequency of Type A tests. As a result, the Primary Containment will continue be maintained as designed and previously evaluated.

Based on this, the requirement of two acceptable as-found Type A tests prior to returning to the Appendix J paragraph III.D frequency of three times in ten years (40 ±10 months) is not necessary to assure that the primary containment remains within the analyzed leakage limits. Containment leakage is an assumption for the dose consequences of accident analyses, and not an accident initiator. Also, no new failure modes are created by this exemption. Therefore this Amendment does not create the possibility of a new or different kind of accident.

e. The request for a partial exemption from paragraph III.D of Appendix J to 10 CFR 50 involves a deletion of the requirement to perform the third Type A test for each 10-year service period during the shutdown for the 10-year plant inservice inspections. The proposed exemption does not involve any change to the plant design or operation. As discussed above, this change cannot increase the consequences of any accident previously evaluated. As a result, no new failure modes are created. Therefore, this proposed change cannot create the possibility of any new of different kind of accident from any accident previously evaluated.

3) Involve a significant reduction in the margin of safety because:

a. Technical Specification 3/4.6.1.2, Primary Containment Leakage, and Surveillance Requirements 4.6.1.1.a, 4.6.4.3, and 4.6.6.1.d are being relocated to specification 3/4.6.1.1, Primary Containment Integrity, as proposed Surveillance Requirement 4.6.1.1.b. The proposed Surveillance Requirement 4.6.1.1.b continues to assure that Primary Containment leakage is maintained within the analyzed limit assumed for accident analysis by testing in accordance with 10 CFR part 50, Appendix J as modified by approved exemptions.

As stated in 1)a. above, the proposed Surveillance Requirement 4.6.1.1.b maintains the acceptance criteria and limits for continued operation of the current specification for primary containment leakage. Therefore, the margin of safety is not reduced by this change. Also, the proposed

addition of a definition for the maximum allowable primary containment leakage rate assures that the margin of safety is maintained.

The leakage limits for MSIVs and hydrostatically tested valves are maintained by relocating the current surveillance requirements to specification 3/4.6.3, with the acceptance criteria of the current specification retained. Thus preserving the current margin of safety by maintaining the leakage rates as assumed in the accident analyses.

b. The Limiting Condition for Operation for Technical Specification 3.6.3, Primary Containment Isolation Valves, is revised by this Technical Specification change to specifically define the components to which the LCO applies. Therefore, removal of Technical Specification Table 3.6.3-1, which lists the specific components to which the LCO applies does not change the scope or applicability of the specification. The component list will be controlled administratively with any changes to the list made in accordance with the 10 CFR 50.59 change process. Therefore, this is an administrative change only and there is no reduction in the margin of safety.

c. The change in the functional test interval for the Drywell and Suppression Chamber Hydrogen Recombiner systems from "once per 6 months" to "once per 18 months" is based on good equipment performance on a 6 month frequency. The expected outcome of the 18 month surveillances, based on the low failure rate at a six month frequency, is to show the hydrogen recombiner subsystems Operable. The change in frequency has no affect on the hydrogen or oxygen generation assumptions or the recombination rate of the hydrogen recombiner subsystems. Therefore, the margin of safety is not reduced or changed by this surveillance interval change.

d. The first exemption is from the requirements of paragraph III.A.6(b) of Appendix J to allow LaSalle County Station Unit Two to return to or resume a Type A test schedule of three times in ten years (40 ±10 months). The limit of total leakage determined from Type B and C tests will remain the same, providing a margin of 40 percent to the maximum

allowable containment leakage rate (L_a) at the design basis accident pressure specified in proposed Technical Specification definition of L_a. This 40 percent is as specified by 10 CFR Part 50, Appendix J. In addition to this, administrative guidelines have been set for eachpenetration/ vaive, so that any abnormal leakage will be corrected by adjustment or repair as needed. Any postponement of repairs is based on a technical evaluation and then only if the total Type B and Type C leakage is maintained at less than 0.60 L_a. Repairs will be required to restore the leakage rate to less than the administrative limit at the next refueling outage.

This request for exemption is based the fact that Type B and C testing minimum path leakage rate penalties are the direct cause of the failure of as-found Type A tests. The leakage through Type B and C tested penetrations is best measured and corrected via a local leak test. Therefore, verification of an adequate margin of safety is assured by conducting Type B and C tests, and not another increased frequency Type A test.

e. The request for a partial exemption from paragraph III.D of Appendix J to 10 CFR 50 involves a deletion of the requirement to perform the third Type A test for each 10-year service period during the shutdown for the 10-year plant inservice inspections. The proposed exemption does not change the acceptance criteria that must be met for inservice inspections, does not relax the condition of containment that must be met prior to plant restart, and does not change the requirements that must be met between plant refueling outages. Therefore, the proposed change does not result in a significant reduction in the margin of safety.

Guidance has been provided in "Final Procedures and Standards on No Significant Hazards Considerations," Final Rule, 51 FR 7744, for the application of standards to license change requests for determination of the existence of significant hazards considerations. This document provides examples of amendments which are and are not considered likely to involve significant hazards considerations.

ATTACHMENT C
SIGNIFICANT HAZARDS CONSIDERATION

a. The relocation of Technical Specification 3/4.6.1.2, Primary Containment Leakage, and Surveillance Requirements 4.6.1.1.a, 4.6.4.3, and 4.6.6.1.d to specification 3/4.6.1.1, Primary Containment Integrity, as Surveillance Requirement 4.6.1.1.b most closely fits the example of a change which may either result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in Standard Review Plan 6.2.6, Containment Leakage Testing. The proposed surveillance requirements and definition retain the current specification limits and acceptance criteria, thus preserving the safety margin and will not increase the consequences of an accident.

b. These proposed amendment for relocation of Table 3.6.3-1 outside of Technical Specifications most closely fits the example of an administrative change to the Technical Specifications. The Limiting Condition for Operation for the affected specification is being revised to clearly define the components to which the specification applies, so the Technical Specification Table which lists the components is redundant to the revised LCO and can be removed without changing the application or interpretation of the LCO. Likewise, other references in Technical Specifications to Table 3.6.3-1 are being changed to clearly define the components to which the specifications apply. Any changes to the component list after removal from Technical Specifications will be performed under Administrative control and in accordance with the 50.59 change process, which will assure compliance with the Technical Specification.

c. The change in the functional test interval for the Drywell and Suppression Chamber Hydrogen Recombiner systems from "once per 6 months" to "once per 18 months" most closely fits the example of a change which may either result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan. The system design is not changed, and therefore the acceptability of the subsystem design is not changed. The frequency extension is consistent with the reliability of the equipment, and thus the hydrogen recombination subsystems will be verified to function as designed by functional testing on an 18 month frequency. Therefore, this functional test interval increase is consistent with Standard review Plan 6.2.5, Combustible Gas Control in Containment.

d. The first exemption is from the requirements of paragraph III.A.6(b) of Appendix J to allow LaSalle County Station Unit Two to return to or resume a Type A test schedule of three times in ten years (40 ±10 months). This proposed exemption most closely fits the example of a change which may either result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan.

When as-found Type A tests fail as a direct result of Type B and C test minimum path leakage penalties, then Type B and C testing in conjunction with the corrective and preventative maintenance for penetrations determined to be poor performers through a comprehensive corrective action plan will assure that the Primary Containment integrity will be maintained without additional Type A tests. Based on this, allowing exemption to the requirement of two acceptable as-found Type A tests prior to returning to the Appendix J paragraph III.D frequency of three times in ten years (40 ±10 months) is not necessary to assure that the primary containment remains within the analyzed leakage limits. The Standard Review Plan section 6.2.6 basically verifies Containment leak rate testing is in conformance to 10 CFR 50 Appendix J. The above provides confidence that the requested exemption assures adequate Containment leak rate testing will be conducted to maintain overall containment leakage within all acceptable criteria.

e. The request for a partial exemption from paragraph III.D of Appendix J to 10 CFR 50 involving a deletion of the requirement to perform the third Type A test for each 10-year service period during the shutdown for the 10-year plant inservice inspections most closely fits the example of a change which may either result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in Standard Review Plan 6.2.6, Containment Leakage Testing. The conduct of Type A tests in accordance

with Appendix J at a frequency of three times in ten years (40 ±10 months) provides assurance of ongoing acceptable overall integrated primary containment leakage without a test coinciding with the last outage of the ten year Inservice Inspection program.

This proposed amendment does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings or a significant relaxation of the bases for the limiting conditions for operations. Therefore, based on the guidance provided in the Federal Register and the criteria established in 10 CFR 50.92(c), the proposed change does not constitute a significant hazards consideration.

ATTACHMENT D

EXEMPTION REQUEST FROM APPENDIX J REQUIREMENTS FOR THE TYPE A INCREASED FREQUENCY AS A RESULT OF CONSECUTIVE FAILURES

BACKGROUND

The primary objective of the regulations is to assure continuance of primary containment leak-tight integrity by the conduct of Type A, B, and C tests on a periodic basis. 10 CFR 50 Appendix J establishes two major categories of tests with separate criteria. The Type B and C tests, Local Leak Rate Tests (LLRTs), are performed during each refueling outage while the Type A test, Containment Integrated Leak Rate Test (CILRT), is only performed every 40 ± 10 months to achieve three Type A tests in a ten year period. The Type B and C tests provide periodic surveillances of components such as isolation valves and resilient seals. The Type A test provides a measure of the overall integrated leakage rate of the containment, including passive and structural components.

Exceeding the allowable leakage rate during the performance of the Type A test is indicative of either a passive or a structural component that is leaking, or an inadequacy in the local leak rate test program. When the failure of a Type A test is due to a passive or structural component, the only test for adequate repair would be the Type A test. If Type B and C leakage rates constitute an identified contributor to a Type A test failure, then as-left leakage rates are best determined by the associated Type B or Type C test.

However, 10 CFR 50 Appendix J, Section III.A.6.(b), requires an increase in the frequency of Type A tests irregardless of the cause of two or more consecutive failures of as-found Type A tests.

LaSalle County Station (LaSalle) Unit 2 has experienced Containment Integrated Leak Rate Test (CILRT) (Type A test) failures for the "as-found" condition at the first, third and fourth refueling outages as a result of penalties from Local Leak Rate Test (LLRT) (Type B and C) failures. Therefore the requirements of 10 CFR 50 Appendix J, Section III.A.6(b) are applicable, necessitating that a Type A test be performed at both the fifth refueling outage (L2R05) and the sixth refueling outage (L2R06) due to consecutive "as-found" Type A test failures. The Type A test performed at the end of L2R05 was acceptable in the as-found condition.

ATTACHMENT D
EXEMPTION REQUEST FROM APPENDIX J REQUIREMENTS FOR THE TYPE A INCREASED FREQUENCY AS A RESULT OF CONSECUTIVE FAILURES

BACKGROUND (continued)

Appendix J establishes two types of tests with separate criteria. The Local Leak Rate Tests (LLRT) (Type B and C) are performed during each refueling outage while the frequency of the Type A test is as follows per 10 CFR 50, Appendix J section III.D.1.(a) Periodic retest schedule:

> "(a) After the preoperational leakage rate tests, a set of three Type A tests shall be performed, at approximately equal intervals during each 10-year service period. The third test of each set shall be conducted when the plant is shutdown for the 10-year plant inservice inspections."

Likewise, the current LaSalle Unit 2 Technical Specification Surveillance Requirement 4.6.1.2.a restates the Appendix J requirement:

> Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10 month intervals during shutdown at P_a, 39 6 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

Therefore, Type A tests are performed every three or four years (40 ± 10 months) to achieve three Type A tests in ten years, with the third Type A test coinciding with the tenth year of each ten year inservice inspection interval. The Type B and C tests provide periodic surveillances of components such as isolation valves and resilient seals. The Type A test provides a measure of the overall integrated leakage rate of the containment including testing of passive and structural components, thereby verifying the overall leakage integrity of the primary containment.

Based on IE Information Notice 85-71 dated August 22, 1985 and an exemption granted to Iowa Electric Light and Power Company for the Duane Arnold Energy Center, the following discussion and attachments provide justification for the granting of a similar exemption to Commonwealth Edison Company for LaSalle County Station Unit Two. Upon approval, this exemption will allow LaSalle Unit 2 to return to a frequency of three times in ten years (40 ±10 months).

BACKGROUND (continued)

The granting of an exemption is contingent upon the Commission's approval of proposed Technical Specification amendments for LaSalle County Station Unit 1 and Unit 2. The proposed amendments are being submitted in conjunction with this exemption request.

DISCUSSION

Pursuant to 10 CFR Part 50.12(a), Commonwealth Edison Company (CECo), requests a one-time exemption from the requirement to conduct an additional Type A test) at LaSalle County Station, Unit Two as required by 10 CFR Part 50, Appendix J, Section III.A.6(b) which states,

> "If two consecutive periodic Type A tests fail to meet the applicable acceptance criteria in III.A.5.(b), not withstanding the periodic retest schedule of III.D., a Type A test shall be performed at each shutdown for refueling or approximately every 18 months, whichever occurs first, until two consecutive Type A tests meet the acceptance criteria in III.A.5.(b), after which time the retest schedule specified in III.D may be resumed."

Likewise, LaSalle Unit 1 and 2 Technical Specifications 4.6.1.2.b require the following:

> "If any periodic Type A test fails to meet 0.75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a, at which time the above test schedule may be resumed."

This exemption is requested to allow LaSalle to return to or resume a Type A test schedule of three times in ten years (40 ±10 months), as specified in 10 CFR Part 50, Appendix J, Section III.D.

DISCUSSION (continued)

Appendix J establishes two types of tests with separate criteria. The Local Leak Rate Tests (LLRT) (Type B and C) are performed during each refueling outage while the Type A test is only performed every three to four years to achieve three Type A tests in a ten year period. The Type B and C tests provide periodic surveillances of components such as isolation valves and resilient seals. The Type A test provides a measure of the overall integrated leakage rate of the containment, including passive and structural components.
Exceeding the allowable leakage rate during the performance of the Type A test is indicative of either a passive or a structural component that is leaking, or an inadequacy in the local leak rate test program. When the failure of a Type A test is due to a passive or structural component, the only test for adequate repair would be the Type A test.

The LaSalle Unit 2 Type A tests for the "as-found" condition at the first, third and fourth refueling outages failed as a result of as-found penalties from Type B and C test minimum path leakage penalty additions. The failures were not due to a non-local leak rate tested component or structure. As a result of the as-found Type A test failures, 10 CFR 50 Appendix J, Section III.A.6(b) requires a Type A test at 18 month frequencies until two consecutive tests are less than 0.75 L_a.

The first of the increased frequency Type A tests was performed during the LaSalle Unit 2 fourth refueling outage (L2R04) and was unacceptable. The second of the increased frequency Type A tests was performed during the LaSalle Unit 2 fifth refueling outage (L2R05) and was acceptable. The third of the increased frequency Type A tests is due to be performed during the upcoming sixth refueling outage (L2R06).

The third LaSalle Unit 2 Type A test, performed in the fourth refueling outage, L2R04, exceeded the 0.75 L_a limit, yet remained below the 1.0 L_a limit. This failure was overwhelmingly due to a local failure of one penetration with two containment isolation valves in the reactor water cleanup system. The leakage of these valves accounted for more than 72% of the Type B and C test penalties which were applied to the "as-found" Type A test results.

ATTACHMENT D
EXEMPTION REQUEST FROM APPENDIX J REQUIREMENTS FOR THE TYPE A INCREASED FREQUENCY AS A RESULT OF CONSECUTIVE FAILURES

DISCUSSION (continued)

The LaSalle Station L2R04 as-found Type A test did not exceed the 1.0 L_a criteria. This was a non-reportable event even though the 0.75 L_a criteria was exceeded, due to the current LaSalle Technical Specification Limiting Condition for Operation 3.6.1.2. of leakage less than L_a, not less than 0.75 L_a. Thus, the design leakage for the primary containment was not exceeded during the previous 18 month cycle.

The following table references the Licensee Event Reports (LER's) and "Reactor Containment Building Integrated Leak Rate Test" for L2R04 (References v) documenting these failures.

REFUELING OUTAGE	DATE	REPORT NO.
1, L2R01	January, 1987	LER 50-374-87-002, Revision 1
3, L2R03	March, 1990	LER 50-374-90-004, Revision 1
4, L2R04	March, 1992	See Reference v.

LaSalle is conducting an aggressive Corrective Action Plan to directly address and eliminate the local sources of leakage as an alternative to conducting an increased frequency of Type A testing. In lieu of this, continuing to perform a Type A test at a frequency greater than 3 times in 10 years, due solely to Type B and C test failures, is only redundant to the performance of Type B and C tests. There is little or no benefit to be gained from performing a Type A test at an increased frequency.

ATTACHMENT D
EXEMPTION REQUEST FROM APPENDIX J REQUIREMENTS FOR THE TYPE A INCREASED FREQUENCY AS A RESULT OF CONSECUTIVE FAILURES

DISCUSSION (continued)

In support of this are the following:

1. IE Information Notice 85-71 dated August 22, 1985 states in part,

   "... if Type B and C leakage rates constitute an identified contributor to this failure of the "as-found" condition for the Type A test, the general purpose of maintaining a high degree of containment integrity might be better served through an improved maintenance and testing program for containment penetration boundaries and isolation valves. In this situation, the licensee may submit a Corrective Action Plan with an alternative leakage test program proposal as an exemption request for NRC staff review. If this submittal is approved by the NRC staff, the licensee may implement the corrective action and alternative leakage test program in lieu of the required increase in Type A test frequency incurred after the failure of two successive Type A tests."

2. 10 CFR Part 50.12(a) indicates that the Commission may grant exemptions if the exemption will not present undue risk to public health and safety, is consistent with the common defense and security and special circumstances are present.

   An exemption is authorized by law because only the regulation requires the test and the NRC is authorized to grant exemptions from its regulations. No undue risk to public health and safety would result from this exemption because local leakage will be corrected in accordance with the Corrective Action Plan which is described in Attachment B. The common defense and security is not affected by this exemption. Therefore the exemption conditions in 10 CFR 50.12(a)(1) are completely satisfied.

   One of the special circumstances presented in Part 50.12(a)(2)(ii) is:

   "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule."

DISCUSSION (continued)

The underlying purpose of 10 CFR 50 Appendix J, Section III.A.6(b) is to ensure that unacceptable containment leakage is identified and corrected. As described herein, performance of a Type A test at the next refueling outage is not necessary to satisfy this purpose. Therefore 10 CFR 50.12(a)(2)(ii) is satisfied.

3. Iowa Electric Light and Power Company requested an exemption to the same retest requirements of 10 CFR 50 Appendix J in a letter dated April 2, 1990 for the Duane Arnold Energy Center. Their Corrective Action Plan involved the repair, modification, maintenance, and special testing of the Main Steam Isolation and Feedwater Check Valves, which were the significant contributors to the as-found Type A test failures due to excess leakage through these valves. The initial problems experienced with LaSalle's Main Steam Isolation and Feedwater Check Valves were corrected through applicable procedure changes and/or modifications. Attachment F, "Corrective Action Plan for Type C Test Failures Contributing to "As-Found" Type A Test Failures", includes similar successful or planned corrective actions for specific valves that have been major contributors to as-found Type A test failures. The Corrective Action Plan is directed at improving the overall Type B and C test program based upon enhanced trending and engineering evaluation.

4. 10 CFR Part 50.12(a)(2)(iv) identifies as another special circumstance:

> "The exemption would result in a benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption."

Exemption from the requirements to perform a Type A test at the next refueling outage will prevent a radiological exposure of approximately 3 manrem to plant personnel associated with performing a Type A test. This health and safety benefit helps compensate for any perceived decrease in safety that may result from granting the requested exemption.

DISCUSSION (continued)

5. There is also a time and cost benefit to not performing a Type A test more frequently. The test would add two to four days (scheduled for 3.5 days) to the end of the sixth LaSalle Unit Two refueling outage (L2R06). The Engineering Staff, Operating, Maintenance, and Radiation Protection departments would need to devote approximately 1000 man-hours of work to prepare for, perform, and recover from this major test.

6. This request for a one-time exemption from Appendix J also merits consideration, because:

   a. Type B and C tests are better at detecting both penetration seal leakage and containment isolation valve leakage.

   b. The third Type A test failure was non-reportable since the 1.0 L_a criteria was not exceeded. This failure was shown to be almost wholly attributable to the local failure of the reactor water cleanup suction isolation valves. The Reactor Water Cleanup Suction containment isolation valves were replaced with double-disc gate valves prior to startup from L2R04.

LaSalle is addressing the concern of excessive leakage found during Type C Local Leak Rate Testing through an aggressive Corrective Action Plan. This plan includes an update of the corrective actions taken or scheduled to be done for the valves specified in Attachment F, using the guidance given in IE Information Notice 85-71. The Corrective Action Plan in summary addresses:

1. Specific corrective actions completed or planned on specific valves identified as problems thus far.

2. An Alternative Leakage Test Program.

ATTACHMENT D
EXEMPTION REQUEST FROM APPENDIX J REQUIREMENTS FOR THE TYPE A INCREASED FREQUENCY AS A RESULT OF CONSECUTIVE FAILURES

DISCUSSION (continued)

3. Development and implementation of an improved trending program for leakage rate performance, comparing valve type, service, and manufacturer for leakage rate performance comparisons.

4. Recommendation and implementation based on engineering evaluation for test method improvement or repair, and modification or replacement of problem Primary Containment Isolation valves identified in 3. above.

SUMMARY

LaSalle is presently required by 10 CFR 50, Appendix J, Section III.A.6.(b) (and the current Technical Specification Surveillance Requirement 4.6.1.2.b.) to conduct a Type A test on Unit Two during the next refueling outage (L2R06) scheduled to begin in February, 1995. This Type A test will be performed unless an exemption from this requirement (and approval of the associated proposed amendment to LaSalle Unit 1 and 2 Technical Specifications) is granted. The granting of an exemption to Iowa Electric Light and Power Company's Duane Arnold Energy Center (June 29, 1990) has established precedence in this matter.

LaSalle's extensive Corrective Action Plan proposed in lieu of the additional Type A testing includes an augmented Local Leak Rate Test program similar to that of Duane Arnold's. The attached Corrective Action Plan follows the guidance in Information Notice 85-71, assuring that Primary Containment integrity can be maintained without increasing the frequency of Type A tests. The plan not only better implements the safety purpose of the rule, but it also avoids unnecessary occupational radiation exposure to station personnel.

The third Unit 2 Type A test failure remained below the reportable limit of 1.0 L_a even though 0.75 L_a was exceeded. This was overwhelmingly due to a local failure of two containment isolation valves in the same penetration in the Reactor Water Cleanup System. The leakage of these valves accounted for the majority of the Type B and C test penalties which were applied to the "as-found" Type A test results.

ATTACHMENT D
EXEMPTION REQUEST FROM APPENDIX J REQUIREMENTS FOR THE TYPE A INCREASED FREQUENCY AS A RESULT OF CONSECUTIVE FAILURES

SUMMARY (continued)

IE Information Notice 85-71 dated August 22, 1985 addresses alternatives to increased frequency Type A testing such as corrective action plans as well as an increased frequency of Type B and C testing. The replacement of the two troublesome Reactor Water Cleanup Suction containment isolation valves combined with the increased frequency of Type B and C testing is consistent with the Commission's philosophy regarding alternatives to Type A testing.

CONCLUSION

LaSalle Station's Type A test one-time exemption request merits consideration, since the as-found Type A test was less than 1.0 L_a, the current LCO, for L2R04 and the as-found Type A test was less than 0.75 L_a for L2R05. Granting LaSalle a one-time exemption from this testing is consistent with NRC requirements and practice. LaSalle meets the requirements for an exemption from the need for an additional Type A test as previously demonstrated. If the Type A test scheduled to be performed in the LaSalle Unit 2 seventh refueling outage, L2R07, is unacceptable in the as-found condition per 10 CFR 50, Appendix J, then the frequency will be adjusted in accordance with Appendix J as if there were 2 consecutive failures. A Type A test summary and proposed Type A test schedule is included as Attachment G.

It is requested that this exemption request be reviewed and approved for implementation by March 15, 1995. This would serve to avoid delays and additional work during the LaSalle Unit Two sixth refueling outage scheduled in February of 1995.

ATTACHMENT E

EXEMPTION REQUEST FROM APPENDIX J REQUIREMENTS FOR THE TYPE A NORMAL TEST SCHEDULE, SEPARATING THE TYPE A TEST SCHEDULE FROM THE INSERVICE INSPECTION SCHEDULE

BACKGROUND

In order to ensure offsite doses remain below those previously evaluated in the event of a design basis accident, leakage from the primary containment must be limited. To ensure that containment leakage remains within these limits, periodic leakage rate tests are performed. Specifically, 10 CFR 50.54(o) requires primary reactor containments for water cooled power reactors to be subject to the leakage rate testing requirements set forth in Appendix J to 10 CFR 50. LaSalle County Station Units 1 and 2 (LaSalle) Technical Specifications 3/4.6.1.2, "Primary Containment Leakage," provides additional requirements for performing leakage rate testing and specifies the associated limits. The leakage rate testing program for LaSalle is described in Section 6 of the LaSalle Updated Final Safety Analysis Report (UFSAR).

LaSalle is requesting an exemption from paragraph III.A.6.(b) to return to a Type A test frequency of three times in ten years, and thus not perform a second increased frequency Type A test in the upcoming Unit 2 sixth refueling outage, L2R06. The exemption request, discussed in Attachment D of this submittal, necessitates that LaSalle request partial exemption from 10 CFR 50 Appendix J in accordance with 10 CFR 50.12, because L2R06 is the last refueling outage in the first ten year Inservice Inspection (ISI) interval for Unit 2.

The attached request, for a one time exemption from the increased Type A test frequency requirements of Appendix J section III.A.6(b), requires a change to the Technical Specification Surveillance Requirements of specification 3/4.6.1.2, Primary Containment Leakage. Likewise, the permanent exemption to decouple the normal frequency of 3 times in 10 years (40 ± 10 months) from the Inservice Inspection Schedule (ISI) requires a change to the Technical Specification Surveillance Requirements of specification 3/4.6.1.2, Primary Containment Leakage.

ATTACHMENT E
EXEMPTION REQUEST FROM APPENDIX J REQUIREMENTS FOR THE TYPE A NORMAL TEST SCHEDULE, SEPARATING THE TYPE A TEST SCHEDULE FROM THE INSERVICE INSPECTION SCHEDULE

BACKGROUND (continued)

Both exemption requests require a change to Technical Specifications 3/4.6.1.2, because the 4.6.1.2 Surveillance Requirements for Type A tests basically duplicate the requirements of 10 CFR 50 Appendix J with respect to Type A testing and frequency of testing. Therefore, any time a rule change occurs or an exemption is required, a Technical Specification amendment is also required. The Technical Specification change, requested in Attachment A of this submittal, is proposed to relocate Primary Containment leakage specification 3/4.6.1.2 to specification 3/4.6.1.1, "Primary Containment Integrity," as new Surveillance Requirement 4.6.1.1.b.

DISCUSSION

In accordance with 10 CFR 50.12, LaSaile is requesting a partial exemption from the 10 CFR 50 Appendix J, Section III.D.1(a) requirement to perform the third Type A test of each 10-year service period when the plant is shut down for the 10-year plant inservice inspections.

LaSalle also requests the following change to LaSalle Technical Specification 3/4.6.1.2, "Primary Containment Leakage." The change is being proposed in accordance with 10 CFR 50.90 and is reflected in Attachment A. The change to Technical Specification 3/4.6.1.2 and its associated bases is to relocate the requirements regarding primary containment leakage to specification 3/4.6.1.1, "Primary Containment Integrity," as new Surveillance Requirement 4.6.1.1.b. This will delete the specifics regarding Type A testing frequency. The proposed change to Technical Specifications will remove the Technical Specification requirement of performing the third Type A test of each 10-year service period during the shutdown for the 10-year plant inservice inspections and is therefore consistent with the proposed partial exemption.

ATTACHMENT E
EXEMPTION REQUEST FROM APPENDIX J REQUIREMENTS FOR THE TYPE A NORMAL TEST SCHEDULE, SEPARATING THE TYPE A TEST SCHEDULE FROM THE INSERVICE INSPECTION SCHEDULE

DISCUSSION (continued)

The frequency of the Type A test is as follows per 10 CFR 50, Appendix J section III.D.1.(a) Periodic retest schedule:

> "(a) After the preoperational leakage rate tests, a set of three Type A tests shall be performed, at approximately equal intervals during each 10-year service period. The third test of each set shall be conducted when the plant is shutdown for the 10-year plant inservice inspections."

Likewise, LaSalle Unit 1 and 2 Technical Specification Surveillance Requirements 4.6.1.2.a restate the Appendix J requirement:

> "Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10 month intervals during shutdown at P_a, 39.6 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection."

LaSalle proposes to perform the three Type A tests at approximately equal intervals within each 10-year period, with the third test of the set conducted as close as practical to the end of the 10-year period without exceeding the 40 ± 10 month band currently specified in specification 4.6.1.2.a. However, there would be no required connection between the Appendix J 10-year interval and the inservice inspection 10-year interval. LaSalle Unit 2's first 10-year Appendix J interval ends in 1994. LaSalle is therefore currently required to perform a Type A test during the spring 1995 refueling outage.

The 10-year plant inservice inspection (ISI) is the series of inspections performed every 10 years in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. LaSalle performs the ISI volumetric, surface and visual examinations of components and system pressure tests in accordance with 10 CFR 50.55a(g)(4) throughout the 10-year inspection interval. The major portion of this effort is presently being performed every 18 months during the refueling outages. LaSalle's FIRST 10-year ISI program ends in October, 1994. LaSalle is scheduled to complete the first 10-year

DISCUSSION (continued)

program during the spring of 1995 as allowed by Section XI IWA 2400(c). The inservice inspections scheduled during the 1995 refueling outage will complete the first 10-year ISI program. The LaSalle Unit 2 first 10-year Appendix J interval will end in October 1994. The fourth Unit 2 Type A test was performed in the fifth refueling outage in December 1993. Approval of this proposed exemption, in conjunction with the approval of the exemption requested by Attachment D of this submittal, will allow scheduling of the next LaSalle Unit 2 Type A test for the seventh refueling outage to start the three-times-in-ten-years frequency for the second ten year interval. This will maintain approximately equal intervals as required by paragraph III.D of Appendix J.

SUMMARY AND CONCLUSION

Each of these two surveillance tests (i.e., the Type A and the 10-year ISI program) is independent of the other and provides assurances of different plant characteristics. The Type A test assures the required leak-tightness to demonstrate compliance with the guidelines of 10 CFR Part 100. The 10-year ISI program provides assurance of the integrity of the structures, systems, and components in compliance with 10 CFR 50.55a. There is no benefit to be gained by coupling these requirements to the same refueling outage in that elements of the LaSalle ISI program are conducted throughout each 10-year cycle rather than during a refueling outage at the end of the 10-year cycle. Consequently, the subject coupling requirement offers no benefit either to safety or to economical operation of the facility. Accordingly, the subject exemption request meets the underlying purpose of the rule [10 CFR 50.12(a)(2)(ii)].

Consistent with this exemption request, a proposed change to LaSalle Technical Specification 3/4.6.1.2 is also being requested, as described in Attachment A.

ATTACHMENT F
CORRECTIVE ACTION PLAN FOR TYPE C TEST FAILURES
CONTRIBUTING TO "AS-FOUND" TYPE A TEST FAILURES

A. Problems:

1. The LaSalle Unit Two Type B and C test leakage exceeded the $0.60\ L_a$ maximum path leakage, for Type B and C components, during each of the first five refueling outages (1987, 1988, 1990, 1992 and 1993, respectively).

2. During the first, third and fourth refueling outages, the LaSalle Unit Two Type A test failed the as-found condition due to the penalty additions from Type C tests (minimum path leakage). The minimum path leakage would have been acceptable if a Type A test had been required during the second refueling outage. The as-found $0.60\ L_a$ minimum path leakage was not exceeded during either the Unit Two fourth refueling outage, L2R04 or the fifth refueling outage, L2R05.

B. Root Causes of the Problems:

1. Containment Isolation Valve leakage,

2. Inappropriate local leak rate test method for some types of valves, and

3. Weak trending and comparison of valve/penetration performance.

C. General Local Leak Rate Test program improvements:

This is achieved through the development of an ongoing quality Type B and C test program coupled with proper maintenance, root cause investigation, and engineering analysis to determine short and long term corrective actions.

LaSalle County Station has a policy to maintain the primary containment leakage as low as possible. Administrative limits are set for each penetration/component. A component is repaired if its limit is exceeded. A technical evaluation is performed on a case-by-case basis to allow an administrative limit to be exceeded only if the overall containment leakage

is exceptionally low. This policy is established by LaSalle Technical Surveillance LTS-300-5, "Local Leak Rate Test, 0.60 L_a Accountability Program".

The total maximum-pathway leakage rate is reported to the start-up on-site review committee for approval prior to the end of each refueling outage. Problem Identification Reports (PIFs) (investigative reports) are initiated as required to document any Type B and/or C failures and describe corrective actions. Type B and C testing is considered an important and effective program where actions are taken to address problem areas independent of the Type A test.

LaSalle County Station has established motor operated valve preventative maintenance on Containment Isolation valves which includes:

1. The valve operator will be scheduled for refurbishment, if its associated motor operator gear box grease sample is unacceptable.

2. Diagnostic testing of motor operated valve thrust capabilities is underway in accordance with LaSalle responses to Generic Letter 89-10 and it's supplements. Static testing will be completed at least once on all applicable valves by the end of the sixth refueling outage on each LaSalle Unit. Dynamic testing will be completed at least once on all applicable valves by the end of the sixth refueling outage for LaSalle Unit 2 and the seventh refueling outage for LaSalle Unit 1.

D. Specific corrective actions completed or planned on specific valves identified as problems thus far which also impact the Type A test:

1. 2RE024/2RE025, Drywell Equipment Drain Sump and 2RF012/2RF013, Drywell Floor Drain Sump Penetrations:

   The drywell equipment and floor drain sumps penetration isolation valves have repeatedly failed Type C tests. The cause of the failures has been attributed to the introduction of dirt/foreign objects into the drywell equipment and/or floor drain sumps during extended or refueling outages. The dirt or foreign material would subsequently be

pumped through the valves causing this material to settle on valve seats. The result of this caused improper/incomplete valve disc seating and/or irregularities to the valve seat/disc during subsequent valve operation.

An evaluation was performed to resolve the recurring failures. It was determined that the best solution to this problem was to install screens at the bottom of the sumps during extended or refueling outages. This would prevent foreign material from entering into the piping/isolation valve seats, thus mitigating valve leakage. The equipment and floor drain sumps have additionally been placed on a periodic cleaning schedule which will also minimize foreign material buildup/transfer.

Sump screens were installed and cleaned during each Unit's fourth refueling outage (Unit 1, L1R04, in the Spring of 1991, and Unit 2, L2R04, in the Spring of 1992). Since the fourth refueling outage for each unit, the sumps have been cleaned and have had screens/standpipes installed to prevent/minimize the intrusion of foreign material into the system piping. This corrective action has demonstrated an improvement in the valves' leakage performance.

Since the implementation of the corrective action, only one valve Type C test failure has occurred. This failure was during L2R05 when it was determined that 2RF012 had excessive leakage. The failure was attributed to dirty seating surface coupled with inadequate spring tension on the valve actuator. the valve seating surfaces were cleaned and the actuator was rebuilt. The minimum pathway leakage was not affected by this failure, as the redundant containment isolation valve 2RF013 was determined to have satisfactory performance.

Ongoing monitoring of valve leakage performance continues to demonstrate that the resolution is in fact appropriate and does not require any further action, at this time.

ATTACHMENT F
CORRECTIVE ACTION PLAN FOR TYPE C TEST FAILURES
CONTRIBUTING TO "AS-FOUND" TYPE A TEST FAILURES

2. 2G33-F001/2G33-F004, Reactor Water Cleanup Suction Penetration:

Recurrent Type C test failures of the Reactor Water Cleanup (RWCU) Suction Isolation Valves have been a contributor to the "as-found" Type A test failures. The causes of the RWCU Isolation Valve failures vary, but were mainly attributed to dirty or scratched seating surfaces. Table 2 shows the as-found Local Leak Rate Test results for both Unit 1 and Unit 2 Reactor Water Cleanup Suction Isolation Valves, starting with each unit's first refueling outage.

Starting with the Unit 2 third refueling outage (L2R03), LaSalle revised the testing methodology for the RWCU isolation valves. Previous isolation valve testing simultaneously pressurized the volume between the two valves. This method created much difficulty in troubleshooting to determine leakage through each valve and identifying the minimum path leakage which would be added to the as-found Type A test results. The new method tests the valves individually in the normal direction (from inside the containment). This simpler method allows the test engineer to identify the exact leakage through each valve.

An engineering evaluation determined that the single flex wedge gate valve design was inappropriate for the RWCU suction application. The evaluation concluded that a double disc gate valve would greatly improve leakage performance for this particular application. Each valve of the new design has two discs and associated seating surfaces, thus doubling the number of valve leakage barriers.

LaSalle surveyed other utilities to obtain information on the performance of double disc gate valves in the RWCU suction application. Double disc gate valves are in use in the same application at Clinton Power Station with no valve leakage problems. The Hope Creek Nuclear Station has not had any recurring problems with double disc gate valves installed since 1985. The Fitzpatrick plant has recently installed double disc gate valves for the same application due to failures with the single flex wedge gate design.

ATTACHMENT F
CORRECTIVE ACTION PLAN FOR TYPE C TEST FAILURES
CONTRIBUTING TO "AS-FOUND" TYPE A TEST FAILURES

Fitzpatrick experienced Type C test problems after initial installation, believed to be due to improper installation. The valves were retested satisfactorily after maintenance.

The L2R04 primary containment as-found Type A test result of 0.6155% per day exceeded the 0.75 L_a criteria of 0.476% per day due to the Calculated Adjusted of 0.2632% per day and was therefore considered a failure. The maximum allowable containment leak rate (L_a) of 0.635% per day was not exceeded. Note: The Calculated Adjusted leakage rate is found by adding any improvements in local leakage rates due to repair and adjustments to the Type A test results using minimum pathway leakage methodology. The Calculated Adjusted is then used to determine the total as-found containment minimum pathway leakage rate which applies to the as-found Type A test.

The Reactor Water Cleanup Suction penetration contributed 115.7 scfh to the total Calculated Adjusted of 159.765 scfh and accounted for 72.4% of the total. The sum of the Calculated Adjusted and the as-left Type A test, less the RWCU contribution, would have been 44.1 scfh, or 0.0726% per day. This would have resulted in an as-found Type A test of 0.4249% per day, which is below the 0.75 L_a criteria of 0.476% per day.

The Reactor Water Cleanup Suction penetration has historically been a major contributor to and cause of as-found Type A test failures at LaSalle Station. During L2R04 the Reactor Water Cleanup Suction Containment Isolation valves 2G33-F001 and 2G33-F004 were replaced with updated double disc design gate valves per planned modification. They were both successfully Local Leak Rate Tested with zero leakage following replacement. Approximately one month later the 2G33-F001 and 2G33-F004 valves were retested due to motor operator maintenance with Type C test results of 1.5 scfh and 1.44 scfh, respectively.

ATTACHMENT F
CORRECTIVE ACTION PLAN FOR TYPE C TEST FAILURES
CONTRIBUTING TO "AS-FOUND" TYPE A TEST FAILURES

After on full operating cycle, the 2G33-F001 and 2G33-F004 valves were tested during the Unit 2 fifth refueling outage (L2R05). As found Type C tests resulted in outstanding performance with 0.37 scfh for both valves. Following VOTES testing activities, the 2G33-F001 and 2G33-F004 valves were retested in the as-left condition with zero leakage for both valves.

LaSalle Station is confident that the leakage performance of the Reactor Water Cleanup penetration will greatly enhance the Type B and C test/Type A test program.

3. 2IN031, Transverse Incore Probe (TIP) Air Purge Supply Penetration:

The TIP Purge Air Supply Isolation Valve, 2IN031, is a single containment isolation valve. This component failed its Type C test during the Unit 2 first refueling outage and greatly contributed to the "as-found" Type A test failure. The cause of the failure was attributed to a dirty seating surface along with corrosion. The valve was disassembled, cleaned and the internals were replaced.

The 2IN031 valve was originally tested in the "reverse" direction. The test procedure, LTS-100-22, "Drywell Pneumatic System Discharge Isolation Valves Local Leak Rate Tests 1(2)IN017, 1(2)IN031, and 1(2)IN018", was revised to test the component in the "normal" direction which is from inside the containment.

The 2IN031 valve has performed satisfactorily with no leakage since the initial failure during the first refueling outage, demonstrating that the corrective actions taken in the Unit 2 first refueling outage (L2R01) were adequate.

ATTACHMENT F
CORRECTIVE ACTION PLAN FOR TYPE C TEST FAILURES
CONTRIBUTING TO "AS-FOUND" TYPE A TEST FAILURES

4. 2E12-F053A, "A" Residual Heat Removal Shutdown Cooling (RHR SDC) Return Penetration:

The "A" RHR SDC Return Isolation Valve, 2E12-F053A, failed its Type C test during the Unit 2 first refueling outage and greatly contributed to the "as-found" Type A test failure. The cause of the leakage was identified to be a defective valve disc. A new valve disc was installed and the valve seat was refaced.

The repair action was very successful based on subsequent testing performed during the following four refueling outages. Testing conducted since the initial Type C test failure yielded minimal or no leakage.

5. 2E12-F053B, "B" RHR Shutdown Cooling (RHR SDC) Return Penetration:

The "B" RHR SDC Return Isolation Valve, 2E12-F053B, failed its Type C test during the Unit 2 third refueling outage and greatly contributed to the "as-found" Type A test failure. The cause of the excessive leakage was determined to be the failure of the motor operator to properly drive the disc far enough into its seat. The motor operator was refurbished and the valve was retested with satisfactory results. A torque switch that may have contributed to the failure was the sole item replaced during the refurbishment. It was replaced only because it was made of the wrong material (melamine torque switches have been systematically replaced due to a Part 21 notification) and no specific problem with the torque switch was noted in the work request documentation.

The root cause of the valve failing to fully close was undetermined. LaSalle Administrative Procedure LAP-300-31, "Motor Operated Valve Program", has since been revised to require root cause determination for any Motor Operated Valve failure to prevent recurrence. This failure is deemed an isolated occurrence based upon a review of motor operated valve work request history.

The testing and refurbishment conducted in accordance with the Motor Operated Valve preventative maintenance program yielded significant valve leakage performance, as was demonstrated in L2R04 and L2R05. This valve was satisfactorily tested with zero leakage during the Unit 2 fourth and fifth refueling outages (L2R04 and L2R05). Prior to the Type C test failure the 2E12-F053B valve was tested with satisfactory results during the first and second refueling outages.

6. 2HG005A/2HG006A, Unit 2 Hydrogen Recombiner 2HG01A Unit 2 Suppression Pool Discharge Penetration:

The Unit 2 Hydrogen Recombiner 2HG01A Unit 2 Suppression Pool Discharge Isolation Valves, 2HG005A and 2HG006A, have had recurring Type C test failures. During the LaSalle Unit 2 first refueling outage (L2R01), the excessive leakage greatly contributed to the "as-found" Type A test failure. The cause of the Type C test failure was determined to be valve seat irregularities. Valve seating surfaces were lapped and the valves were retested satisfactorily.

Only the 2HG006A isolation valve continued to experience leakage during the second, third, and fifth refueling outages (L2R02, L2R03 and L2R05). In each of these instances the valve was determined to have a dirty and irregular seating surface. Retests were performed satisfactorily following cleaning and lapping of the seating surfaces. Type A test "as-found" results would not have been affected by the single valve Type C test failure since the first refueling outage.

Because of the 2HG005A/2HG006A Isolation valve Type C test failures, all Hydrogen Recombiner Primary Containment Isolation Valves are under an ongoing evaluation to determine what long term corrective action is required to prevent recurring Type C test failures. There are six other isolation valves of the same size and function as 2HG005A and 2HG006A, associated with the Unit 1 and Unit 2 Combustible Gas Control Systems. The only other valve failing in a similar manner is the Unit 2 Hydrogen Recombiner 2HG01A Unit 1 Suppression Pool Discharge Valve, 1HG005B, which failed the Type C test administrative limit.

ATTACHMENT F
CORRECTIVE ACTION PLAN FOR TYPE C TEST FAILURES
CONTRIBUTING TO "AS-FOUND" TYPE A TEST FAILURES

There have been a total of seven Type C test failures in 24 Type C tests for these four penetrations and four of these are associated with the 2HG005A/2HG006A penetration. The three remaining failures were associated with the 1HG005B/1HG006B penetration, which were corrected by cleaning and lapping the valve seat of the 1HG005B. This Unit 1 penetration was satisfactorily Type C tested three times after the initial failure during the Unit 1 reactor recirculation pumps forced maintenance and surveillance outage, 5/28/87- 9/14/87 (hereafter designated as "RR/PP"). However, the penetration has failed the last two refueling outages, L1R05 and L1R06, in the as-found condition.

There are four other penetrations associated with the supply lines to the Hydrogen Recombiners. The Unit 1 Hydrogen Recombiner 1HG01A Unit 1 Drywell Suction Valve, 1HG001A, and the Unit 2 Hydrogen Recombiner 2HG01A Unit 1 Drywell Suction Valve, 1HG001B, each failed their respective Type C tests during the Unit 1 third refueling outage and were repaired by lapping the valve seats. Both of these penetrations were satisfactorily Local Leak Rate Tested during the Unit 1 fourth, fifth and sixth refueling outages. The Unit 1 Type C test failures were shown by test to be due to only one of the two valves in each penetration, so the as-found Type A test was not affected.

The Unit 2 2HG005A and 2HG006A containment isolation valves are to be tested during any non-refueling outage with Unit 2 in cold shutdown for 14 days or longer, unless tested within the previous 6 months. The Unit 1 1HG005B and 1HG006B containment isolation valves are to be tested during any non-refueling outage with Unit 1 in cold shutdown for 14 days or longer, unless tested within the previous 6 months.

The overall performance of the Hydrogen Recombiner containment isolation valves has been satisfactory, based upon test results since the first refueling outage of each LaSalle unit.

ATTACHMENT F
CORRECTIVE ACTION PLAN FOR TYPE C TEST FAILURES
CONTRIBUTING TO "AS-FOUND" TYPE A TEST FAILURES

The Unit 2 components have all performed satisfactorily except for the 2HG006A containment isolation valve which has failed four out of five Type C tests.

Upon further investigation into the 2HG006A valve Type C test failures, it was discovered that after the L2R01 and L2R02 failures the leakage was reduced to an acceptable level as a result of machine lapping the seating surface of the valve. However, after machine lapping during L2R01, the 2HG006A valve failed its Type C test the following L2R02 outage. After machine lapping during L2R02, the 2HG006A valve failed its Type C test the following outage L2R03.

A different approach was taken to reduce the 2HG006A leakage following the failure during L2R03. T' ~ seating surface of the 2HG006A valve was lapped by hand, en blue checked to verify proper seating. Per discussions with members of the Mechanical Maintenance Department, this resulted in a smoother and flatter seating surface. As a result of these actions the 2HG006A valve passed its Type C test during the following L2R04 outage.

During the next Unit 2 refueling outage, l2R05, the 2HG006A valve once again failed its Type C test. An evaluation is being conducted to determine long term corrective action required to resolve the recurring Type C test failures of the 2HG006A and 1HG005B containment isolation valves.

A tabulation of the historical Local Leak Rate Test performance of the above Unit Two 2HG005A and 2HG006A valves is included in Table 1; The performance of the Combustible Gas Control Valves 1(2)HG001A/B, 2A/B, 5A/B, and 6A/B are included in Table 3.

ATTACHMENT F
CORRECTIVE ACTION PLAN FOR TYPE C TEST FAILURES CONTRIBUTING TO "AS-FOUND" TYPE A TEST FAILURES

7. 2VP053A, Unit 2 Primary Containment Chilled Water Supply

An evaluation was performed to address the Type C test failures of the 2VP053A containment isolation valve. All eight Containment Chilled Water Outboard isolation valves of Units 1 and 2 were evaluated. They are 8" 150# single flex wedge gate valves supplied by Anchor/Darling. They are all installed at a 45 degree angle. The 1(2)VP053A/B and the 1(2)VP063A/B are the Containment Chilled Water return and supply lines, respectively.

The overall performance of the Unit 1 and 2 Containment Chilled Water Outboard Isolation Valves has been satisfactory, with the exception of the 2VP053A valve.

Unit 1 has experienced two failures over a six outage period which included 24 individual tests. Only one Type C test failure occurred for the 1VP053A/1VP114A valves (RR/PP). The second Type C test failure occurred fcr the 1VP063A/1VP113A valves (RR/PP). The Unit 1 components have not experienced recurring failures.

The Unit 2 Containment Chilled Water Return Isolation Valve, 2VP053A, has failed its Type C test during all five refueling outages. The valve was repaired after each failure, then successfully Local Leak Rate Tested prior to Unit startup. The repair actions consisted of valve disassembly, cleaning, lapping of the disc and seating surface, frost and/or blue checking and reassembly. Although the valve has been successfully retested after each initial outage failure, the accepted final Type C test results have been substantially higher than all the other Containment Chilled Water Outboard Isolation Valve Type C test results.

The 2VP053A valve appears to continuously degrade during each operating cycle, requiring further attention and investigation. The valve was disassembled and inspected during the Unit 2 fifth refueling outage (L2R05) to determine the root cause of the failures, but was indeterminate. A Problem Identification Form (PIF) has

been generated for root cause evaluation and a work request has been generated to perform more extensive repairs to 2VP053A during L2R06.

A tabulation of the historical Local Leak Rate Test performance of the above chilled water valves is included in Table 4.

E. Corrective Actions for the Local Leak Rate Test Program:

1. Development and implementation of an improved trending program to track penetration and valve leakage rate performance. The assembly of an automated plant-specific database will permit identifying valve type, service application, and manufacturer for leakage rate performance comparison of all Primary Containment Isolation Valves that are required to be Local Leak Rate Tested. This will help to determine any patterns or groups of valves that are exceptionally good performers with minimal or no leakage, or poor performers with several cases of high leakage.

2. Alternative Leakage Test Program:

   LaSalle will perform Type B and C tests on penetrations identified to be susceptible to excessive leakage in accordance with the above stated item 1. These Type B and C tests will be performed during any non-refueling outage with the unit in cold shutdown for 14 days or longer. These tests are in addition to the scope of Type B and C tests performed during refueling outages for as-found maximum path leakage.

3. Recommendation and implementation of test method improvement, or repair, modification, or replacement of problem Primary Containment Isolation valves identified in 1. above, based upon engineering evaluation.

   The test method improvement will include a technical review of Type B and C test procedures associated with poorly performing components to verify the following:

   - Proper type of test for the penetration involved (for example, test direction from the Primary Containment outward versus between a pair of isolation valves in the same penetration, or, from the outside in),
   - Test boundary adequacy, and,
   - Other aspects of test methodology such as clarity and user friendliness.

ATTACHMENT F
TABLE 1
Unit Two Local Leak Rate, Type C, Test Performance Since 1987

VALVE IDENTIFICATION	TYPE C LEAKAGE TEST, "AS-FOUND" MAXIMUM PATH LEAKAGE, SCFH					LEAKAGE CONTRIBUTION TO "AS-FOUND" TYPE A LEAKAGE TEST			
	1987 L2R01	1988/89 L2R02**	1990 L2R03	1992 L2R04	1993 L2R05	L2R01	L2R03	L2R04	L2R05
2RE024 2RE025	29.1	5.9	5.2	5.05	19.4	0.0	N/A*	0.84	N/A*
2RF012 2RF013	0.51	9.19	225.5	1.02	107.66	N/A*	110.32	N/A*	20.04
2G33-F001 2G33-F004	17.7	200.1	GROSS	127.7	0.37	8.67	GROSS	115.7	0.37
2IN031	250.9	0.0	0.0	0.0	0.0	250.88	N/A*	N/A*	N/A*
2E12-F053A	88.2	0.0	2.65	0.0	4.43	0.0	N/A*	N/A*	N/A*
2E12-F053B	0.46	0.0	65.7	0.0	0.0	0.0	63.28	N/A*	N/A*
2HG005A 2HG006A	121.6	31.9	111.0	4.17	21.16	60.57	0.5	0.0	0.2
2HG005B 2HG006B	0.37	0.32	0.37	0.0	0.0	N/A*	N/A*	0.0	N/A*

* No work performed on valves **Non-Type A test outage.

ATTACHMENT F
TABLE 2
Local Leak Rate, Type C, Test Performance for Unit 1 and 2
Reactor Water Cleanup Suction Isolation Valves

LASALLE UNIT	VALVE	FIRST REFUEL	RR/PP*,**	SECOND REFUEL *	THIRD REFUEL	FOURTH REFUEL*	FIFTH REFUEL	SIXTH REFUEL
(VALUES SHOWN ARE AS-FOUND/AS LEFT)								
1	1G33-F001 1G33-F004	1.71/1.72	1.84/1.84	2.7/2.7	GROSS[1]/37.4	23.38[2]/0.0 28.06[3]/0.0	25.6[8]/0.0 0.0/0.0	0.76/0.76 75.83[9]/0.0
2	2G33-F001 2G33-F004	17.7[4]/0.37	N/A	200.1[5]/3.31	GROSS[6]/3.31 GROSS[7]/4.19	115.7[8]/0.0 127.7[8]/0.0	0.37/0.0	(NEXT OUTAGE)

NOTES:

* Non-Type A test outage, L2R04 and L2R05 were Type A test outages due to Type A test failures in L2R01 and L2R03.

** Unit 1 reactor recirculation pumps forced maintenance and surveillance outage, 5/28/87- 9/14/87.

1 Incomplete seating surfaces on 1G33-F004. The valve seating surfaces were cleaned and lapped. The retest was accepted based on low total maximum combined leakage.

2 Incomplete seating surfaces on 1G33-F001. The valve seating surfaces were cleaned and lapped and the valve was repacked.

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ATTACHMENT F
TABLE 2
Local Leak Rate, Type C, Test Performance for Unit 1 and 2
Reactor Water Cleanup Suction Isolation Valves

3 Incomplete seating surfaces on 1G33-F004. The valve seating surfaces were cleaned and lapped.

4 Incomplete seating surfaces on 2G33-F001. The valve seating surfaces were cleaned and lapped. The seating surface was slightly damaged and a crack was found in the disc of 2G33-F004. Lapped the valve seat, machined and lapped the disc to match the seat, and repacked 2G33-F004.

5 Incomplete seating surfaces on 2G33-F001. The valve seating surfaces were cleaned and lapped. Replaced the disc on 2G33-F004 due to crack found the previous outage. Lapped the valve seat, machined and lapped the new disc to match the seat and repacked 2G33-F004.

6 Incomplete seating surfaces on 2G33-F001. The valve seating surfaces were cleaned and lapped.

7 Incomplete seating surfaces and severe packing leak on 2G33-F004. The valve seating surfaces were cleaned and lapped and the valve was repacked.

8 Single flex wedge gate valves removed and replaced with double disc gate valves.

9 Valve disk was found to be cracked and seats warped. This is believed to be caused as a result of initial installation problems where the valve had to be welded/stress relieved several times.

ATTACHMENT F
TABLE 3
Local Leak Rate, Type C, Test Performance for Unit 1 and 2
Combustible Gas Control Isolation Valves

LaSalle Unit 1 (Values shown are as found/as left)							
Valves	L1R01	RR/PP*	L1R02	L1R03	L1R04	L1R05	L1R06
1HG001A/ 1HG002A	1.76/0.89	0.56/0.56	0.6/0.6	125.9/0.0	3.8/0.38	0.0	0.0
1HG005A/ 1HG006A	1.15/1.15	1.39/1.39	1.75/1.75	0.186/0.186	1.58/1.58	2.9/2.32	1.67/0.37
1HG001B/ 1HG002B	1.72/1.72	1.21/1.21	3.24/3.24	19.3/0.0	0.56/0.0	0.46/0.46	0.37/0.0
1HG005B/ 1HG006B	4.8/4.8	9.86/1.61	1.29/1.29	2.96/1.67	0.28/0.65	13.33/3.52	27.75/4.16

** Unit 1 reactor recirculation pumps forced maintenance and surveillance outage, 5/28/87- 9/14/87.

ATTACHMENT F
TABLE 3
Local Leak Rate, Type C, Test Performance for Unit 1 and 2
Combustible Gas Control Isolation Valves

LaSalle Unit 2 (Values shown are as found/as left)					
Valves	L2R01	L2R02	L2R03	L2R04	L2R05
2HG001A/ 2HG002A	0.37/0.37	0.28/0.28	0.52/0.52	1.07/1.07	0.83/1.11
2HG005A/ 2HG006A	121.6/0.47	31.9/2.61	111.0/1.0	4.17/6.51	21.16/6.01
2HG001B/ 2HG002B	1.3/1.3	0.28/0.28	0.0/0.0	0.56/0.56	0.74/0.37
2HG005B/ 2HG006B	0.37/0.37	0.32/0.32	0.37/0.0	0.0/0.0	0.0/0.0

ATTACHMENT F
TABLE 4
Local Leak Rate, Type C, Test Performance for Unit 1 and 2
Primary Containment Chilled Water (VP) Isolation Valve Performance

LaSalle Unit 1 (Values shown are as found/as left)							
Valves	L1R01	RR/PP*	L1R02	L1R03	L1R04	L1R05	L1R06
1VP053A/ 1VP114A	1.15/1.15	GROSS/1.25	0.05/0.05	0.78/0.78	0.0/0.37	0.0/0.37	0.37/0.0
1VP053B/ 1VP114B	1.38/1.38	2.39/2.39	1.67/1.67	1.3/1.3	1.71/1.71	1.71/1.71	1.39/1.39
1VP063A/ 1VP113A	3.57/3.57	25.3/1.21	1.07/1.07	1.49/1.49	0.0/0.0	0.0/0.0	0.37/0.37
1VP063B/ 1VP113B	0.05/0.05	1.1/1.1	1.86/1.2	2.03/2.03	1.35/1.35	1.35/1.3	1.58/1.57

** Unit 1 reactor recirculation pumps forced maintenance and surveillance outage, 5/28/87- 9/14/87.

ATTACHMENT F
TABLE 4
Local Leak Rate, Type C, Test Performance for Unit 1 and 2
Primary Containment Chilled Water (VP) Isolation Valve Performance

LaSalle Unit 2 (Values shown are as found/as left)					
Valves	L2R01	L2R02	L2R03	L2R04	L2R05
2VP053A/ 2VP114A	9.38/3.4	32.5/7.4	GROSS/16.66	24.5/5.6	0.83/12.16
2VP053B/ 2VP114B	0.47/0.47	0.55/0.55	0.0/0.46	0.37/0.37	0.0/0.0
2VP063A/ 2VP113A	1.04/1.04	0.0/0.0	0.0/0.0	1.39/1.39	0.0/1.38
2VP063B/ 2VP113B	1.85/1.85	0.83/0.83	0.37/9.34	0.46/0.37	0.69/0.69

ATTACHMENT G

Previous and Proposed Type A Test Schedule for Unit 2

DATE	TYPE A TEST DESCRIPTION	CONDUCTED	PASS/FAIL
JANUARY 1987	L2R01	YES	FAIL
OCTOBER 1988	L2R02	NO[1]	----
MARCH 1990	L2R03	YES	FAIL
MARCH 1992	L2R04	YES	FAIL[3]
DECEMBER 1993	L2R05	YES	PASS
FEBRUARY - APRIL 1995	L2R06	NO[2]	----
NOVEMBER 1996 - JANUARY 1997	L2R07	YES[2]	----

NOTE 1: Minimum path leakage for determining leakage penalties for an as-found Type A test would have been acceptable.

NOTE 2: Proposed Type A test Schedule

NOTE 3: The L2R04 primary containment as-found Type A test result of 0.6155% per day exceeded the 0.75 L_a criteria of 0.476% per day due to the Calculated Adjusted of 0.2632% per day and was therefore considered a failure. The maximum allowable containment leak rate (L_a) of 0.635% per day was not exceeded. The Reactor Water Cleanup Suction penetration contributed 115.7 scfh to the total Calculated Adjusted of 159.765 scfh and accounted for 72.4% of the total. The sum of the Calculated Adjusted and the as-left Type A test, less the RWCU contribution, would have been 44.1 scfh, or 0.0726% per day. This would have resulted in a satisfactory as-found Type A test of 0.4249% per day, which is below the 0.75 L_a criteria of 0.476% per day.

ATTACHMENT H

ENVIRONMENTAL ASSESSMENT STATEMENT APPLICABILITY REVIEW

Commonwealth Edison has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed changes meet the criteria for a categorical exclusion as provided under 10 CFR 51.22 (c)(9). This conclusion has been determined because the changes requested do not pose significant hazards consideration or do not involve a significant increase in the amounts, and no significant changes in the types, of any effluents that may be released offsite. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure.