TECHNICAL SPECIFICATIONS
LIST OF TABLES (Continued)

15. PRIMARY CONTAINMENT INTEGRITY

Primary Containment Integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

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 de-activated automatic valve secured in its closed position. (These valves may be opened to perform necessary operational activities.)

b. At least one door in each airlock is closed and sealed.

c. All blind flanges and manways are closed.

16. SECONDARY CONTAINMENT INTEGRITY

Secondary containment integrity means that the reactor building is intact and the following conditions are met:

a. At least one door in each access opening is closed.

b. The standby gas treatment system is OPERABLE.

c. 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 de-activated automatic valve or damper secured in its closed position. (These valves/dampers may be opened to perform necessary operational activities.)

17. OPERATING CYCLE

For the purpose of designating surveillance test frequencies, the duration of an operating cycle shall not exceed 18 months. Surveillance tests designated "once per operating cycle" shall be conducted at least once per operating cycle except that surveillance tests performed during an outage which commences before expiration of the operating cycle may be considered timely.

18. REFUELING OUTAGE

Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the unit after that refueling. For surveillance test purposes, tests are to be performed at least once during a refueling outage as indicated in these technical specifications. In cases where the surveillance test frequency is required to be performed more than once during a refueling outage (e.g., once per week during refueling), the surveillance test shall not be performed less frequently than required by these technical specifications.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
c) At least one pretreatment steam air ejector offgas system radiation monitor shall be operable during reactor power operation. The monitors shall be set to initiate an alarm if the monitor exceeds a trip setting equivalent to 1.0 Ci/sec of noble gases after 30 minutes delay in the offgas holdup line. In the event the noble gas flow in the air ejector offgas exceeds the equivalent of 1.0 Ci/sec after 30 minutes delay in the offgas holdup line, restore the rate to less than this limit within 72 hours or be in at least hot standby within the next 12 hours.	
d) In the event no pre- treatment monitor is operable, gases from the steam air ejector offgas system may be released for up to 30 days provided (1) the charcoal bed of the offgas system is not bypassed, (2) Grab samples are collected and analyzed weekly, and (3) the offgas stack noble gas activity monitor is operable, or at least 1 post-treatment monitor is operable. Otherwise, be in at least HOT STANDBY within the following 24 hours.	
2. Reactor Building Isolation and Standby Gas Treatment System The limiting conditions for operation are given in Section 3.7.	2. Reactor Building Isolation and Standby Gas Treatement System Instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.2-D. System logic shall be functionally tested as indicated in Table 4.2-D.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
specified in Specification 3.5.G.3.b(1) and (2), below. A diesel generator required for operation of at least one of these pumps shall be OPERABLE.	
(1) With one of the two pumps inoperable, restore the inoperable pump to OPERABLE status within four hours or suspend all operations with a potential for draining the reactor vessel.	
(2) With both pumps inoperable, suspend all operations with a potential for draining the reactor vessel.	
4. During a refueling outage, CORE ALTERATIONS may continue with the suppression pool volume below the minimum values specified in section 3.7 provided all of the following conditions are met:	
(a) The reactor head is removed, the cavity is flooded, the spent fuel pool gates are removed and spent fuel pool water level is maintained within the limits of Specification 3.9.C.	
(b) At least one Core Spray pump capable of transferring water to the vessel is OPERABLE with suction aligned to the condensate storage tank(s).	
(c) The condensate storage tanks contain at least 75,000 gallons of water which is available to the core spray subsystem. Condensate storage tank(s) level shall be recorded at least every 12 hours.	
(d) No work is being performed which has the potential for draining the reactor vessel.	

B & C. Containment Spray and RHR Service Water

The containment spray subsystem for DAEC consists of 2 loops each with 2 LPCI pumps and 2 RHR service water pumps per loop. The water pumped through the RHR heat exchangers may be diverted to two spray headers in the drywell and one above the suppression pool. The design of these systems is predicted upon use of 1 LPCI, and 2 RHR service water pumps for heat removal after a design basis event. Thus, there are ample spares for margin above the design conditions. Loss of margin should be avoided and the equipment maintained in a state of operability so a 30-day out-of-service time is chosen for this equipment. If one loop is out-of-service, or one pump in each loop is out-of-service, reactor operation is permitted for seven days. The surveillance requirements provide adequate assurance that the Containment Spray subsystem and RHRSW system will be operable when required.

Analyses were performed to determine the minimum required flow rate of the RHR Service Water pumps in order to meet the design basis case (Reference 4) and the NUREG-0783 requirements (Reference 5). (See Section 3.7 Bases for a discussion of the NUREG requirements.) The results of these analyses justify reducing the required flowrate to 2040 gpm per pump, a 15% reduction in the original 2400 gpm per pump requirement.

LIMITING CONDITIONS FOR OPERATION

3.7 PLANT CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment Integrity

1. PRIMARY CONTAINMENT INTEGRITY shall be maintained at all times when the reactor is critical or when the temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 Mw(t). Compliance with Subsection 3.7.B.2 satisfies the requirement to maintain PRIMARY CONTAINMENT INTEGRITY.

2. 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 REQUIREMENT

4.7 PLANT CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment system integrity.

Objective:

To verify the integrity of the primary and secondary containments.

Specification:

A. Primary Containment Integrity

1. PRIMARY CONTAINMENT INTEGRITY shall be demonstrated as follows:

a. Type A Test

Primary Reactor Containment Integrated Leakage Rate Test

1) The interior surfaces of the drywell and torus shall be visually inspected each operating cycle for evidence of deterioration. In addition, the external surfaces of the torus below the water level shall be inspected on a routine basis for evidence of torus corrosion or leakage.

Except for the initial Type A test, all Type A tests shall be performed without any preliminary leak detection surveys and leak repairs immediately prior to the test.

If a Type A test is completed but the acceptance criteria of Specification 4.7.A.1.a.(8) is not satisfied and repairs are necessary, the Type A test need not be repeated provided locally measured leakage reductions, achieved by repairs, reduce the containment's overall measured leakage rate sufficiently to meet the acceptance criteria.

2) Closure of containment isolation valves for the Type A test shall be accomplished by normal mode of actuation and without any preliminary exercising or adjustments.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT

3) The containment test pressure shall be allowed to stabilize for a period of about 4 hours prior to the start of a leakage rate test.

4) The reactor coolant pressure boundary shall be vented to the containment atmosphere prior to the test and remain open during the test.

5) Test methods are to comply with ANSI N45.4-1972.

6) The accuracy of the Type A test shall be verified by a supplemental test. An acceptable method is described in Appendix C of ANSI N45.4-1972.

| 7) Periodic Leakage Rate Tests

Periodic leakage rate tests shall be performed at or above the peak pressure (Pa) of 43 psig.

| 8) Acceptance Criteria

The maximum allowable leakage rate (Lam) is 0.75 La, where La is defined as the design basis accident leakage rate of 2.0 weight percent of contained air per 24 hours at 43 psig.

| 9) Additional Requirements

If any periodic Type A test fails to meet the applicable acceptance criteria the test schedule applicable to subsequent Type A tests will be reviewed and approved by the Commission.

If two consecutive periodic Type A tests fail to meet the acceptance criteria of 4.7.A.1.a.(8) a Type A test shall be performed each operating cycle, or approximately every 18 months, whichever occurs first, until two consecutive Type A tests meet the subject acceptance criteria after which time the retest schedule of 4.7.A.1.d may be resumed.

| b. Type B Tests

Type B tests refer to penetrations with gasketed seals, expansion bellows or other type of resilient seals.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT

| 1) Test Pressure

All Type B tests shall be performed by local pneumatic pressurization of the containment penetrations, either individually or in groups, at a pressure not less than Pa.

| 2) Acceptance Criteria

The combined leakage rate of all penetrations subject to Type B and C tests shall be less than 0.60 La.

| c. Type C Tests

1) Type C tests shall be performed on containment isolation valves. Each valve to be tested shall be closed by normal operation and without any preliminary exercising or adjustments.

2) Acceptance criteria - The combined leakage rate for all penetrations subject to Type B and C tests shall be less than 0.60 La.

3) The leakage from any one main steam isolation valve shall not exceed 11.5 scf/hr at an initial test pressure of 24 psig.

4) The leakage rate from any containment isolation valve whose seating surface remains water covered post-LOCA, and which is hydrostatically Type C tested, shall be included in the Type C test total.

| d. Periodic Retest Schedule

| 1) Type A Test

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. (These intervals may be extended up to eight months if necessary to coincide with refueling outages.) The third test of each set shall be conducted when the plant is shut down for the 10-year plant in-service inspections.

The performance of Type A tests shall be limited to periods when the plant facility is

LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENT

nonoperational and secured in the shutdown condition under administrative control and in accordance with the plant safety procedures.

| 2) Type B Tests

a) Penetrations and seals of this type (except air locks) shall be leak tested at greater than or equal to 43 psig (P_a) during each reactor shutdown for major refueling or other convenient interval but in no case at intervals greater than two years.

b) The personnel airlock shall be pressurized to greater than or equal to 43 psig (P_a) and leak tested at least once every six (6) months. This test interval may be extended to the next refueling outage (up to a maximum interval between P_a tests of 24 months) provided there have been no airlock openings since the last successful test at P_a.

c) Within three (3) days after securing the airlock when containment integrity is required, the airlock gaskets shall be leak tested at a pressure of P_a.

| 3) Type C Tests

Type C tests shall be performed during each reactor shutdown for major refueling or other convenient interval but in no case at intervals greater than two years.

| 4) Additional Periodic Tests

Additional purge system isolation valve leakage integrity testing shall be performed at least once every three months in order to detect excessive leakage of the purge isolation valve resilient seats. The purge system isolation valves will be tested in three groups, by penetration: drywell purge exhaust group (CV-4302 and CV-4303), torus purge exhaust group (CV-4300 and CV-4301), and drywell/torus purge supply group (CV-4307, CV-4308 and CV-4306).

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENT

e. Seal Replacement and Mechanical Limiter

The T-ring inflatable seals for purge isolation valves CV-4300, CV-4301, CV-4302, CV-4303, CV-4306, CV-4307 and CV-4308 shall be replaced at intervals not to exceed four years.

During Type C testing, it shall be verified that the mechanical modification which limits the maximum opening angle for purge isolation valves CV-4300, CV-4301, CV-4302, CV-4303, CV-4306, CV-4307 and CV-4308 is intact.

The baseline for this requirement shall be established during the Cycle 6/7 refuel outage.

f. Containment Modification

Any major modification, replacement of a component which is part of the primary reactor containment boundary, or resealing a seal-welded door, performed after the preoperational leakage rate test shall be followed by either a Type A, Type B, or Type C test, as applicable, for the area affected by the modification. The measured leakage from this test shall be included in this test report. The acceptance criteria as appropriate, shall be met. Minor modifications, replacements, or resealing of seal-welded doors, performed directly prior to the conduct of a scheduled Type A test do not require a separate test.

g. Reporting

Periodic tests shall be the subject of a summary technical report submitted to the Commission approximately 3 months after the conduct of each test. The report will be titled "Reactor Containment Integrated Leakage Rate Test."

The results of the periodic testing performed to satisfy the requirements of 4.7.A.1.d.(4) shall be reported with the summary technical report prepared to provide the results of the testing performed in accordance with Section 4.7.A.1.d.(3).

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT

The report shall include a schematic arrangement or description of the leakage rate measurement system, the instrumentation used, the supplemental test method, the test program selected, and all subsequent periodic tests. The report shall contain an analysis and interpretation of the leakage rate test data for the Type A test results to the extent necessary to demonstrate the acceptability of the containment's leakage rate in meeting the acceptance criteria.

For each periodic test, leakage test results from Type A, B, and C tests shall be reported. The report shall contain an analysis and interpretation of the Type A test results and a summary analysis of periodic Type B and Type C tests that were performed since the last Type A test. Leakage test results from Type A, B, and C tests that failed to meet the acceptance criteria shall be reported in a separate accompanying summary report. The Type A test summary report shall include an analysis and interpretation of the test data, the least-squares fit analysis of the test data, the instrumentation error analysis, and the structural conditions of the containment or components, if any, which contributed to the failure in meeting the acceptance criteria. Results and analyses of the supplemental verification test employed to demonstrate the validity of the leakage rate test measurements shall also be included.

The Type B and C tests summary report shall include an analysis and interpretation of the data and the condition of the components which contributed to any failure in meeting the acceptance criteria.

LIMITING CONDITIONS FOR OPERATION

B. Primary Containment Power Operated Isolation Valves

1. During reactor power operating conditions, all primary containment isolation valves and all instrument line flow check valves shall be OPERABLE except as specified in 3.7.B.2.

2. With one or more of the primary containment isolation valves inoperable, 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 isolation valve secured in the isolated position,* or

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

SURVEILLANCE REQUIREMENT

B. Primary Containment Power Operated Isolation Valves

1. The primary containment isolation valves surveillance shall be performed as follows:

a. At least once per operating cycle the OPERABLE isolation valves# that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.

b. At least once per quarter:

1) All normally open power operated isolation valves## shall be fully closed and reopened.

2) With the reactor power less than 75%, trip main steam isolation valves individually and verify closure time.

c. At least once per operating cycle the operability of the reactor coolant system instrument line flow check valves shall be verified.

#Due to operation limitations, the Main Steam Line Isolation Valves are exempt from Subsection 4.7.B.1.a.

##Due to plant operational limitations, the Well Cooling Water Supply/Return Valves, Reactor Building Closed Cooling Water Supply/Return Valves and the Containment Compressor Discharge and Suction valves are exempt from the requirements of subsection 4.7.B.1.b.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT

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

3. If Specifications 3.7.B.1, and 3.7.B.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in at least HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the following 24 hours.

4. Purging

a. Containment vent/purge valves (CV-4300, CV-4301, CV-4302, CV-4303, CV-4306, CV-4307, CV-4308, CV-4309, and CV-4310) may not be opened so as to create a flow path from the primary containment while PRIMARY CONTAINMENT INTEGRITY is required except for inerting, de-inerting, vent/purge valve testing, or pressure control.

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

LIMITING CONDITIONS FOR OPERATION

C. Drywell Average Air Temperature

1. Drywell average air temperature shall not exceed 135°F whenever the reactor is critical or when the reactor temperature is above 212°F and fuel is in the reactor vessel.

2. With the drywell average air temperature greater than 135°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENT

C. Drywell Average Air Temperature

1. Verify drywell average air temperature is ≤ 135°F at least once/24 hours.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
D. Pressure Suppression Chamber - Reactor Building Vacuum Breakers	D. Pressure Suppression Chamber - Reactor Building Vacuum Breakers
1. Each Pressure Suppression Chamber - Reactor Building vacuum breaker assembly consisting of a vacuum breaker valve and a butterfly isolation valve shall be OPERABLE and closed at all times when PRIMARY CONTAINMENT INTEGRITY is required.	1. Each Pressure Suppression Chamber - Reactor Building vacuum breaker assembly shall be verified olosed at least once per 7 days.
2. If one valve of a Pressure Suppression Chamber - Reactor Building vacuum breaker assembly is inoperable for opening but known to be closed, restore the inoperable vacuum breaker assembly valve to OPERABLE 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.	2. Once/quarter, cycle each vacuum breaker assembly valve through at least one complete cycle of full travel. Verify each position indicator OPERABLE by observing expected valve indication during the cycling test.
3. If one valve of a Pressure Suppression Chamber - Reactor Building vacuum breaker assembly is open, within 2 hours verify the other vacuum breaker assembly valve in that line to be closed. Restore the open vacuum breaker assembly valve to the closed position within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.	3. Once/quarter, demonstrate that the opening setpoint of each vacuum breaker is the equivalent of ≤ 0.5 psid.
4. If the position indication of any Pressure Suppression Chamber - Reactor Building vacuum breaker assembly valve is inoperable, restore it to operable status within 14 days or verify the affected vacuum breaker assembly valve to be closed at least once/24 hours by visual inspection. Otherwise declare the vacuum breaker assembly valve inoperable or be in at least HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the following 24 hours.	

LIMITING CONDITIONS FOR OPERATION

E. Drywell - Pressure Suppression Chamber Vacuum Breakers

1. Each drywell - pressure suppression chamber vacuum breaker shall be OPERABLE and closed at all times when PRIMARY CONTAINMENT INTEGRITY is required.

2. If one of the drywell-pressure suppression chamber vacuum breakers is inoperable for opening but known to be closed, restore the inoperable vacuum breaker to OPERABLE 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.

3. With one or more drywell - pressure suppression chamber vacuum breakers open, close the open vacuum breaker(s) within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With one of the position indicators of any drywell-pressure suppression chamber vacuum breaker inoperable:

a. Verify the other position indicator OPERABLE within 2 hours and at least once per 14 days thereafter or,

b. Verify that the vacuum breaker is closed by determining that the total drywell to suppression pool bypass area is less than 0.2 ft^2 within 24 hours and at least once per 14 days thereafter.

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

SURVEILLANCE REQUIREMENT

E. Drywell - Pressure Suppression Chamber Vacuum Breakers

1. Each drywell-pressure suppression chamber vacuum breaker shall be verified closed at least once per 7 days.

2. At least once/month, cycle each drywell-pressure suppression chamber vacuum breaker through at least one cycle of full travel. Verify each position indicator OPERABLE by observing expected valve movement during the cycling test.

3. Once/cycle, each drywell-pressure suppression chamber vacuum breaker shall be visually inspected to insure proper maintenance and operation.

4. A leak test of the drywell to suppression chamber structure shall be conducted once per operating cycle.

LIMITING CONDITIONS FOR OPERATION

F. Main Steam Isolation Valve Leakage Control System (MSIV-LCS)

1. The MSIV-LCS shall be OPERABLE whenever the reactor is critical or when the reactor temperature is above 212°F and fuel is in the reactor vessel, except as specified in 3.7.F.2 below.

2. From and after the date that one MSIV-LCS subsystem or one blower is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding thirty days provided that during such thirty days all active components of the other MSIV-LCS subsystems are verified to be OPERABLE.

3. If the requirements of specification 3.7.F cannot be met, the reactor shall be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENT

F. Main Steam Isolation Valve Leakage Control System

1. MSIV-LCS Testing

	Item	Frequency
a.	Simulated Actuation Test	Once/ Operating Cycle
b.	Blower Operability	Once/Month
c.	Motor-operated Valve Operability	Once/3 Months
d.	Heater Operability	Once/Month
e.	Blower Capacity	Once/ Operating Cycle

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
G. Suppression Pool Level and Temperature	G. Suppression Pool Level and Temperature
At any time that the nuclear system is pressurized above atmospheric, the suppression pool shall be OPERABLE with:	
1. Suppression Pool Level	1. Suppression Pool Level
a. The volume of the suppression pool shall be between 61,500 ft^3 (60%) and 58,900 ft^3 (40%).	a. The suppression pool water level shall be verified to be within the limits at least once per day.
b. If the suppression pool water level is not within the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.	
2. Suppression Pool Temperature	2. Suppression Pool Temperature
a. The suppression pool average water temperature shall be ≤ 95°F during normal power operation.	a. The suppression pool average water temperature shall be verified to be within the applicable limits at least once per day, except:
b. If the suppression pool average water temperature is > 95°F but < 110°F during normal power operation and not performing testing which adds heat to the suppression pool, verify suppression pool average water temperature is < 110°F once per hour and restore suppression pool average water temperature to ≤ 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the following 24 hours.	b. The suppression pool average water temperature shall be verified to be ≤ 105°F at least once every 5 minutes during testing which adds heat to the pool.
c. If the suppression pool average water temperature is > 105°F during testing which adds heat to the suppression pool, immediately suspend all testing which adds heat to the suppression pool, verify suppression pool average water temperature is < 110°F once per hour, and restore suppression pool average temperature to ≤ 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the following 24 hours.	c. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 200°F or more, an external visual inspection of the suppression chamber shall be conducted before resuming power operation.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
d. If the suppression pool average water temperature is ≥ 110°F, the reactor shall be scrammed.	d. A visual inspection of the suppression chamber interior, including water line regions, shall be made once per operating cycle.
e. If the suppression pool average water temperature is ≥ 120°F, depressurize the reactor to less than 200 psig within 12 hours.	

LIMITING CONDITIONS FOR OPERATION

H. Containment Atmosphere Dilution

1. Whenever the reactor is in power operation and the primary containment is required to be inerted per TS section 3.7.I.1, the Post-LOCA Containment Atmosphere Dilution System must be OPERABLE and capable of supplying nitrogen to the containment for atmosphere dilution if required by post-LOCA conditions. If this specification cannot be met, the system must be restored to an operable condition within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. Whenever the reactor is in power operation, the post-LOCA Containment Atmosphere Dilution System shall contain a minimum of 50,000 scf of N_2 as determined by pressure and temperature measurements. If this specification cannot be met, the minimum volume will be restored within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. The limiting conditions for operation for the CAD system H_2 and O_2 analyzers serving the drywell and the suppression chamber are specified in Table 3.2-H.

SURVEILLANCE REQUIREMENT

H. Containment Atmosphere Dilution

1. The post-LOCA containment atmosphere dilution system shall be functionally tested annually.

2. The volume in the N_2 storage bank shall be recorded weekly.

3. Surveillance requirements for the CAD system H_2 and O_2 analyzers are specified in Table 4.2-H. The atmosphere analyzing system shall be functionally tested annually in conjunction with specification 4.7.H.1.

LIMITING CONDITIONS FOR OPERATION

I. Oxygen Concentration

1. The drywell and suppression chamber atmosphere oxygen concentration shall be less than 4% by volume during REACTOR POWER OPERATION, during the time period:

a. from 24 hours after placing the reactor mode switch in RUN following startup, to

b. 24 hours prior to taking the reactor mode switch out of RUN prior to reactor shutdown.

2. If the drywell or suppression chamber atmospheric oxygen concentration is not within the limit, restore the oxygen concentration to within the limit within 24 hours or be in at least STARTUP/HOT STANDBY within the next 8 hours.

SURVEILLANCE REQUIREMENT

I. Oxygen Concentration

1. The drywell and suppression chamber oxygen concentration shall be verified to be within the limit within 24 hours after placing the reactor mode switch in RUN and at least once every 7 days thereafter.

LIMITING CONDITIONS FOR OPERATION

J. Secondary Containment

1. Secondary containment integrity shall be maintained during all modes of plant operation except when all of the following conditions are met.

a. The reactor is subcritical and Specification 3.3.A is met.

b. The reactor water temperature is below 212°F and the reactor coolant system is vented.

c. No activity is being performed which can reduce the shutdown margin below that specified in Specification 3.3.A.

d. The fuel cask or irradiated fuel is not being moved in the reactor building.

2. If Specification 3.7.J.1 cannot be met:

a. Suspend reactor building fuel cask and irradiated fuel movement, and

b. Restore secondary containment integrity within one hour; or,

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

SURVEILLANCE REQUIREMENT

J. Secondary Containment

1. Secondary containment surveillance shall be performed as indicated below:

a. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind conditions (< 15 mph) with a filter train flow rate of not more than 4,000 cfm, shall be demonstrated at each refueling outage prior to refueling.

LIMITING CONDITIONS FOR OPERATION

K. Secondary Containment Automatic Isolation Dampers

1. All secondary containment automatic isolation valves/dampers shall be OPERABLE at all times when SECONDARY CONTAINMENT INTEGRITY is required.

2. With one or more of the secondary containment automatic isolation valves/dampers inoperable, maintain at least one isolation valve/damper OPERABLE in each affected penetration that is open and within 8 hours either:

   a. Restore the inoperable valve/damper to OPERABLE status, or

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

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

3. If the above specifications cannot be met, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours and suspend reactor building fuel cask and irradiated fuel movement.

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

SURVEILLANCE REQUIREMENT

K. Secondary Containment Automatic Isolation Dampers

1. At least once per operating cycle, the OPERABLE isolation dampers that are power operated and automatically initiated shall be tested for simulated automatic initiation.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
L. Standby Gas Treatment System	**L. Standby Gas Treatment System**
1. Except as specified in Specifications 3.7.L.3 and 3.9.D, both trains of the standby gas treatment system shall be OPERABLE at all times when SECONDARY CONTAINMENT INTEGRITY is required.	1.a Annually it shall be demonstrated that pressure drop across the combined high efficiency and charcoal filters is less than 11 inches or water in the flow range of 3600 to 4000 cfm.
	b. Annually demonstrate that the inlet heaters on each train are capable of an output of at least 22 Kw.
	c. After each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing, demonstrate that air distribution is uniform within 20% of averaged flow per unit across HEPA filters.
	d. Once per operating cycle automatic initiation of each branch of the standby gas treatment system shall be demonstrated.
	e. Manual operability of the bypass system for filter cooling shall be demonstrated annually.
	f. System drains shall be inspected quarterly for adequate water level in loop seals.
	g. Each bed will be visually inspected in conjunction with the sampling in Specification 3.7.L.2.b to assure that no flow blockage has occurred.
2.a The results of the inplace cold DOP and halogenated hydrocarbon tests in the flow range of 3600-4000 cfm on HEPA filters and charcoal adsorber banks shall show ≥ 99.9% DOP removal and ≥ 99.9% halogenated hydrocarbon removal.	2.a The tests and sample analysis of Specification 3.7.L.2 shall be performed initially and then annually for standby service or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.
b. The results of laboratory carbon sample analysis shall show < 1.0% penetration of radioactive methyl iodide at 70% R.H., 150°F, 40 ± 4 FPM face velocity with an inlet concentration of 0.5 to 1.5 mg/m^3 inlet concentration methyl iodide.	b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.
c. Fans shall be shown to be capable of operation from 1800 cfm to the flow range of 3600-4000 cfm.	c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
	d. Each circuit shall be operated with the heaters on at least 10 hours every month.
3. With one train of SGTS inoperable, operation or fuel handling may continue provided the remaining SGTS is verified to be OPERABLE; restore the inoperable SGTS train to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours and suspend reactor building fuel cask and irradiated fuel movement.	

LIMITING CONDITIONS FOR OPERATION

M. Mechanical Vacuum Pump

1. The mechanical vacuum pump shall be capable of being isolated and secured on a signal of high radioactivity in the steam lines whenever the main steam isolation valves are open.

2. During mechanical vacuum pump operation the release rate of gross activity except for halogens and particulates with half lives longer than eight days shall not exceed 1 curie/sec.

3. If the requirements of 3.7.M.1 or 3.7.M.2 are not met, the Mechanical Vacuum Pump suction valves shall be closed.

SURVEILLANCE REQUIREMENT

M. Mechanical Vacuum Pump

1. Surveillance requirements are given in Table 4.2-D.

3.7.A & 4.7.A BASES:

Primary Containment Integrity

The integrity of the primary containment and operation of the core standby cooling system in combination, limit the offsite doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time, thus greatly reducing the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth such that a rod drop would not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep offsite doses well below 10 CFR 100 limits.

In the event primary containment is inoperable, primary containment must be restored within 1 hour. The 1 hour time provides a period of time commensurate with the importance of maintaining primary containment and also ensures that the probability of an accident requiring primary containment during this time period is minimal.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response corresponding to the design basis loss-of-coolant accident. The peak drywell pressure would be about 43 psig which would rapidly reduce to 27 psig within 30 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to about 25 psig within 30 seconds, equalizes with drywell pressure shortly thereafter and then rapidly decays with the drywell pressure decay, (Reference 1).*

The design pressure of the drywell and suppression chamber is 56 psig, (Reference 2). The design basis accident leakage rate is 2.0%/day at a pressure of 43 psig. As pointed out above, the drywell and suppression chamber pressure following an accident would equalize fairly rapidly. Based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The design basis loss-of-coolant accident was evaluated by the AEC staff incorporating the primary containment design basis accident leak rate of 2.0%/day, (Ref. 3). The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90% for halogens, 90% for particulate iodine, and assuming the fission product release fractions stated in TID-14844, the maximum total whole body passing cloud dose is about 2 rem and the maximum thyroid dose is about 32 rem at the site boundary over an exposure duration of two hours. The resultant thyroid dose that would occur over the course of the accident is 98 rem at the boundary of the low population zone (LPZ). Thus, these doses are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment, resulting in a direct release of fission products from the primary containment through the filters and stack to the environs.

*NOTE: The initial leak rate testing performed during plant startup was conducted at a pressure of 54 psig in accordance with the original FSAR analysis of peak containment pressure (Pa).

Therefore, the specified primary containment leak rate is conservative and provides additional margin between expected offsite doses and 10 CFR 100 guidelines.

The design basis accident leak rate (L_a)at the peak accident pressure of 43 psig (P_a) is 2.0 weight percent per day. To allow a margin for possible leakage deterioration during the interval between Type A tests, the maximum allowable containment operational leak rate (L_{am}), is 0.75 L_a.

Type B and Type C tests are performed on testable penetrations and isolation valves during the interim period between Type A tests. This provided assurance that components most likely to undergo degradation between Type A tests maintain leaktight integrity. A controlled list of the testable penetrations and isolation valves subject to Type B and Type C testing is located in the Plant Administrative Control Procedures.

The containment leakage testing program is based on NRC guidelines for development of leak rate testing and surveillance schedules for reactor containment vessels, (Reference 4).

3.7.B and 4.7.B Bases

Primary Containment Power Operated Isolation Valves

Automatic isolation valves are provided on process piping which penetrates the containment and communicates with the containment atmosphere. The maximum closure times for these valves are selected in consideration of the design intent to contain released fission products following pipe breaks inside containment. Several of the automatic isolation valves serve a dual role as both reactor coolant pressure boundary isolation valves and containment isolation valves. The function of such valves on reactor coolant pressure boundary process piping which penetrates containment (except for those lines which are required to operate to mitigate the consequences of a loss-of-coolant accident) is to provide closure at a rate which will prevent core uncovery following pipe breaks outside primary containment. A controlled list of the primary containment power operated isolation valves is located in the Plant Administrative Control Procedures.

In order to assure that the doses that may result from a steam line break are within 10 CFR 100 guidelines, it is necessary that no fuel rod perforation results from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate the fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds. The test closure time limit of 5 seconds for these main steam isolation valves provides sufficient margin to assure that cladding perforations are avoided. Redundant valves in each line insure that isolation will meet the single failure criteria.

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.

The containment is penetrated by a large number of small diameter instrument lines. The excess flow check valves in these lines shall be tested once each operating cycle.

Containment vent/purge valves (CV-4300, CV-4301, CV-4302, CV-4303, CV-4306, CV-4307, and CV-4308) have been mechanically modified to limit the maximum opening angle to 30 degrees. This has been done to ensure these valves are able to close against the maximum differential pressure expected to occur during a design basis accident.

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 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 containment.

In the event that one or more primary containment isolation valves (PCIVs) are inoperable, either the inoperable valve must be restored to OPERABLE status or the affected penetration must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and deactivated automatic PCIV, a closed manual valve, a blind flange, or a check valve inside primary containment with flow through the valve secured. The specified time period of 4 hours is reasonable considering the time required to isolate the penetration and the relative importance of maintaining primary containment integrity.

3.7.C and 4.7.C Bases

Drywell Average Air Temperature

The drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell coolers remove heat and maintain a suitable environment. The average airspace temperature affects equipment OPERABILITY, personnel access, and the calculated response to postulated Design Basis Accidents (DBAs). The limitation on the drywell average air temperature was develcped as a reasonable upper bound based on operating plant experience. The limitation on drywell temperature is used in the safety analyses. Among the inputs to the design basis analysis is the initial drywell average air temperature. Analyses assume an initial average drywell air temperature of 135°F. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell temperature does not exceed the maximum allowable.

In the event of a DBA, with an initial drywell average temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the primary containment design temperature. As a result, the ability of primary containment to perform its design function is ensured.

With drywell average air temperature not within the limit of the LCO, drywell average air temperature must be restored within 8 hours. The Required Action is necessary to return operation to within the bounds of the primary containment analysis. The 8-hour Completion Time is acceptable considering the sensitivity of the analysis to variations in this parameter, and provides sufficient time to correct minor problems or to prepare the plant for an orderly shutdown.

Drywell air temperature is monitored at various elevations in the drywell. Due to the shape of the drywell, a volumetric average is used to determine an accurate representation of the actual average temperature.

The 24-hour frequency of the surveillance requirement was developed considering operating experience related to drywell average air temperature variations. Furthermore, the 24-hour frequency is considered adequate in view of other indications available in the control room.

3.7.D and 4.7.D Bases

Pressure Suppression Chamber - Reactor Building Vacuum Breakers

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and reactor building so that the structural integrity of the containment is maintained. The vacuum relief system from the pressure suppression chamber to reactor building consists of two 100% vacuum relief breakers (2 parallel sets of 2 valves in series). Operation of either system will maintain the pressure differential less than 2 psi, the external design pressure.

With one valve of a vacuum breaker assembly inoperable (incapable of opening) but known to be closed, the leak-tight primary containment boundary is intact.

The ability to mitigate an event that causes a containment depressurization is threatened, however, if both vacuum breakers in at least one vacuum breaker penetration are not OPERABLE. Therefore, the inoperable vacuum breaker must be restored to OPERABLE status within 72 hours based on the fact that the leak-tight primary containment boundary is being maintained.

With one valve of a vacuum breaker assembly open, the leak-tight primary containment boundary may be threatened. Therefore, it must be confirmed that at least one vacuum breaker in each affected line is closed. Failure to verify a closed vacuum breaker would imply that a breach in primary containment exists. The inoperable vacuum breakers must be restored to OPERABLE status within 72 hours. The 72-hour Completion Time takes into account the redundancy capability afforded by the remaining breakers, the fact that the OPERABLE breaker in each of the lines is closed, and the low probability of an event occurring that would require the vacuum breakers to be operable during this period.

3.7.E and 4.7.E Bases

Drywell - Pressure Suppression Chamber Vacuum Breakers

The capacity of the 7 drywell vacuum relief valves are sized to limit the pressure differential between the suppression chamber and drywell during post-accident drywell cooling operations to well under the design limit of 2 psi. They are sized on the basis of the Bodega Bay pressure suppression system tests. The ASME Boiler and Pressure Vessel Code, Section III, Subsection B, for this vessel allows a 2 psi differential; therefore, with one vacuum relief valve secured in the closed position and 6 operable valves, containment integrity is not impaired.

With one of the required vacuum breakers inoperable for opening but known to be closed (e.g., the vacuum breaker is not open, and may be stuck closed or not within its opening setpoint limit, such that it would not function as designed during an event that depressurized the drywell), the remaining six

OPERABLE vacuum breakers are capable of providing the vacuum relief function. A Completion Time of 72 hcurs is allowed to restore vacuum breakers to OPERABLE status. The 72-hour Completion Time takes into account the redundant capability afforded by the remaining breakers, reasonable time for the repairs, and the low probability of an event occurring during this period requiring the vacuum breakers to function.

An open vacuum breaker allows ccmmunication between the drywell and suppression chamber airspace, and, as a result, there is the potential for suppression chamber overpressurization due to this bypass leakage if a LOCA were to occur. Therefore, the open vacuum breaker must be closed. The 2-hour Completion Time is based on the time required to complete the alternate method of verifying that the vacuum breakers are closed, and the low probability of a DBA occurring during this period.

3.7.F and 4.7.F Bases

Main Steam Isolation Valve Leakage Control System (MSIV-LCS)

The MSIV-LCS system is provided to minimize the fission products which could bypass the standby gas treatment system after a LOCA. It is designed to be manually initiated after it has been determined that a LOCA has occurred and that the pressure between the MSIV's has decayed to less than 35 psig. The System is also inhibited from operating unless the inboard MSIV associated with the MSIV-LCS subsystem is closed and the reactor vessel pressure has decayed to less than 35 psig.

Checking the operability of the various components of the MSIV-LCS system monthly, and the motor-operated valves once every 3 months, assures that the MSIV-LCS system will be available in the remote possibility of a LOCA. Performance of a capacity test of the blowers and initiation of the entire system once per operating cycle assures that the MSIV-LCS system meets its design criteria. The testing frequency of the motor-operated valves is based on Section XI of the ASME Code. Allowance of thirty days to return a MSIV-LCS subsystem or blower to an operable status allows operational flexibility while maintaining protective capabilities.

3.7.G and 4.7.G BASES

Suppression Pool Level and Temperature

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1040 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum allowable pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 43 psig which is below the design pressure of 56 psig. The maximum volume of 61,500 ft^3 (equivalent to an indicated level of 60%) ensures the clearing loads from SRV discharges are not excessive and do not result in excessive pool swell loads during a Design Bases LOCA. The minimum volume of 58,900 (equivalent to an indicated level of 40%) ft^3 results in a submergence of approximately 3 feet. Based on Humboldt Bay, Bodega Bay, and Marviken test facility data as utilized in General Electric Company document number NEDE-21885-P and data presented in Nutech document, Iowa Electric document number 7884-M325-002, the following technical assessment results were arrived at:

1. Condensation effectiveness of the suppression pool can be maintained for both short and long term phases of the Design Basis

Accident (DBA), Intermediate Break Accident (IBA), and Small Break Accident (SBA) cases with three feet submergence.

2. There is no significant thermal stratification in the condensation oscillation regime after LOCA with three feet submergence.

3. There is some thermal stratification in the chugging regime for all break sizes. However, this will not inhibit the pressure suppression function of the suppression pool.

4. Seismic induced waves will not cause downcomer vent uncovering with three feet submergence.

5. Post-LOCA pool waves will not cause downcomer vent uncovering with three feet submergence.

6. Maximum post-LOCA drawdown will not cause downcomer vent uncovering and condensation effectiveness of the suppression pool will be maintained.

Therefore, with respect to downcomer submergence, this specification is adequate. The maximum temperature at the end of blowdown tested during the Humbolt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Using a 50°F rise (Table 6.2-1, UFSAR) in the suppression chamber water temperature and a minimum water volume of 58,900 ft^3, the 170° temperature which is used for complete condensation would be approached only if the suppression pool temperature is 120°F prior to the DBA-LOCA. Maintaining a pool temperature of 95°F will assure that the 170°F limit is not approached.

As part of the program to reduce the loads on BWR containments, the NRC issued NUREG-0783, which limits local suppression pool temperatures during Safety Relief Valve (SRV) actuations. Stable steam condensation is assured in the vicinity of T-type quenchers on SRV discharge lines if the following limits on local suppression pool temperatures are met:

1. For all plant transients involving SRV operations during which the steam flux through the quencher perforations exceeds 94 lbm/ft^2-sec, the suppression pocl local temperature shall not exceed 200°F.

2. For all plant transients involving SRV operations during which the steam flux through the quencher perforations is less than 42 lbm/ft^2-sec, the suppression pool local temperature shall be at least 20°F subcooled.

3. For all plant transients involving SRV operations during which the steam flux through the quencher perforations exceeds 42 lbm/ft^2-sec, but less than 94 lbm/ft^2-sec, the suppression pool local temperature is obtained by linearly interpolating the local temperatures established under aforementioned items 1 and 2.

| Maintaining the suppression pool temperature at or below the normal operating
| limit of 95°F, and scramming the reactor if the pool temperature reaches 110°F, will ensure that the local temperature limits outlined above are not exceeded during plant transients.(7)

Experimental data indicate that excessive steam condensing loads can be avoided if the peak local temperature of the suppression pool is maintained below 200°F during any period of relief valve operation. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

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 relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

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 continually monitored and frequently logged 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. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for core standby cooling systems operability as explained in Basis 3.5.G or the requirements of Specification 3.5.G.4 are met.

The interiors of the drywell and suppression chamber are coated to prevent corrosion and for ease of decontamination. The inspection of the coating during each major refueling outage, assures the paint is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.

3.7.H and 4.7.H BASES

Containment Atmosphere Dilution

In order to ensure that the containment atmosphere remains inerted, i.e., the oxygen-hydrogen mixture below the flammable limit, the capability to inject nitrogen into the containment after a LOCA is provided. The CAD system serves as the post-LOCA Containment Atmosphere Dilution System. By maintaining a minimum of 50,000 scf of liquid N_2 in the storage bank it is assured that a seven-day supply of N_2 for post-LOCA containment inerting is available.

The Post-LOCA Containment Atmosphere Dilution System design basis and description are presented in Section 6.2.5 of the Updated FSAR. In summary, the limiting criteria, based on the assumptions of Safety Guide No. 7 are:

1. Maintain oxygen concentration in the containment during post-LOCA conditions to less than 4 Volume %.

2. Limit the buildup in the containment pressure due to nitrogen addition to less than 30 psig.

3. To limit the offsite dose due to containment venting (for pressure control) to less than 30 rem to the thyroid.

By maintaining at least a 7-day supply of N_2 on site there will be sufficient time after the occurrence of a LOCA for obtaining additional nitrogen supply from local commercial sources. The system design contains sufficient redundancy to ensure its reliability. Thus, it is sufficient to test the operability of the whole system annually. The H_2 and O_2 analyzers are provided redundantly. There are two H_2 and two O_2 analyzers. By permitting continued reactor operation at rated power with one of the two analyzers of a given type (H_2 or O_2) inoperable, redundancy of analyzing capability will be maintained while not imposing an unnecessary interruption in plant operation. If one of the two analyzers of a particular type (H_2 or O_2) fails, the frequency of testing of the other analyzer of the same type will be increased from monthly to weekly to assure its continued availability. Monthly testing of the analyzers using bottled H_2 or O_2 will be adequate to ensure the system's readiness because of the multiplicity of design.

Due to the nitrogen addition, the pressure in the containment after a LOCA could possibly increase with time. Under the worst expected conditions the containment pressure will reach 30 psig in approximately 70 days. If and when that pressure is reached, venting from the containment shall be manually initiated. The venting path will be through the Standby Gas Treatment System in order to minimize the offsite dose.

Following a LOCA, periodic operation of the drywell and torus sprays may be used to assist the natural convection and diffusion mixing of hydrogen and oxygen.

| 3.7.I and 4.7.I BASES

| Oxygen Concentration

Safety Guide No. 7 assumptions for metal-water reactions result in hydrogen concentrations in excess of the Safety Guide No. 7 flammability limit. By keeping oxygen concentrations less than 5% (AEC has recommended 4%), Safety Guide No. 7 requirements are satisfied. The Containment Atmosphere Dilution System further assures that a combustible hydrogen/oxygen atmosphere will not be created in a post-LOCA condition.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the loss-of-coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration. The CAD system is not required to be OPERABLE during these inspections and when the containment is not inerted. This is to ensure personnel safety.

The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least once per week the oxygen concentration will be determined as added assurance.

3.7.J and 4.7.J BASES

<u>Secondary Containment</u>

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shut down and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling.

| 3.7.K and 4.7.K BASES

| Secondary Containment Automatic Isolation Dampers

| The function of the SCIVs, in combination with other accident-mitigation systems, is to limit fission-product release during the following postulated Design Basis Accidents such that offsite radiation exposures are maintained within the requirements of 10 CFR 100 or the NRC staff-approved licensing basis. Secondary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that fission products that escape from primary containment following a DBA, or which are released during certain operations when primary containment is not required to be OPERABLE or take place outside primary containment, are maintained within applicable limits. A controlled list of secondary containment automatic isolation dampers is located in the plant Administrative Control Procedures.

| The OPERABILITY requirements for SCIVs help ensure that adequate secondary containment leak tightness is maintained during and after an accident by minimizing potential paths to the environment. These isolation devices consist of either passive devices or active (automatic) devices. Locked-closed manual valves, deactivated automatic valves secured in their closed position, blind flanges, and closed systems are considered passive devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation (and possibly loss of secondary containment OPERABILITY).

| With one or more SCIVs inoperable, at least one isolation valve must be verified to be OPERABLE in each affected open penetration. This action may be satisfied by examining logs or other information to determine whether the valve is out of service for maintenance or other reasons.

| In the event that one or more SCIVs are inoperable, either the inoperable valve must be restored to OPERABLE status or the affected penetration must be isolated. The method of isolation must include the use of at least one

isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criteria are a closed and deactivated automatic SCIV, a closed manual valve, or a blind flange.

Demonstrating the isolation capabilities of each power-operated and automatic SCIV is required to demonstrate OPERABILITY. The simulated automatic initiation ensures that the valve will isolate as assumed in the safety analyses. The frequency of this SR is in accordance with the Inservice Testing Program.

3.7.L and 4.7.L BASES

Standby Gas Treatment System

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environs. Both standby gas treatment fans are designed to automatically start upon containment isolation and to maintain the reactor building pressure at approximately a negative 1/4-inch water gauge pressure; all leakage should be in-leakage. Only one of the two standby gas treatment systems is needed to cleanup the reactor building atmosphere upon containment isolation. If one system is made or found to be inoperable during reactor operation or core alterations, there is no immediate threat to the containment system performance. Thus, reactor or refueling operation(s) may continue while repairs are being made, provided the requirements of Specifications 3.7.L.3 and 3.9.D, respectively, are met. If neither circuit is operable, the plant is brought to a condition where the standby gas treatment system is not required.

High efficiency particulate absolute (HEPA) filters are installed before and after the charcoal adsorbers to minimize potential release of particulates to the environment and to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of ≤ 0.1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99.9 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 99% for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10 CFR 100 guidelines for the accidents analyzed, as the Updated FSAR Section 15.6.6 for the loss-of-coolant accident shows compliance with 10 CFR 100 guidelines with an assumed efficiency of 99% for the adsorber. Operation of the fans significantly different from the design flow envelope will change the removal efficiency of the HEPA filters and charcoal adsorbers.

A pressure drop test across the combined HEPA filters and charcoal adsorbers will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Heater capability, pressure drop and air distribution should be determined annually to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with USAEC Report DP-1082. Iodine removal efficiency tests shall follow RDT Standard M-16-1T. (The design of the SGTS system allows the removal of charcoal samples from the bed directly through the use of a grain thief.) Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 4.7-1. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N101.1-1972. Any HEPA filters found defective shall be replaced. The replacement HEPA filters should be steel cased and designed to military specifications MIL-F-51068C and MIL-F-51079A. The HEPA filters should satisfy the requirements of UL-586. The HEPA filter separators should be capable of withstanding iodine removal sprays. HEPA filters should be tested individually by the appropriate Filter Test Facility listed in the current USNRC Health and Safety Bulletin for Filter Unit Inspection and Testing Service. The Filter Test Facility should test each filter at 100%, and 20% of rated flow, with the filter encapsulated to disclose frame and gasket leaks.

All elements of the heater are demonstrated to be functional and operable during the test of heater capacity. Demonstration of 22 KW capability assures relative humidity below 70%.

System drains are present in the filter/adsorber banks, loop-seal water level is checked to ensure no bypass leakage from the banks.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

Demonstration of the automatic initiation capability and operability of filter cooling is necessary to assure system performance capability.

Initiating reactor building isolation and operation of the standby gas treatment system to maintain at least a 1/4 inch of water vacuum within the secondary containment provides an adequate test of the operation of the reactor building isolation valves, leaktightness of the reactor building and performance of the standby gas treatment system. During the performance of this test, the averaging of individual manometer readings compensates for wind effects with wind speeds up to 15 mph (NG-91-0273). Functionally testing the initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Performing these tests prior to refueling will demonstrate secondary containment capability prior to the time the primary containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

3.7.M and 4.7.M BASES

Mechanical Vacuum Pump

The purpose of isolating the mechanical vacuum pump line is to limit the release of activity from the main condenser. During an accident, fission products could be transported from the reactor through the main steam lines to the condenser. The fission product radioactivity would be sensed by the main steam line radioactivity monitors which initiate isolation.

3.7.A & 4.7.A REFERENCES

1. "Duane Arnold Energy Center Power Uprate", NEDC-30603-P, May, 1984 and Attachment 1 to letter L. Lucas to R.E. Lessly, "Power Update BOP Study Report," June 18, 1984.

2. ASME Boiler and Pressure Vessel Code, Nuclear Vessels, Section III, maximum allowable internal pressure is 62 psig.

3. Staff Safety Evaluation of DAEC, USAEC, Directorate of Licensing, January 23, 1973.

4. 10 CFR Part 50, Appendix J, Reactor Containment Testing Requirements, Federal Register, April 19, 1976.

5. Deleted

6. Deleted

7. General Electric Company, Duane Arnold Energy Center Suppression Pool Temperature Response, NEDC-22082-P, March 1982.

TABLE 4.7-1
SUMMARY TABLE OF NEW ACTIVATED CARBON PHYSICAL PROPERTIES

	TEST	ACCEPTABLE TEST METHOD	ACCEPTABLE RESULTS		TEST SCHEDULE ON BASE MATERIAL	TEST SCHEDULE ON FINISHED ADSORBENT
1.	Particle Size Distribution	ASTM D 2862	Retained on #6 ASTM E11 Sieve: Retained on #8 ASTM E11 Sieve: Through #8,retained on #12 Sieve: Through #12,retained on #16 Sieve: Through #16 ASTM E11 Sieve: Through #16 ASTM E323 Sieve:	0.0% 5.0% maximum 40% to 60% 40% to 60% 5.0% maximum 1.0% to maximum	-	Batch[c]
2.	Hardness Number	MIL-C17605B para. 4.6.4			Batch	
3.	Ignition Temperature	RDT M16-1T. Appendix C	340°C minimum at 100 fpm		-	Batch
4.	Surface Area	BET Surface Area	1000 m^2/gr minimum		Batch	
5.	Radioiodine Removal Efficiency					
	a. Elemental Iodine, DBA Temperature and Pressure	RDT M16-1T, para. 4.5.2 except DBA Temperature and pressure are used[a]	99.9%		-	Qualification[b]
	b. Methyl Iodide, DBA Temperature and Pressure	RDT M16-1T, para. 4.5.4 except DBA Temperature and pressure are used[a]	95% for 95% relative humidity 99.5% for 70% relative humidity		-	Batch
	c. Retention	RDT M16-1T, para. 4.5.5	99%		-	Qualification
6.	Moisture Content Efficiency	ASTM D2867, Xylene Method	3% maximum			Batch
7.	Ash Content	ASTM D2866	6% maximum		Qualification	-
8.	Bulk Density	ASTM D2854	Report value		-	Batch
9.	Impregnant Content	State Procedure	State type (not to exceed 5% by weight)		-	Batch
10.	Impregnant Leachout	State Procedure	Report value		-	Qualification

[a]DBA Maximum Temperature (rounded to the next highest decade in °F, i.e., 252°F is 260°F) and Maximum Pressure (rounded to the next highest decade in psig, i.e., 51 psig is 60 psig).
[b]Qualification test: Test which establishes the suitability of a product for a general application normally a one-time test reflecting historical typical performance of material.
[c]Batch test: Test made on a production batch of product to establish suitability for a specific application.

ENVIRONMENTAL CONSIDERATION

10 CFR 51.22(c)(9) identifies certain licensing and regulatory actions which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; and (3) result in an increase in individual or cumulative occupational radiation exposure. Iowa Electric Light and Power has reviewed this request and determined that the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment. The basis for this determination follows:

Basis

The change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons:

1. As demonstrated in Attachment 1, the proposed amendment does not involve a significant hazards consideration.

2. The proposed revisions to the limiting conditions for operation and surveillance requirements for primary containment integrity, secondary containment integrity and other systems and equipment of TS sections 3.7, "Containment Systems" have no effect on the types or amounts of effluents released offsite.

3. The proposed revisions to the limiting conditions for operation and surveillance requirements for primary containment integrity, secondary containment integrity and other systems and equipment of TS sections 3.7, "Containment Systems" have no effect on individual or cumulative occupational radiation exposure.