PROPOSED CHANGE RTS-207 TO THE DUANE ARNOLD ENERGY CENTER TECHNICAL SPECIFICATIONS

The holders of license DPR-49 for the Duane Arnold Energy Center propose to amend Appendix A (Technical Specifications) to said license by deleting current pages and replacing them with the attached, new pages.

LIST OF AFFECTED PAGES

iii	3.7-19a	3.7-35
vi	3.7-19b	3.7-36
3.7-1	3.7-20	3.7-37
3.7-2	3.7-21	3.7-38
3.7-3	3.7-22	3.7-39
3.7-4	3.7-23	3.7-40
3.7-5	3.7-23a	3.7-41
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3.7-15	3.7-32b	3.7-49a
3.7-16	3.7-33	3.7-50
3.7-17	3.7-34	6.11-6
3.7-18		
3.7-19		

	Existing Page	Summary of Change
1)	iii	Change page numbers for Section 3.7 and add items E and F.
2)	vi	Revise page numbers for Tables 3.7-1, 3.7-2, 3.7-3 and 4.7-1.
3)	3.7-1	Add a colon to Specification 3.7.A.1. Change "95F" (in Specification 3.7.A.1.c(1)) to "95°F" and add the word "suppression" to the word "pool." Revise Surveillance Requirement 4.7.A.1.d to incorporate wording of Surveillance Requirement 4.7.A.2.a(1) to eliminate redundancy.
4)	3.7-2 & 3.7-3	Add the word "suppression" to the word "pool" in Specification 3.7.A.1.c(3) and (4). Revise Specification 3.7.A.2 and Surveillance Requirement 4.7.A.2 to describe leakage rate restrictions as required by 10 CFR Part 50, Appendix J. Move Surveillance Requirement 4.7.A.2.a(1), 1st paragraph to 4.7.A.1.d. Move Surveillance Requirement 4.7.A.2.a.6) to 4.7.A.2.c and add new Surveillance Requirement 4.7.A.2.c(1), (2) and (3) to conform to 10 CFR Part 50, Appendix J. Add new Specification 3.7.A.2.d.
5)	3.7-4	Move Surveillance Requirement 4.7.A.2.a.7) and 8) to Specification 3.7.A.2.a and .b and provide clarifying language found in the Standard Technical Specifications. Move Surveillance Requirement 4.7.A.2.a.9) to Surveillance Requirement 4.7.A.2.a. and b.
6)	3.7-5	Move Surveillance Requirement 4.7.A.2.b.1) and 2) to 4.7.A.2.d. Move Surveillance Requirement 4.7.A.2.c.3) to Specification 3.7.A.2.c.
7)	3.7-6	Move Surveillance Requirement 4.7.A.2.d.1) to 4.7.A.2.a. Add new Surveillance Requirement 4.7.A.2.d(3). Move Surveillance Requirement 4.7.A.2.d.2a) and b) to 4.7.A.2.d(1) and (2). Change "major refueling" to "refueling."
8)	3.7-6a	Move Surveillance Requirement 4.7.A.2.d.2.c to 4.7.A.2.d(4).
9)	3.7-7	Move Surveillance Requirement 4.7.A.2.d.3) to 4.7.A.2.e and change "major refueling" to "refueling." Move Surveillance Requirement 4.7.A.2.d.4) to 4.7.A.2.f, 4.7.A.2.e to 4.7.A.2.g and 4.7.A.2.f to 4.7.A.2.h.
10)	3.7-7a, 3.7-8, & 3.7-9	Replace Surveillance Requirement 4.7.A.2.g with 4.7.A.2.i which substitutes alternate wording to require submittal of Type A, B and C test results in accordance with 10 CFR Part 50, Appendix J.
11)	3.7-10 & 3.7-11	Move Specification 3.7.A.4 and Surveillance Requirement 4.7.A.4 to page 3.7-6 and 3.7-7. Change "once each operating cycle" to "once per operating cycle." Delete reference to continuous leak rate monitoring system.
12)	3.7-12	Move Specification 3.7.A.5 and Surveillance Requirement 4.7.A.5 to page 3.7-8.

	Existing Page	Summary of Change
13)	3.7-13	Move Specification 3.7.A.6 and Surveillance Requirement 4.7.A.6 to pages 3.7-8 and 3.7-9.
14)	3.7-14	Move Specifications 3.7.A.9 and 10 and Surveillance Requirements 4.7.A.9 and 10 to page 3.7-10 and renumber to 3.7.A.7 and 3.7.A.8, and 4.7.A.7 and 4.7.A.8 respectively. Delete Specification 3.7.A.8 and replace with individual Specifications on pages 3.7-2 and 3.7-6.
15)	3.7-15 & 3.7-16	Move Specification 3.7.B and Surveillance Requirement 4.7.B to pages 3.7-11 and 3.7-12.
16)	3.7-17 thru 3.7-19	Move Specifications 3.7.C.1, 3.7.C.2 and Surveillance Requirement 4.7.C.1 and 4.7.C.2 to page 3.7-13. Move Surveillance Requirement 4.7.D and Specification 3.7.D to pages 3.7-14 and 3.7-15.
17)	3.7-19a	Move Specification 3.7.E and Surveillance Requirement 4.7.E to page 3.7-16.
18)	3.7-19b	Move Specification 3.7.F and Surveillance Requirement 4.7.F to page 3.7-17. Change Surveillance Requirement 4.7.F.1 from "once during each operating cycle" to "once per operating cycle."
19)	3.7-20	Change page number to 3.7-18 and add footnotes to table. (These footnotes are already included on existing page 3.7-21.)
20)	3.7-21	Change page number to 3.7-19 and change reference to Specification 4.7.A.2.d.2 to 4.7.A.2.d.
21)	3.7-22	Change page number to 3.7-20.
22)	3.7-23	Change page number to 3.7-21.
23)	3.7-23a & 3.7-24	Move information to page 3.7-22.
24)	3.7-25	Change page number to 3.7-23 and change orientation of table.
25)	3.7-26	Change page number to 3.7-24.
26)	3.7-27, 3.7-28, 3.7-29, & 3.7-29a	Change page numbers to 3.7-25 and 3.7-26.
27)	3.7-30	Change page number to 3.7-28 and numbering scheme. Change "colant" to "coolant." Add new guidance for new Specification 3.7.A.2.d.
28)	3.7-31 & 3.7-32	Change page numbers to 3.7-29 and 3.7-30. Correct spelling from "Humbolt Bay" to "Humboldt Bay."

	Existing Page	Summary of Change
29)	3.7-32a	Change page number to 3.7-30.
30)	3.7-32b, 3.7-36, 3.7-37, & 3.7-38	Move the information found on these pages to 3.7-31, 3.7-32, 3.7-33, 3.7-34 and 3.7-35. ChaNGE Reference 7 to Reference 5.
31)	3.7-33	Change "major refueling outage" to "refueling outage" and move information to page 3.7-34.
32)	3.7-34 & 3.7-35	Move information to pages 3.7-33 and 3.7-34. Move information regarding suppression chamber temperature and volume instrument checks and drywell/suppression chamber inspection to page 3.7-33. Change frequency of drywell/suppression chamber inspection from "during each major refueling outage, approximately once per year" to "once per operating cycle."
33)	3.7-38 & 3.7-39	Delete interior drywell inspection requirement because it is redundant with the inspection discussed on new page 3.7-33.
34)	3.7-39 & 3.7-40	Change page numbers to 3.7-35 and 3.7-36.
35)	3.7-41 thru 3.7-46	Change page numbers to 3.7-37 through 3.7-39.
36)	3.7-46 3.7-47, & 3.7-48	Change page numbers to 3.7-40. Move information regarding instrument lines to page 3.7-33.
37)	3.7-48a	Move information regarding experimental data for excessive steam condensing loads to page 3.7-31.
38)	3.7-48b	Move information to page 3.7-41.
39)	3.7-49	Move information to page 3.7-42 and change Reference 7 to Reference 5.
40)	3.7-49a	Move information to pages 3.7-40 and 41.
41)	3.7-50	Move information to page 3.7-27.
42)	6.11-6	Change reference to Specification 4.7.A.2.f to 4.7.A.2.i.

LIMITING CONDITION 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

1. At any time that the nuclear system is pressurized above atmospheric or work is being done which has the potential to drain the vessel, the suppression pool water volume and temperature shall be maintained with the following limits:

   a. Maximum water volume - 61,500 cubic feet

   b. Minimum water volume - 58,900 cubic feet

   c. Maximum water temperature

      (1) During normal power operation - 95°F.

      (2) During testing which adds heat to the suppression pool, the water temperature shall not exceed 10°F above the normal power operation limit specified in (1) above. In connection with such testing, the suppression pool temperature must be reduced to below the normal power operation limit specified in (1) above within 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

1.a. The pressure suppression pool water level and temperature shall be checked once per day.

   b. Whenever there is indication of relief valve operation or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

   c. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 200°F or more, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

   d. The interior surfaces of the drywell and torus, including waterline regions, 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.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
(3) The reactor shall be scrammed from any operating condition if the suppression pool temperature reaches 110°F. Power operation shall not be resumed until the suppression pool temperature is reduced below the normal power operation limit specified in (1) above.	
(4) During reactor isolation conditions, the reactor shall be depressurized to less than 200 psig at normal cooldown rates if the suppression pool temperature reaches 120°F.	
d. If Specification 3.7.A.1 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a COLD SHUTDOWN Condition within the following 24 hours.	
2. 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).	
Primary containment leakage rates shall be limited to:	
a. An overall integrated leakage rate of less than or equal to L_a [2.0 percent by weight of the containment air per 24 hours at P_a (43 psig)].	2. The primary containment leakage rates shall be demonstrated at the following test schedules and shall be determined in conformance with the criteria specified in Appendix J to 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972:
	TYPE A TESTS
	a. A set of three Type A integrated containment leakage rate tests shall be conducted at approximately equal intervals (at test pressure P_a) such that three tests are conducted during each 10-year period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.
b. A combined leakage rate for all penetrations and valves subject to Type B and C tests, except for main steam isolation valves** and valves that are hydrostatically tested per Table 3.7-2, shall be less than 0.60 L_a.	b. If any periodic Type A test leakage is greater than 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 meet the 0.75 L_a criteria the test schedule in Specification 4.7.A.2.a may be resumed.

**Exemption to Appendix J

LIMITING CONDITION FOR OPERATION

c. Less than or equal to 11.5 scf per hour for any one main steam isolation valve when pressurized at greater than or equal to 24 psig.

d. If Specification 3.7.A.2 cannot be met, either:

(1) Restore PRIMARY CONTAINMENT INTEGRITY within 1 hour from time of initial loss or,

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

SURVEILLANCE REQUIREMENT

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 test 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 percent of the total measured leakage at P_a (43 psig).

d. Type B Tests

(1) 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 refueling, or other convenient intervals, but in no case at intervals greater than two years.

(2) 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.

LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT

(3) Any penetration subject to Type B testing (except the personnel air lock) shall be tested at P_a if opened subsequent to performing a Type A or Type B test.

It shall be verified that, when the measured leakage rate for this penetration is added to the leakage rates determined pursuant to Surveillance Requirement 4.7.A.2.d for all other Type B and C penetrations and valves, the combined leakage rate is less than or equal to 0.60 L_a.

(4) 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.

e. Type C Tests

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

f. 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 CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	g. Seal Replacement 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. The baseline for this requirement was established during the 1983 refueling outage.
	h. 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 the test report required pursuant to Surveillance Requirement 4.7.A.2.i. 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.
	i. Reporting Type A, B and C test results shall be submitted to the Commission approximately 3 months after the performance of each test. The report shall be prepared in accordance with 10 CFR Part 50, Appendix J, Paragraph V.B.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers	3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers
a. Except as specified in Specification 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be OPERABLE at all times when PRIMARY CONTAINMENT INTEGRITY is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building vacuum breakers shall be 0.5 psid.	a. The pressure suppression chamber-reactor building vacuum breakers and associated instrumentation including setpoint shall be checked for proper operation every three months.
b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is sooner made OPERABLE, provided that the repair procedure does not violate PRIMARY CONTAINMENT INTEGRITY.	
c. If Specification 3.7.A.3 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a COLD SHUTDOWN Condition within the following 24 hours.	
4. Drywell-Pressure Suppression Chamber Vacuum Breakers	4. Drywell-Pressure Suppression Chamber Vacuum Breakers
a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be OPERABLE and positioned in the fully closed position (except during testing) except as specified in Specifications 3.7.A.4.b & 3.7.A.4.c, below.	a. Each drywell-suppression chamber vacuum breaker shall be exercised through an opening-closing cycle every 30 days.
b. One drywell-suppression chamber vacuum breaker may be non-fully closed as indicated by its position lights so long as it is determined that total	b. When a vacuum breaker indicates non-fully closed, IMMEDIATELY and every 30 days thereafter determine that a bypass area

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
drywell to suppression pool bypass area of less than 0.2 square feet exists.	of not more than 0.2 square feet exists. A detailed description of allowable drywell to suppression pool bypass leakage is found in Section 6.2.1.3.5 of the Updated FSAR.
c. One drywell-suppression chamber vacuum breaker may be determined to be inoperable for opening.	c. When it is determined that a vacuum breaker valve is inoperable for opening at a time when operability is required, all other vacuum breaker valves shall be exercised IMMEDIATELY and every 15 days thereafter until the inoperable valve has been returned to normal service. Once per OPERATING CYCLE, each vacuum breaker valve shall be visually inspected to insure proper maintenance and operation.
d. If Specifications 3.7.A.4.a, .b, or .c cannot be met, the situation shall be corrected within 24 hours or the unit will be placed in a COLD SHUTDOWN Condition in an orderly manner.	d. A leak test of the drywell to suppression chamber structure shall be conducted at each refueling. The drywell pressure will be increased by approximately 1 psi with respect to the wetwell pressure and held constant. The 2 psig scram setpoint will not be exceeded. The subsequent wetwell pressure transient (if any) will be monitored with a precision pressure gauge capable of detecting a small pressure increase. If the drywell pressure cannot be increased by 1 psi over the wetwell pressure, it would be because excess leakage exists. Excess leakage would require the leakage source to be identified and eliminated before primary system pressurization. A more detailed description of this test is found in Section 6.2.6.3.5.3 of the Updated FSAR.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
5. Oxygen Concentration	5. Oxygen Concentration
a. After completion of the startup test program and demonstration of plant electrical output, the primary containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during REACTOR POWER OPERATION with reactor coolant pressure above 90 psig, except as specified in Specification 3.7.A.5.b.	The primary containment oxygen concentration shall be measured and recorded at least twice weekly.
b. Within the 24-hour period subsequent to placing the reactor in the Run Mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.	
6. Containment Atmosphere Dilution	6. Containment Atmosphere Dilution
a. Whenever the reactor is in POWER OPERATION, 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 the reactor must be taken out of POWER OPERATION.	a. The post-LOCA containment atmosphere dilution system shall be functionally tested once per OPERATING CYCLE.
b. 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 the reactor must be taken out of POWER OPERATION.	b. The volume in the N_2 storage bank shall be recorded weekly.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
c. 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.	c. 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 once per OPERATING CYCLE in conjunction with Specification 4.7.A.6.a.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
7. Purging The time which containment vent/purge valves (CV-4302, CV-4303, CV-4300, CV-4301 and CV-4307) can be open is limited to a maximum of 90 hours per calendar year, not including the 24 hour period prior to shutdown and the 24 hour period subsequent to placing the reactor in the Run Mode following a shutdown as specified in Specification 3.7.A.5.b. This restriction applies whenever PRIMARY CONTAINMENT INTEGRITY is required.	
8. If Specification 3.7.A.7 cannot be met, prepare and submit a Special Report to the Commission pursuant to Specification 6.11.3 within the next 30 days outlining the cause of the limits being exceeded and the plans for limiting the time which these valves will be open.	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
B. Standby Gas Treatment System	B. Standby Gas Treatment System
1. Except as specified in Specification 3.7.B.3 below, both trains of the standby gas treatment system and the diesel generators required for operation of such trains shall be OPERABLE at all times when SECONDARY CONTAINMENT INTEGRITY is required.	1.a At least once per OPERATING CYCLE it shall be demonstrated that pressure drop across the combined high efficiency and charcoal filters is less than 11 inches of water at 4000 cfm.
	b. At least once per OPERATING CYCLE demonstrate that the inlet heaters on each train are capable of an output of at least 11 Kw.
	c. At least once per OPERATING CYCLE demonstrate that air distribution is uniform within 20% of averaged flow per unit across HEPA filters.
	d. At least once per OPERATING CYCLE automatic initiation of each branch of the standby gas treatment system shall be demonstrated.
	e. At least once per OPERATING CYCLE manual operability of the bypass system for filter cooling shall be demonstrated.
	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.B.2.b to assure that no flow blockage has occurred.
2.a The results of the inplace cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq$ 99.9% DOP removal and $\geq$ 99.9% halogenated hydrocarbon removal.	2.a The tests and sample analysis of Specification 3.7.B.2 shall be performed initially and at least once per year 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.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
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 to 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.
	d. Each circuit shall be operated with the heaters on at least 10 hours every month.
3. From and after the date that one train of the standby gas treatment system is made or found to be inoperable for any reason, continued reactor operation or fuel handling is permissible only during the succeeding seven days unless such train is sooner made OPERABLE, provided that during such seven days all active components of the other standby gas treatment train shall be OPERABLE.	3. When one train of the standby gas treatment system becomes inoperable, the OPERABLE train shall be demonstrated to be OPERABLE IMMEDIATELY and daily thereafter.
4. If Specifications 3.7.B.1, 3.7.B.2 and 3.7.B.3 are not met, the reactor shall be placed in the COLD SHUTDOWN Condition and fuel handling operations shall be prohibited.	

LIMITING CONDITION FOR OPERATION

C. 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.C.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 COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENT

C. Secondary Containment

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

   a. A preoperational secondary containment capability test shall be conducted after isolating the reactor building and placing either standby gas treatment system filter train in operation. Such tests shall demonstrate the capability to maintain 1/4 inch of water vacuum under calm wind (<5 mph) conditions with a filter train flow rate of not more than 4000 cfm.

   b. Additional tests shall be performed during the first OPERATING CYCLE under an adequate number of different environmental wind conditions to enable valid extrapolation of the test results.

   c. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind (<5 mph) conditions with a filter train flow rate of not more than 4000 cfm, shall be demonstrated at each REFUELING OUTAGE prior to refueling.

   d. After a secondary containment violation is determined, the standby gas treatment system will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4 inch of water negative pressure under calm wind conditions.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
D. Primary Containment Power Operated Isolation Valves	D. Primary Containment Power Operated Isolation Valves
1. During reactor power operating conditions, all isolation valves listed in Table 3.7-3 and all instrument line flow check valves shall be OPERABLE except as specified in 3.7.D.2.	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 (except for those exempted as noted in Table 3.7-3) 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 week the main steam line power-operated isolation valves shall be exercised by partial closure and subsequent reopening. d. At least once per OPERATING CYCLE the operability of the reactor coolant system instrument line flow check valves shall be verified.
2. In the event any isolation valve specified in Table 3.7-3 becomes inoperable, (except for those exempted as noted in Table 3.7-3) REACTOR POWER OPERATION may continue provided at least one valve in each line having an inoperable valve shall be in the mode corresponding to the isolated condition.	2. Wherever an isolation valve listed in Table 3.7-3 is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3. If Specifications 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the COLD SHUTDOWN Condition within 24 hours.	

LIMITING CONDITION FOR OPERATION

E. 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.E.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 OPERABLE.

3. If the requirements of 3.7.E cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the COLD SHUTDOWN Condition within 24 hours.

SURVEILLANCE REQUIREMENT

E. 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/Month
d. Heater Operability	Once/Month
e. Blower Capacity	Once/Operating Cycle

2. When it is determined that one MSIV-LCS subsystem or one blower is inoperable, the other MSIV-LCS subsystem or blower shall be demonstrated to be OPERABLE IMMEDIATELY. The OPERABLE MSIV-LCS subsystems shall be demonstrated to be OPERABLE weekly thereafter.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
F. Mechanical Vacuum Pump	F. 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.	1. At least once per OPERATING CYCLE verify automatic securing and isolation of the mechanical vacuum pump.
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 limits of 3.7.F.2 are not met, the vacuum pump shall be isolated.	

TABLE 3.7-1

CONTAINMENT PENETRATIONS SUBJECT TO TYPE B TEST REQUIREMENTS

Penetration #	Type	Description
1	Testable Gaskets[2]	Personnel Lock Equipment Door
1	Personnel Lock[2]	Personnel Lock Doors and Penetrations
2	Testable Gaskets	Equipment Access
4	Testable Gaskets	Head Access
6	Testable Gaskets	CRD Removal Hatch
35A-D	Testable Gaskets	TIP Drives (4)
53	Testable Gaskets	Spare
----	Testable Gaskets	Drywell Head Flange
58 A-H	Testable Gaskets	Stabilizer Access Ports (8)
200A-B	Testable Gaskets	Torus Access Hatches (2)
100B,C,E,F,G	Electrical Canister	(B,C,E,F) Neutron Monitoring, (G) RPV Vibration Monitoring
101A,C	Electrical Canister	(C) (A) Recirc Pump Power
103	Electrical Canister	Thermocouples
104A-D	Electrical Canister	CRD Rod Position Indicator
105B,D	Electrical Canister	(B,D) Power & Control
106A,C	Electrical Canister	(A,C) Power & Control
230B	Electrical Canister	Vacuum Breakers Electrical Cables

[1]Test inboard flange of designated valves.

[2]Testing to be in accordance with Technical Specification Section 4.7.A.2.d.

TABLE 3.7-1 (Continued)

CONTAINMENT PENETRATIONS SUBJECT TO TYPE B TEST REQUIREMENTS

Penetration #	Type	Description
7A-D	Expansion Bellows	Steam to Turbine
9A,B	Expansion Bellows	RPV Feedwater
10	Expansion Bellows	Steam to RCIC Turbine
11	Expansion Bellows	Steam to HPCI Turbine
12	Expansion Bellows	Shutdown Pump Supply RHR
13A,B	Expansion Bellows	RHR Pump Discharge
15	Expansion Bellows	RWCU Supply
16A,B	Expansion Bellows	Core Spray Pump Discharge
17	Expansion Bellows	RPV Head Spray
201A-H	Expansion Bellows	Vent Lines
25	Flange "O" Rings[1]	Drywell Purge Outlet CV-4302
26,220	Flange "O" Rings	Drywell & Torus Purge Supply, CV-4307, CV-4308
205	Flange "O" Rings	Torus Purge Outlet, CV-4300
213A, B	Flange "O" Rings	Torus Drain Lines
231	Flange "O" Rings	Torus Vacuum Breakers, CV-4304, CV-4305

[1]Test inboard flange of designated valves.

[2]Testing to be in accordance with Technical Specification Section 4.7.A.2.d.

TABLE 3.7-2

CONTAINMENT ISOLATION VALVES
SUBJECT TO TYPE C TEST REQUIREMENTS

PENETRATION #	SYSTEM	BOUNDARY VALVES
7A	Main Steam Line	CV-4412[4], 4413
7B	Main Steam Line	CV-4415[4], 4416
7C	Main Steam Line	CV-4418[4], 4419
7D	Main Steam Line	CV-4420[4], 4421
8	Main Steam Line Drain	MO-4424
9A	Feedwater & HPCI Feed	V-14-3
9A[2]	Feedwater & HPCI Feed	MO-4441, MO-2312
9B	Feedwater	V-14-1
9B[2]	Feedwater & RCIC Feed & RWCU Return	MO-2740, MO-4442, MO-2512
10	RCIC Condensate Return	CV-2411
10	Steam to RCIC Turbine	MO-2401
11	Steam to HPCI Turbine	MO-2239
11	HPCI Condensate Return	CV-2212
15	RWCU Supply	MO-2700, MO-2701
16A	Core Spray Pump Discharge	MO-2115, MO-2117
16B	Core Spray Pump Discharge	MO-2135, MO-2137
19	Drywell Floor Drain Discharge	CV-3704, CV-3705
20	Demineralized Water Supply	V-09-65, V-09-111
21	Service Air Supply	V-30-287, Blind Flange
22, 229	Containment Compressor Discharge	CV-4371A, CV-4371C, V-43-214
23A[3], B[3]	Well Cooling Water Supply	CV-5718A, CV-5718B, V-57-75, V-57-76,
24A[3], B[3]	Well Cooling Water Return	CV-5704A, CV-5704B, V-57-77, V-57-78
25	Drywell Purge Outlet	CV-4302[4], CV-4303, CV-4310
26, 220	Drywell and Torus Purge Supply	CV-4306, CV-4307[4], CV-4308[4]
26, 220	Drywell and Torus Nitrogen Makeup	CV-4311, CV-4312, CV-4313

TABLE 3.7-2 (Continued)

CONTAINMENT ISOLATION VALVES
SUBJECT TO TYPE C TEST REQUIREMENTS

PENETRATION #	SYSTEM	BOUNDARY VALVES
32D	Containment Compressor Suction	CV-4378A, CV-4378B
32E	Recirc Pump "A" Seal Purge	V-17-96, CV-1804B
32F	Recirc Pump "B" Seal Purge	V-17-83, CV-1804A
35A,B,C,D	T.I.P Drives	T.I.P Ball Valves and Check Valve on X-35A
36[1]	CRD Return	V-17-53, V-17-52, V-17-54
39A	Containment Spray/CAD Supply	SV-4332A, SV-4332B
39B	Containment Spray/CAD Supply	SV-4331A, SV-4331B
40C,D	Post-Accident Sampling/Jet Pump Sample	SV-4594A, SV-4594B, SV-4595A, SV-4595B
41	Recirc Loop Sample	CV-4639[4], CV-4640
42	Standby Liquid Control	V-26-8, V-26-9
46E	O_2 Analyzer	SV-8105B, SV-8106B
48	Drywell Equipment Drain Discharge	CV-3728, CV-3729
50B	O_2 Analyzer	SV-8101A, SV-8102A,
50E	O_2 Analyzer	SV-8103A, SV-8104A,
50D	O_2 Analyzer	SV-8105A, SV-8106A
54[3]	Reactor Building Closed Cooling Water Return	MO-4841A
55[3]	Reactor Building Closed Cooling Water Supply	MO-4841B
56C	O_2 Analyzer	SV-8101B, SV-8102B,
56D	O_2 Analyzer	SV-8103B, SV-8104B
205	Torus Purge Outlet	CV-4300[4], CV-4301, CV-4309
211A	Torus Spray/CAD Supply	SV-4333A, SV-4333B
211B	Torus Spray/CAD Supply	SV-4334A, SV-4334B
212[1]	RCIC Turbine Exhaust	V-24-8[4], V-24-23 V-24-46, V-24-47
214[1]	HPCI Turbine Exhaust	V-22-16, V-22-17[4] V-22-63, V-22-64

TABLE 3.7-2 (Continued)

CONTAINMENT ISOLATION VALVES
SUBJECT TO TYPE C TEST REQUIREMENTS

PENETRATION #	SYSTEM	BOUNDARY VALVES
219	HPCI/RCIC Exhaust Vacuum Breaker	MO-2290A, MO-2290B
222[1]	HPCI Condensate	V-22-21, V-22-22[4]
229B	O_2 Analyzer	SV-8107A, SV-8108A,
229C	O_2 Analyzer	SV-8109A, SV-8110A,
229G	O_2 Analyzer	SV-8107B, SV-8108B,
229F	O_2 Analyzer	SV-8109B, SV-8110B,
229H	Post-Accident Sampling System Liquid Sample Return	SV-8772A, SV-8772B
231	Torus Vacuum Breakers	CV-4304[4], V-43-169
231	Torus Vacuum Breakers	CV-4305[4], V-43-168

NOTES TO TABLE 3.7-2

[1]Test volume is filled with demineralized water then pressurized to 1.10 P_a with air or nitrogen for test. For all other penetrations (except 7A-D), test volumes are pressurized to P_a with air or nitrogen for test.

[2]MO-4441, MO-4442 will be remote manually closed.

[3]In accordance with 10 CFR 50, Appendix A, General Design Criterion 57, the redundant barriers are a single isolation valve outside containment and a closed system inside. Testing of the single isolation valve only is required. Manual valves V-57-75, V-57-76, V-57-77 and V-57-78 will be normally locked closed.

[4]Tested in reverse direction.

TABLE 3.7-3

PRIMARY CONTAINMENT POWER OPERATED ISOLATION VALVES

Isolation Group (Note 1)	Valve Identification	Number of Power Operated Valves	Maximum Operating Time (Seconds)	Normal Position	Action on Initiating Signal
1	*Main Steam Line	8	3<T<5	O	GC
1	Main Steam Line Drain	2	15	C	SC
1	Recirculation Loop Sample	2	NA	C	SC
3	Recirculation Pump Seal Purge	2	5	O	GC
3	O_2 Analyzer	20	NA	O	GC
2	Drywell Floor Drain Discharge	2	4	O	GC
2	Drywell Equipment Drain Discharge	2	4	O	GC
3	Drywell Purge Inlet	1	5	C	SC
3	Drywell Purge Outlet	3	5	C	SC
3	Torus Purge Outlet	3	5	C	SC
3	Drywell and Torus Nitrogen Makeup	2	NA	O	GC
4	RHR Shutdown Cooling Supply	2	22	C	SC
3	*Containment Compressor Suction	2	25	O	GC
3	Suppression Pool/Drywell and Suppression Pool Purge Inlet	2	5	O	GC
5	RWCU Supply	2	20	O	GC
5	RWCU Return	1	10	O	GC
6	Steam to HPCI Turbine	2	13	O	GC

TABLE 3.7-3 (Continued)

PRIMARY CONTAINMENT POWER OPERATED ISOLATION VALVES

Isolation Group (Note 1)	Valve Identification	Number of Power Operated Valves	Maximum Operating Time (Seconds)	Normal Position	Action on Initiating Signal
6***	HPCI Discharge to Feedwater	1	20	C	GC
6	Steam to RCIC Turbine	2	20	O	GC
6***	RCIC Discharge to Feedwater	1	15	C	GC
8	Condensate from HPCI	2	NA	O	GC
8**	Condensate from RCIC	2	NA	O	GC
3	*Containment Compressor Discharge	3	NA	O	GC
7	*Reactor Building Closed Cooling Water Supply/Return	2	20	O	GC
7	*Well Cooling Water Supply/Return	4	NA	O	GC
9	HPCI/RCIC Exhaust Vacuum Breaker	2	10	O	GC
3	Post-Accident Sampling Liquid Sample Return	2	NA	C	SC
3	Post-Accident Sampling Jet Pump Sample	4	NA	C	SC

*Due to plant operational limitations, these valves will be subject to the requirements of 4.7.D.1.a only.

**Low-Low Water Level Only

***These valves close only upon sensing closure of their respective turbine steam supply or turbine stop valve closure.

NOTES FOR TABLE 3.7-3

1. Isolation Signals are as follows:

Group 1:

The valves in Group 1 are closed upon any one of the following conditions:

1. Reactor vessel low-low-low water level.
2. Main steam line high radiation.
3. Main steam line high flow.
4. Main steam line tunnel high temperature.
5. Low main steam line pressure at turbine inlet (run mode only).
6. Main condenser low vacuum.

Group 2:

The valves in Group 2 are closed upon any of the following conditions:

1. Reactor vessel low water level.
2. High drywell pressure.

Group 3:

The valves in Group 3 are closed upon any of the following conditions:

1. Reactor low water level.
2. High drywell pressure.
3. High/low radiation - reactor building ventilation exhaust plenum or refueling floor.

Group 4:

The valves in Group 4 are closed upon any one of the following conditions:

1. Reactor low water level.
2. High drywell pressure.
3. Reactor pressure above 135 psig.

NOTES FOR TABLE 3.7-3 (Continued)

Group 5:

The valves in Group 5 are closed upon low reactor water level, or upon a signal representing a line break in the reactor water cleanup system.

Group 6:

The valves in Group 6 are closed upon any signal representing a steam line break in the HPCI system's or RCIC system's respective steam line.

Group 7:

The valves in Group 7 are closed upon low-low-low reactor water level signal. (Note: The level sensors utilized for this function are part of the core and containment cooling logic.)

Group 8:

The valves in Group 8 are closed upon any of the following conditions:

1. Reactor vessel low-low water level
2. High drywell pressure

Group 9:

The valves in Group 9 auto-isolate on the combined trip of both reactor steam supply low pressure (PS-N001A-D) and high drywell pressure (PSE11-N011A-D).

KEY: O = Open
C = Closed
SC = Stays Closed
GC = Goes Closed

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: 0.0% Retained on #8 ASTM E11 Sieve: 5.0% maximum Through #8, retained on #12 Sieve: 40% to 60% Through #12, retained on #16 Sieve: 40% to 60% Through #16 ASTM E11 Sieve: 5.0% maximum Through #16 ASTM E323 Sieve: 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.

3.7 & 4.7 BASES:

A.1, 2 and 3 Primary Containment

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. If primary containment integrity cannot be maintained as required by Specification 3.7.A.2, the reactor operator is required to follow Specification 3.7.A.2.d. This specification requires that the reactor operator take action to restore primary containment within 1 hour from the time of initial loss. If, after 1 hour, the restoring of primary containment integrity is not successful, the reactor operator has 12 hours to be in the Hot Shutdown Condition and an additional 24 hours to be in the Cold Shutdown Condition.

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 minimum volume of 58,900 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 Humboldt 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 pool 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 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. (5)

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 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 (L_a) is 2.0%/day at a peak accident pressure (P_a) 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. Therefore, the specified primary containment leak rate is conservative and provides additional margin between expected offsite doses and 10 CFR 100 guidelines.

*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 (P_a).

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 provides assurance that components most likely to undergo degradation between Type A tests maintain leaktight integrity.

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).

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.

The interiors of the drywell and suppression chamber are coated to prevent rusting and for ease of decontamination. The inspection of the coating once per operating cycle assures the coating is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.

The water in the suppression chamber is used only for cooling in the event of an accident and for meeting the requirements of operational transients; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

A.3 and 4 Vacuum Relief

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. One valve may be inoperable, either non-fully closed or inoperable for opening subject to the requirements as stated in Specifications 3.7.A.4.b, 3.7.A.4.c, 4.7.A.4.b, and 4.7.A.4.c. If these specifications cannot be met, the reactor coolant system is brought to a condition where vacuum relief is not required.

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.

A.5 Inerting (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 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 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 twice a week the oxygen concentration will be determined as added assurance.

A.6 Post LOCA Atmosphere Dilution (CAD)

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 once per operating cycle. 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.

B. and C. Standby Gas Treatment System and Secondary Containment

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.

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 clean up the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance, and reactor operation or refueling operation may continue while repairs are being made. 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 less than 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

99.9 percent 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.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 11 inches of water at the system design flow rate 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 at least once per operating cycle 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 11 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. If one standby gas treatment system is inoperable, the other system must be tested daily. This substantiates the availability of the operable system and thus reactor operation or refueling operation can continue for a limited period of time.

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. 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.

D. 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.

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.

E. 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 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 assures that the MSIV-LCS system will be available in the remote possibility of a LOCA. An annual capacity test of the blowers and an annual initiation of the entire system assure that the MSIV-LCS system meets its design criteria. Allowance of thirty days to return a MSIV-LCS subsystem or blower to an operable status allows operational flexibility while maintaining protective capabilities.

F. 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 & 4.7 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 Uprate 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 50.54, Appendix J, Reactor Containment Testing Requirements, Federal Register, August 27, 1971.

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

6.11.3 UNIQUE REPORTING REQUIREMENTS

Special reports shall be submitted to the Director of Inspection and Enforcement Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification.

a. Reactor vessel base, weld and heat affected zone metal test specimens (Specification 4.6.A.2).

b. I-131 dose equivalent exceeding 50% of equilibrium value (Specification 4.6.B.1.h).

c. Inservice inspection (Specification 4.6.G).

d. Reactor Containment Integrated Leakage Rate Test (Specification 4.7.A.2.h).

e. deleted

f. Fire Protection Systems (Specifications 3.13.A.3, 3.13.B.2, 3.13.B.3, 3.13.C.3, and 3.13.D.3).

g. deleted