October 26, 1994

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Mr. Lee Liu
Chairman of the Board and
Chief Executive Officer
IES Utilities Inc.
Post Office Box 351
Cedar Rapids, IA 52406

SUBJECT: AMENDMENT NO. 201 TO FACILITY OPERATING LICENSE NO. DPR-49 - DUANE ARNOLD ENERGY CENTER (TAC NO. M83088)

Dear Mr. Liu:

The Commission has issued the enclosed Amendment No. 201 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center. This amendment consists of changes to the Technical Specifications in response to your application dated March 27, 1992, as supplemented on January 6, May 27 and October 20, 1994.

The amendment revises the Technical Specifications by changing the limiting conditions for operation and surveillance requirements for primary containment integrity, secondary containment integrity, and other systems and equipment of Section 3.7, Containment Systems. Limiting conditions for operation and surveillance requirements for drywell average air temperature and secondary containment automatic isolation dampers were also added.

A copy of the related Safety Evaluation is also enclosed. Notice of issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,
Original signed by Anthony H. Hsia

Anthony H. Hsia, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosures: 1. Amendment No. 201 to License No. DPR-49
2. Safety Evaluation

cc w/encls: See next page
*See previous concurrence

LA:PD3-3:DRPW	*PM:PD3-3:DRPW	*D:SCSB	*D:PD3-3	*OGC-WF1
MRushbrook	RPulsifer/bam	RBarrett	JHannon	CMarco
10/24/94	08/25/94 10/24/94	09/12/94	09/14/94	09/20/94

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Mr. Lee Liu
IES Utilities Inc.

Duane Arnold Energy Center

cc:

Jack Newman, Esquire
Kathleen H. Shea, Esquire
Newman, Bouknight and Edgar, P.C.
1615 L Street, N. W.
Washington, D. C. 20036

Chairman, Linn County
Board of Supervisors
Cedar Rapids, Iowa 52406

IES Utilities Inc.
ATTN: David L. Wilson
Plant Superintendent, Nuclear
3277 DAEC Road
Palo, Iowa 52324

Mr. John F. Franz, Jr.
Vice President, Nuclear
Duane Arnold Energy Center
3277 DAEC Road
Palo, Iowa 52324

Mr. Keith Young
Manager, Nuclear Licensing
Duane Arnold Energy Center
3277 DAEC Road
Palo, Iowa 52324

U. S. Nuclear Regulatory Commission
Resident Inspector's Office
Rural Route #1
Palo, Iowa 52324

Regional Administrator, RIII
U. S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, Illinois 60532-4531

Mr. Stephen N. Brown
Utilities Division
Iowa Department of Commerce
Lucas Office Building, 5th floor
Des Moines, Iowa 50319

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

IES UTILITIES INC.
CENTRAL IOWA POWER COOPERATIVE
CORN BELT POWER COOPERATIVE

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 201
License No. DPR-49

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by IES Utilities Inc., et al., formally known as Iowa Electric Light and Power Company, dated March 27, 1992, as supplemented on January 6 and May 27, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-49 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 201, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of the date of issuance and shall be implemented within 120 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Anthony H. Hsia

Anthony H. Hsia, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of issuance: October 26, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 201

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

LIST OF AFFECTED PAGES

REMOVE	INSERT
iii	iii
iiia	iiia
vi	vi
1.0-4	1.0-4
3.2-26	3.2-26
3.2-27	3.2-27
3.5-10a	3.5-10a
3.5-16	3.5-16
3.7-1 through 3.7-50	3.7-1 through 3.7-43
6.11-5	6.11-5

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.

Table 3.2-D
RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATING MODES	ALARM/TRIP SETPOINT	VALVE(s) OPERATED BY SIGNAL	ACTION
Offgas Post-Treatment Radiation Monitors	1	*	(a)	(a)	50
Offgas Pre-Treatment Radiation Monitors	1	*	(b)	NA	51
Main Steam Line Radiation Monitors	2	**	≤ 3X Normal Full Power Background	(c)	**

* When the offgas system is operating.
** Refer to Specification 3.7.M.

(a) The monitors shall be set to initiate immediate closure of the charcoal bed bypass valve and the air ejector offgas isolation valve at a setting equivalent to or below the dose rate limits in ODAM Section 6.2.2.1.

(b) 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.

(c) Trips Mechanical Vacuum Pump which results in a subsequent isolation of the Mechanical Vacuum Pump suction valves.

ACTION

RADIATION MONITORING INSTRUMENTATION

ACTION 50 - With the number of OPERABLE channels less than required by the Minimum Channels Operable requirement, gases from the steam air ejector offgas system may be released to the environment for up to 72 hours provided (1) the charcoal bed of the offgas system is not bypassed, and (2) the offgas stack noble gas activity monitor is operable.

Otherwise, be in at least HOT STANDBY within the following 24 hours.

ACTION 51 - With the number of OPERABLE channels less than required by the Minimum Channels Operable requirement, 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.

Table 4.2-D

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	SOURCE CHECK	OPERATING MODES FOR WHICH SURVEILLANCE REQUIRED
Offgas Post-Treatment Radiation Monitors	D	Q**	R	M	*
Offgas Pre-Treatment Radiation Monitors	D	Q**	R	M	*
Main Steam Line Radiation Monitors	Once/shift	Q(a)	R	R	(b)

* When the offgas system is operating

** The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

a. Instrument indicates measured levels above the alarm/trip setpoint
b. Instrument indicates a downscale failure
c. Instrument controls not set in the operate mode.

(a) This channel functional test will consist of injecting a simulated electrical signal into the measurement channels.

(b) Refer to Specification 3.7.M.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

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.

Consequently, loss of margin should be avoided and the equipment maintained in a state of OPERABILITY, thus a 30-day out-of-service time is chosen for one loop of each (suppression pool and drywell) spray being inoperable.

For the RHRSW system, having one pump out of service degrades the system but sufficient redundancy remains to support the safety function; thus, a 30-day out-of-service time is appropriate. If one loop is out of service, or one RHRSW pump in each loop is out of service, reactor operation is permitted for seven days, as the system has lost its required redundancy. The surveillance requirements, including In-Service Testing, 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 REQUIREMENTS

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 REQUIREMENTS

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 REQUIREMENTS

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 nonoperational and secured in the

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

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 REQUIREMENTS

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 REQUIREMENTS

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 flow path.*

* Penetrations isolated to satisfy these requirements may be reopened on an intermittent basis under administrative control.

SURVEILLANCE REQUIREMENTS

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

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.

SURVEILLANCE REQUIREMENTS

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 REQUIREMENTS

C. Drywell Average Air Temperature

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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
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 closed 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	SURVEILLANCE REQUIREMENTS
E. Drywell - Pressure Suppression Chamber Vacuum Breakers	E. Drywell - Pressure Suppression Chamber Vacuum Breakers
1. Six drywell-pressure suppression chamber vacuum breakers shall be OPERABLE and seven drywell-pressure suppression chamber vacuum breakers shall be closed at all times when PRIMARY CONTAINMENT INTEGRITY is required.*	1. Each drywell-pressure suppression chamber vacuum breaker shall be verified closed at least once per 7 days.
2. If one of the required six 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.	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. 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 closed position indicators of any drywell-pressure suppression chamber vacuum breaker inoperable:	3. Once/cycle, each drywell-pressure suppression chamber vacuum breaker shall be visually inspected to insure proper maintenance and operation.
a. Verify the vacuum breaker's other closed position indicator OPERABLE within 2 hours and at least once per 14 days thereafter or,	4. A leak test of the drywell to suppression chamber structure shall be conducted once per operating cycle.
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.	
* Except when the vacuum breaker(s) are performing their intended function.	

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 REQUIREMENTS

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 REQUIREMENTS
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

d. If the suppression pool average water temperature is ≥ 110°F, the reactor shall be scrammed.

e. If the suppression pool average water temperature is ≥ 120°F, depressurize the reactor to less than 200 psig within 12 hours.

SURVEILLANCE REQUIREMENTS

d. A visual inspection of the suppression chamber interior, including water line regions, shall be made once per operating cycle.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
H. Containment Atmosphere Dilution	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.	1. The post-LOCA containment atmosphere dilution system shall be functionally tested annually.
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.	2. The volume in the N_2 storage bank shall be recorded weekly.
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.	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 REQUIREMENTS

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	SURVEILLANCE REQUIREMENTS
J. Secondary Containment	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.	1. Secondary containment surveillance shall be performed as indicated below:
a. The reactor is subcritical and Specification 3.3.A is met.	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.
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.	

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.*

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.

* Penetrations isolated to satisfy these requirements may be reopened on an intermittent basis under administrative control.

SURVEILLANCE REQUIREMENTS

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 REQUIREMENTS
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 of 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.

LIMITING CONDITIONS FOR OPERATION

c. Fans shall be shown to be capable of operation from 1800 cfm to the flow range of 3600-4000 cfm.

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.

SURVEILLANCE REQUIREMENTS

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.

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 REQUIREMENTS

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 provides 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 developed 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 six 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), a Completion Time of

72 hours is allowed to restore the vacuum breaker 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 communication 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, IES Utilities Inc. 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 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 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

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.

3.7.K and 4.7.K BASES

Secondary Containment Automatic Isolation Dampers

The function of the secondary containment isolation valves/dampers, 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 secondary containment isolation valves/dampers 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 secondary containment isolation valves/dampers 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 secondary containment isolation valves/dampers are inoperable, either the inoperable valve/damper 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 secondary containment isolation valve/damper, a closed manual valve/damper, or a blind flange.

Demonstrating the isolation capabilities of each power-operated and automatic secondary containment isolation valve/damper is required to demonstrate OPERABILITY. The simulated automatic initiation ensures that the valve/damper 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 and pressure drop should be determined annually to show system performance capability. Annual demonstration of air distribution is not required. Changes to the flow distribution would be expected to occur after changes are made to the filters or filter housing rather than on a time-dependent basis.

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.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.11.3 <u>UNIQUE REPORTING REQUIREMENTS</u>

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. deleted

c. Inservice inspection (Specification 4.6.G.).

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

e. deleted

f. deleted

g. deleted

h. Radioactive Liquid or Gaseous Effluent - calculated dose exceeding specified limit (ODAM Sections 6.1.3, 6.2.3 and 6.2.4).

i. Off-Gas System inoperable (ODAM Section 6.2.5).

j. Measured levels of radioactivity in an environmental sampling medium determined to exceed the reporting level values of ODAM Table 6.3-3 when averaged over any calendar quarter sampling period (ODAM Section 6.3.2.1).

k. Annual dose to a MEMBER OF THE PUBLIC determined to exceed 40 CFR Part 190 dose limit (ODAM Section 6.3.1.1).

l. Radioactive liquid waste released without treatment when activity concentration is equal to or greater than 0.01μci/ml (ODAM Section 6.1.4.1).

m. Explosive Gas Monitoring Instrumentation Inoperable (Specification 3.2.I.1).

n. Liquid Holdup Tank Instrumentation Inoperable (Specification 3.14.B.1).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 201 TO FACILITY OPERATING LICENSE NO. DPR-49

IES UTILITIES INC.
CENTRAL IOWA POWER COOPERATIVE
CORN BELT POWER COOPERATIVE

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 INTRODUCTION

By application dated March 27, 1992, IES Utilities Inc, formerly known as Iowa Electric Light and Power Company, requested an amendment to the Duane Arnold Energy Center facility operating license. The proposed amendment requested revision of the Technical Specifications (TS) limiting conditions for operation (LCOs) and surveillance requirements (SRs) for primary containment integrity, secondary containment integrity and other systems and equipment of TS Section 3.7 "Containment Systems" to improve their clarity and consistency with the Standard Technical Specifications (STS). The staff's technical assistance contractor reviewed the application and provided a Technical Evaluation Report (TER) which was forwarded to the licensee by letter dated November 12, 1993. A copy of the TER is attached. The TER identified deficiencies which the staff agreed should be addressed by the licensee prior to issuance of an amendment.

By letter dated January 6, 1994, the licensee responded to the deficiencies cited in the Attachment. In a subsequent letter dated March 30, 1994, the licensee advised the staff that a revised application was being prepared that would not only incorporate changes resulting from the deficiencies cited in the TER, but would add additional changes. The licensee subsequently, by letter dated May 27, 1994, submitted a revised application. The staff has reviewed the TER and concurs with the contractor's conclusions regarding acceptability of the changes proposed in the March 27, 1992, application. Section 2.0 below is limited to discussion and evaluation of: (a) the licensee's response to the deficiencies, and (b) the additional changes.

2.0 Discussion And Evaluation

2.1 Deficiencies Cited in TER

This section discusses the discrepancy items identified in the attached TER.

2.1.1 Secondary Containment Negative Pressure

Deficiency cited in TER: Section 6.5 of the DAEC Updated Final Safety Analysis Report (UFSAR) states that the secondary containment is maintained at a pressure of -¼"w.g. (¼-inch of water, gauge negative) during normal operation. The TER noted that there is no requirement in either the existing

TS or proposed TS to periodically verify the negative pressure during operation. The TER also noted that the TS do not specify or require periodic surveillance testing to verify standby gas treatment system (SGTS) capability to achieve -¼"w.g. pressure within a specific drawdown time limit. Section 50.36 of 10 CFR requires that Limiting Conditions for Operation (LCO) and Surveillance Requirements (SRs) be specified in TS for equipment required for safe operation.

Licensee Response: The licensee responded that a -¼" w.g. pressure, and a specific drawdown time are not design or licensing basis requirements for the DAEC facility.

Staff Evaluation: The TS for facilities having a secondary containment for which a fission product control capability is credited in the analyses of radiological consequences of design basis accidents typically include surveillance test requirements to periodically ensure the operability of the equipment needed to establish and maintain a negative pressure in the secondary containment. The purpose of a negative secondary containment pressure is to preclude ex-filtration. Ex-filtration is the direct release of primary containment fission product leakage without cleanup by the SGTS HEPA and charcoal filtration equipment. Radiological dose consequence calculational methodology does not credit the secondary containment fission product control function during periods when the pressure is positive with respect to outside pressure.

DAEC is one of a group of early BWR facilities, whose design and licensing bases, are not typical of similar, but later facilities. For these early BWRs, the secondary containment is normally maintained at a negative pressure, and, in the event of a accident, it is assumed that the negative pressure is maintained during the period when the secondary containment isolates, and the SGTS begins operating. The DAEC radiological dose models for these facilities do not assume a period of secondary containment ex-filtration and the TS do not include either: (1) a surveillance requirement for periodic verification of secondary containment negative pressure during normal operation, or (2) a requirement that drawdown be demonstrated within a specific, analytically-based time interval during SGTS testing. The question arose as to whether the proposed DAEC amendment should be denied, because the proposed changes would not fully upgrade the secondary containment surveillance requirements to current standards.

Although the staff found that the proposed changes would not fully upgrade the existing surveillance requirements, the staff recognized that issues relating to the improvement of TS have already been addressed.

Licensees of earlier facilities, such as DAEC, are being encouraged to upgrade the TS of their facilities by voluntarily submitting improved TS amendment requests (Ref: "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," 58 FR 39132 Published July 22, 1993). Accordingly, additional SGTS surveillance test requirements that would: (1) demonstrate a drawdown time capability during SGTS surveillance testing, and (2) periodically verify, during normal operation, that the secondary containment is being maintained at -0.25 "w.g. subatmospheric pressure, need not be included as part of this amendment. The staff, therefore, finds the licensee's response acceptable.

2.1.2 Secured Check Value

Deficiency Cited in TER: The TER noted that the proposed Bases for new TS Sections 3.7.B and 4.7.B identify the use of "a check valve inside primary containment with flow through the valve secured," as an isolation barrier is not consistent with the associated LCO.

Licensee Response: The licensee replied that the LCO, including an associated footnote, would be revised to delete text identifying the specific methods of isolating penetrations, and that such information would be specified in the BASES only. The licensee's revised May 27, 1994, application eliminates the inconsistency by deleting text from the LCO that would describe how a penetration is to be isolated. The reader must thus defer to the TS Bases for this additional information.

Staff Evaluation: The revised application resolves the inconsistency, allowing use of a "check valve inside containment with flow through the valve secured," as a means of isolating an open containment penetration in the event one or more of its isolation valves becomes inoperable. The staff considers a check valve inside containment to be an acceptable isolation barrier, if it is provided with means for positive closure. The use of additional descriptive text provided in TS Bases or Definitions (or in the UFSAR), is an acceptable means of clarifying specific TS operability requirements.

The staff concludes that the licensee's response is acceptable, and the TER discrepancy is resolved.

2.1.3 Periodic Testing Of HEPA Filter Air Flow Distribution

Deficiency Cited in TER: The licensee's proposed TS change added a requirement to perform an air distribution surveillance test following each complete or partial replacement of an HEPA filter bank or any structural maintenance on the HEPA filter housing, and deleted a requirement for annual periodic testing. The application did not include changes to the associated Bases needed for consistency.

Licensee Response: The revised application includes proposed changes to the Bases to make them consistent with the proposed changes to the SR.

Staff Evaluation: As indicated in the TER, the replacement of the requirement for an annual airflow distribution test by a requirement for an airflow distribution test following each complete or partial replacement of a HEPA filter bank or any structural maintenance on the filter housing is acceptable. The licensee has proposed consistent SR and Bases changes. The staff concludes that the licensee's response is acceptable, and the TER discrepancy is resolved.

2.2.3 Requirement For Drywell Vacuum Breakers To Be Closed While Performing Their Intended Function

Additional TS Change Added to Revised Application: TS 3.7.E.3 presently requires that the drywell vacuum breakers be closed at all times during the applicable modes of operation. An additional proposed change would provide an exception for occasions when a vacuum breaker opens in the performance of its intended function.

Staff Evaluation: Occasionally, during normal plant operations such as inerting or pressure adjustment, a vacuum breaker may be subjected to ΔP conditions for which it is intended to open. Such occasion need not invoke entry into the required action statement and associated reports and notifications, since no malfunction or degradation of safety systems has occurred. The proposed change is therefore acceptable.

2.2.4 Methods Of Isolating Secondary Containment Automatic Isolation Dampers

Additional TS Change Added to Revised Application: TS 3.7.K.2 would be revised to delete specific details on methods to isolate secondary containment penetrations. The information would be provided in the Bases only.

Staff Evaluation: This change is being made for consistency with 2.1.2 above. In 2.1.2 above, the staff concludes that specific information on means to isolate a primary containment penetration may be identified in the TS Bases, in lieu of in the LCO. This change is similarly acceptable for secondary containment isolation.

2.2.5 Editorial Correction

Additional TS Change Added to Revised Application: TS 4.7.L.1.a states:

> 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.

The licensee proposes to change it to read:

> Annually it shall be demonstrated that pressure drop across the combined high efficiency and charcoal filters is less than 11 inches *of* water in the flow range of 3600 to 4000 cfm.

Staff Evaluation: The proposed change is a typographical correction only. It would have no effect on the associated LCO or SR and is therefore acceptable.

2.2.6 Secondary Containment Isolation Devices

Additional TS Change Added to Revised Application: In the 3/4.7.K Bases, references to "SCIVs" would be changed to "secondary containment isolation valves/dampers."

Staff Evaluation: The proposed change would bring the Bases terminology into consistency with the LCO/SR terminology with no effect on actual operability and surveillance requirements of the associated safety systems. The change is therefore acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Iowa State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATIONS

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (59 FR 34665). Accordingly, the 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.

5.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: W. Long

Date: October 26, 1994

Attachment:
Technical Evaluation Report
prepared by SCIENTECH, Inc.

Technical Evaluation Report

Prepared By

SCIENTECH, Inc.

Related to Request for Technical Specification Change

IOWA ELECTRIC LIGHT AND POWER COMPANY
CENTRAL IOWA POWER COOPERATIVE
CORN BELT POWER COOPERATIVE

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

INTRODUCTION

By letter dated March 27, 1992, Iowa Electric Light and Power Company, the licensee, requested changes to the Technical Specifications (TS) for the Duane Arnold Energy Center (DAEC). On August 13, 1993, SCIENTECH, Inc. was tasked by the NRC to review the requested changes to the DAEC TS and prepare a Technical Evaluation Report (TER). A draft TER was provided to the NRC for review on September 29, 1993. The report was finalized following discussions with the NRC staff on October 13, 1993.

DISCUSSION

The changes requested by the licensee would revise the limiting conditions for operation and the surveillance requirements for primary containment integrity, secondary containment integrity, and associated systems and equipment addressed in section 3.7 of the DAEC Technical Specifications, to improve their clarity and consistency with the Standard Technical Specifications (STS). The requested changes also would add limiting conditions for operation and surveillance requirements for drywell average air temperature and secondary containment automatic isolation dampers to the existing TS.

The requested changes resulted from an independent evaluation of the DAEC TS completed in 1991, conducted as part of DAEC TS Improvement Program, which included comparisons of the DAEC TS with TS from peer plants, Standard Technical Specifications, and the draft Improved Technical Specifications (NUREG-1433). This evaluation identified a number of improvements that should be made to the DAEC TS, including the addition of specifications related to the drywell temperature and secondary containment isolation dampers. For comparison with the Standard Technical Specifications, the licensee used the Technical Specifications issued by the NRC for the Hope Creek Generating Station (NUREG-1202) in July 1986. These are the latest TS issued by the NRC for an operating BWR-4 plant and comparison of the DAEC TS to the Hope Creek TS is acceptable to the NRC staff. Accordingly, we have used NUREG-1202 as an example of the STS for comparison with the DAEC TS.

EVALUATION

We have performed an item by item evaluation of each of the changes requested by the licensee. The results are presented in the enclosure. A large number of the changes are administrative or editorial in nature, resulting from a reorganization of the TS Section 3.7 material to track more closely the organization of the STS. These have no effect on plant safety. A number of proposed changes would result in no change or no change in intent from the existing limiting conditions for operation, action statements, and surveillance requirements of the DAEC TS; they merely move these requirements into the revised organizational format for Section 3.7, and do not impact plant safety. In many instances, where plant-specific considerations allow, the proposed changes would incorporate the limits of the STS in place of those in the existing DAEC TS. Finally, there are several instances where the proposed changes are not in accord with either the STS or the existing DAEC TS. In these cases, we have analyzed the changes and found them to be acceptable. However, in several instances, as noted in this TER and the evaluation matrix, we have identified shortcomings which we believe the licensee should be urged to correct.

For purposes of shorthand identification of the types of changes, the enclosure indicates for each change the assigned review category or categories. These review categories are defined as follows :

1. Administrative/editorial change and therefore acceptable.

2. No change from the intent of the existing TS. Does not result in a decrease in safety from the existing TS and therefore acceptable.

3. Consistent with the STS and therefore acceptable.

4. Not in accordance with the STS or the existing DAEC TS, but has been analyzed and found acceptable.

5. Unacceptable without additional justification.

A problem arises when attempting to compare the DAEC TS with the STS due to variations in the definitions of Operational Conditions. These differences are compared in the following table:

Operational Condition Definitions

Standard Tech Specs	Duane Arnold
Power Operation - Mode switch in RUN position Reactor coolant at any temperature	**Reactor Power Operation** Mode switch in STARTUP or RUN Reactor critical and above 1% rated power
Startup Mode switch in STARTUP / HOT STANDBY Reactor coolant at any temperature	**Hot Standby Condition** Mode switch in STARTUP / HOT STANDBY Reactor coolant temperature >212°F Reactor pressure <1055 psig
Hot Shutdown Mode switch in SHUTDOWN Reactor coolant >200°F	**Hot Shutdown** Reactor in SHUTDOWN mode Reactor coolant >212°F
Cold Shutdown Mode switch in SHUTDOWN Reactor coolant ≤200°F	**Cold Condition** Reactor coolant ≤212°F
Refueling Mode switch in SHUTDOWN or REFUEL Reactor coolant ≤140°F	**Cold Shutdown** Reactor in SHUTDOWN mode Reactor coolant ≤212°F Reactor vessel vented to atmosphere

When attempting to adjust the DAEC TS to the STS format, these variations in definitions of the operating conditions result in specified temperature limits for DAEC that are slightly higher than the limits specified in the STS. However, the 212°F used by DAEC has been previously reviewed and approved by the NRC staff, and there is no justification caused by this requested change for reducing this to the 200°F value used in the STS.

In each instance, the proposed changes would result in TS which are as good as or better than the existing DAEC TS. The addition of requirements regarding the drywell average air temperature and secondary containment isolation dampers are distinct improvements from the existing TS. The licensee's letter states that as of the time of the submittal of the request, an acceptable method of performing surveillance testing for actuation times on the secondary containment isolation dampers had not been developed, but that a test was under development and would be submitted as a separate request at a later date. While the absence of a suitable surveillance test for the isolation damper actuation times represents a deficiency in the proposed TS, we do not view it as critical at this juncture. The addition of requirements in the revised TS on the secondary containment isolation dampers is a considerable improvement in and of itself. The surveillance test can be added later. However, the licensee should be encouraged to remedy this deficiency at an early date.

In addition, we have identified the following deficiencies which are not cause for rejection of the proposed TS change, but which we believe should be corrected by the licensee at an early date.

1. While the DAEC FSAR Section 6.5 states that the secondary containment is maintained at a negative 1/4-inch of water pressure during normal operation, there is no requirement in either the existing or proposed TS to periodically verify this negative pressure. Further, the TS do not specify or require testing to verify the maximum time for SGTS operation to achieve the 1/4-inch of water vacuum in secondary containment. These are shortcomings in the TS which the licensee should be urged to correct.

2. The Bases for new TS sections 3.7.B and 4.7.B include an added discussion of the actions to be taken in the event that one or more primary containment isolation valves are inoperable. In general, this is an improvement. However, the discussion includes use of "a check valve inside primary containment with flow through the valve secured" as an acceptable isolation barrier. These words are not consistent with TS 3.7.B and should be corrected.

3. The Bases for Section 3.7.L and 4.7.L in the proposed revised TS, which have not been changed from the Bases of the existing DAEC TS, state that "...air distribution (across the HEPA filter bank) should be determined annually...." However, proposed revised TS 4.7.L.1.c requires an air distribution demonstration to be performed "after each complete or partial replacement of a HEPA filter bank or after any structural maintenance on the system housing." The Bases should be revised to support the revised requirement for the air flow demonstration.

CONCLUSION

Overall, we conclude that the revised TS Section 3.7 would be a distinct improvement over the existing DAEC TS. The revised TS would track the STS more closely and would be more precise and easier to understand. We recommend that the NRC accept the requested changes to the DAEC TS and give consideration to the three deficiencies discussed above.

Enclosure: Evaluation Matrix -- Proposed Changes to DAEC Technical Specifications Revisions to Section 3.7 (RTS 246)

EVALUATION MATRIX

PROPOSED CHANGES TO THE DAEC

TECHNICAL SPECIFICATIONS

CONTAINMENT SYSTEMS

Section 3.7 (RTS 246)

Prepared By:

SCIENTECH, Inc.

Contract Number	NRC-03-93-031
Task Order Number	93-06

DUANE ARNOLD ENERGY CENTER (DAEC)
TECHNICAL SPECIFICATION CHANGE (RTS-246)
REVISIONS TO TS SECTION 3.7 -- CONTAINMENT SYSTEMS

TS Page No.	Proposed Change	Evaluation of the Proposed Change	Review Category
iii	The Table of Contents has been revised to reflect that Section 3.7, "Containment Systems," has been renamed "Plant Containment Systems." Subsections A-D have been deleted and replaced with subsections A-M which correspond to the revision to TS section 3.7. The Surveillance Requirements and page numbers have also been revised accordingly.	This is an administrative change to conform the Table of Contents for Section 3.7 to the revised contents of the section as proposed by this requested change. It does not affect plant safety and is therefore acceptable.	1
vi	In the List of Tables, the page at which Table 4.7-1 appears has been revised to correspond to the pagination of TS section 3.7.	Revised pagination to reflect the proposed changes results in Table 4.7-1 appearing on a different page. This is an administrative change and is acceptable.	1
1.0-4	Definition 15, "Primary Containment Integrity", has been revised to be more consistent with STS definition 1.31, "Primary Containment Integrity", Subsection a. and c. of previous TS definition 15 have been replaced with new subsection a. which is identical to subsection a. of STS definition 1.31 except for specific discussions of the PCIV Table (which have been relocated to an Administrative Procedure) and incorporation of a statement allowing the valves to be opened to perform necessary operational activities. Subsection d. of previous TS definition 15 has been re-designated as subsection c.	This proposed change would bring the DAEC definition of Primary Containment Integrity more nearly into conformance with the Standard Technical Specifications (STS) definition. Adoption of this revised definition would not result in any relaxation of the requirements for closure of primary containment penetrations from that required by the existing DAEC Technical Specifications (TS), and would not result in any decrease in the integrity of the primary containment. The change is, therefore, acceptable.	2

TS Page No.	Proposed Change	Evaluation of the Proposed Change	Review Category
1.0-4	Definition 16, "Secondary Containment Integrity", has been revised for clarity and to be more consistent with STS definition 1.38, "Secondary Containment Integrity". Subsection c. of previous TS definition 16 has been replaced with subsection a. of previous TS definition 1.38 except for the discussion of a secondary containment isolation valve/ damper table. The list of applicable valves/dampers will be located in an Administrative Procedure. Additionally, a note allowing the valves/dampers to be opened to perform operational activities has been added. The term "OPERABLE" has been capitalized in subsections b. and c. to dénote that it is a term defined in TS section 1.0.	This proposed change would bring the DAEC definition of Secondary Containment Integrity more nearly into conformance with the STS definition. Adoption of this revised definition would not result in any relaxation of the requirements for closure of secondary containment penetrations from that required by the existing DAEC TS, and would not result in any decrease in the integrity of the secondary containment. The change is, therefore, acceptable.	2
3.2-3	The reference to specification 3.7.B in TS section 3.2.D.2, "Reactor Building Isolation and Standby Gas Treatment System," has been changed to section 3.7. This more general reference reflects the re-organization of TS section 3.7.	Page 3.2-3 was changed by Amendment 196 to the DAEC TS, issued April 14, 1993, after this amendment request was submitted by the licensee. This requested change, therefore, is no longer applicable.	NA
3.5-10a	The reference to specification 3.7.A.1 in TS section 3.5.G.4 has been changed to section 3.7. This more general reference reflects the re-organization of TS section 3.7.	This proposed change is administrative in nature and merely reflects the revisions in the organization of Section 3.7. It has no impact upon plant safety. It is, therefore, acceptable.	1
3.5-16	The reference to section 3.7.A.1 in the Bases to TS sections 3.5.B and 3.5.C has been changed to section 3.7. This more general reference reflects the re-organization of TS section 3.7.	This proposed change is administrative in nature and merely reflects the reorganization of Section 3.7 of the DAEC TS that would result from this amendment request. It is, therefore, acceptable.	1

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TS Page No.	Proposed Change	Evaluation of the Proposed Change	Review Category
3.7-1	The title of TS section 3.7.A has been changed from "Primary Containment" to "Primary Containment Integrity."	This change is administrative in nature and is necessary to accommodate the revised organization of Section 3.7. It does not affect plant safety and it is, therefore, acceptable.	1
3.7-1	Revised TS section 3.7.A.1 contains the Primary Containment LCO previously located in TS section 3.7.A.2. The reference to section 3.7.D.2 has been revised to 3.7.B.2. The term, "Primary Containment Integrity" has been capitalized to denote that it is a term defined in TS section 1.0.	This is an administrative and editorial change that moves the LCO for primary containment integrity from its position in the existing DAEC TS to a new position in Section 3.7 and adds capitalization to the term "Primary Containment Integrity" to indicate that it is defined in Section 1. No changes are made to the LCO itself, and the change is, therefore, acceptable.	1
3.7-1	The specifications in previous TS section 3.7.A. 1 are now located in TS section 3.7.G.	This is an administrative change necessitated by the reorganization of Section 3.7. The movement of the specifications to the new position does not affect plant safety and is acceptable.	1
3.7-1	New TS section 3.7.A.2 specifies what actions are to be taken when the requirements of Primary Containment Integrity are not met. The actions (previously located in TS section 3.7.A.8) have been revised and are now consistent with the actions of STS section 3.6.1.1.	The movement of the LCO is administrative in nature and is acceptable. The change to the LCO conforms the DAEC TS to the STS, results in a more precise statement of the requirements for maintaining Primary Containment Integrity, and does not in any way relax the requirements provided by the existing DAEC TS. The changes are, therefore, acceptable.	1, 3
3.7-1	The title of TS section 4.7.A has been changed from "Primary Containment" to "Primary Containment Integrity."	This is an administrative change, conforming the title to its counterpart in Section 3.7.A. It does not affect plant safety and is, therefore, acceptable.	1

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TS Page No.	Proposed Change	Evaluation of the Proposed Change	Review Category
3.7-1	Revised TS section 4.7.A.1 contains the Primary Containment Integrity surveillance requirements previously located in TS section 4.7.A.2. The term, "Primary Containment Integrity" has been capitalized to denote that it is a term defined in the TS section 1.0.	This is an administrative and editorial change that moves the surveillance requirements to verify primary containment integrity from their position in the existing DAEC TS to a new position in Section 4.7.A.1 and adds capitalization to the term "Primary Containment Integrity" to indicate that it is defined in Section 1. There are no changes to the surveillance requirements themselves. The change is, therefore, acceptable.	1
3.7-1	The subtitle of TS section 4.7.A.1.a, "Type A Test" is no longer underlined.	This is an editorial change and is acceptable.	1
3.7-1	In TS section 4.7.A.1.a. (1), the reference to TS section 4.7.A.2.a. (9) has been revised to 4.7.A.1.a. (8).	This is an administrative change necessitated by the revised organization of Section 3.7. The existing TS erroneously references 4.7.A.2.a. (9) instead of 4.7.A.2.a. (8). This change corrects the reference in the revised TS to 4.7.A.1.a. (8) . The changes are, therefore, acceptable.	1
3.7-2	The underlining of subtitles of TS sections 4.7.A.1.a. (7) - 4.7.A. 1.a. (9) and 4.7.A.1.b has been deleted.	These are editorial changes and are acceptable.	1
3.7-2	In TS section 4.7.A.1.a. (9), the reference to TS section 4.7.A.2. (a) (8) has been revised to 4.7.A.1.a. (8). The reference to TS section 4.7.A.2. (d) has been revised to 4.7.A.1.d.	These are administrative changes to conform to the revised organization of Section 3.7 and are acceptable.	1
3.7-3	The underlining of subtitles of TS sections 4.7.A.1.b. (1), 4.7.A.1.b. (2), 4.7.A.1.c, 4.7.A.1.d, and 4.7.A.1.d. (1) has been deleted.	These are editorial changes only and are acceptable.	1
3.7-4	The underlining of subtitles of TS sections 4.7.A.1.d. (2) - 4.7.A.1.d. (4) has been deleted.	These are editorial changes only and are acceptable.	1

TS Page No.	Proposed Change	Evaluation of the Proposed Change	Review Category
3.7-7	The note denoted by "*" has been deleted due to the changes to TS section 3.7.B.2. The note denoted by "**" has been changed to "*."	This is an administrative change necessitated by the revised wording of proposed Section 3.7.B.2.b. It has no affect on plant safety and is acceptable.	1
3.7-7	TS section 4.7.B. has been changed from "Standby Gas Treatment System" to "Primary Containment Power Operated Isolation Valves."	This is an administrative change necessitated by the proposed change in the title of Section 3.7.B. It is acceptable.	1
3.7-7	TS section 4.7.B.1 contains the Primary Containment Power Operated Isolation Valve surveillance requirements previously located in TS section 4.7.D.1. The note previously denoted by "*" has been changed to "#." The note previously denoted by "**" has been changed to "##."	These surveillance requirements are moved without change in content to the new location. This change is administrative in nature and is acceptable. The change in reference to the footnotes avoids possible confusion with the footnote to Section 3.7.B.2.b and is editorial in nature. It is acceptable.	1
3.7-7	In the note denoted by "#", the reference to TS section 4.7.D.1.a has been revised to 4.7.B.1.a.	This is an administrative change necessitated by the reorganization of the material in Section 3.7. It is acceptable.	1
3.7-7	In the note denoted by "##", the reference to TS section 4.7.D.1.b has been revised to 4.7.B.1.b. The word "suction" has been capitalized.	This is an administrative change necessitated by the reorganization of the material in Section 3.7. It is acceptable. In the proposed revised TS, the word "suction" has not been capitalized as stated in the amendment request. However, this does not affect the acceptability of the proposed change.	1
3.7-8	TS section 3.7.B.2.c (formerly section 3.7.D.2.c) has been revised to be consistent with action a.3 of STS section 3.6.3. The footnote "**" has been changed to "*."	The proposed revision clarifies the wording of this section and brings it into conformance with the wording of the STS. There is no change to the requirement of the LCO. It is, therefore, acceptable.	2, 3

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3.7-8	In TS section 3.7.B.3 (formerly 3.7.D.3), the references to TS sections 3.7.D.1 and 3.7.D.2 have been revised to 3.7.B.1 and 3.7.B.2 respectively. The requirement to "be in the Cold Shutdown condition within 24 hours" has been changed to "be in at least HOT SHUTDOWN within 12 hours and COLD SHUTDOWN in the following 24 hours." These shutdown requirements are consistent with the other shutdown requirements of this chapter.	The changes in the referenced TS sections are made to accommodate the revised organization of Section 3.7, and are acceptable. The revised wording of the requirement provides a more precise description of what is required, conforms the wording to the STS requirement, and does not result in any decrease in the protection afforded to the primary containment integrity. It is, therefore, acceptable.	1, 2, 3
3.7-8	The note denoted by "*" has been added and is identical to the note on TS page 3.7-7.	The addition of this footnote is an administrative change necessitated by a page change for 3.7.B.2.c. It represents no change from the requirement of the existing TS, and it is, therefore, acceptable.	1
3.7-8	TS sections 3.7.B.4, "Purging," and TS section 3.7.B.4.a contain the requirements previously located in TS section 3.7.A.9. The underlining of the subtitle of TS section 3.7.A.9 has been deleted.	This is an administrative change to accommodate the revised organization of Section 3.7. There is no change to the requirement from that in the existing TS and it is, therefore, acceptable.	1, 2
3.7-9	The title of TS section 3.7.C has been changed from "Secondary Containment to "Drywell Average Air Temperature."	This is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable.	1

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3.7-9	TS section 3.7.C.1 is a new specification for drywell average air temperature that is consistent with the drywell average air temperature LCO contained in STS 3.6.1.7. The specified applicability, however, is the same as specified in TS section 3.7.A.1, "Primary Containment Integrity" (without the exception for low power physics testing).	This new specification makes the DAEC TS conform more closely to the STS by the addition of an LCO on drywell temperature. The temperature limit is set at 135°F, which is the same as the STS temperature limit, and is the temperature used in the DAEC Design Basis Accident calculations. The LCO is applicable when the reactor is critical or when fuel is in the reactor vessel and the reactor temperature is above 212°F. The comparable STS requirement is that the drywell temperature LCO is applicable during Operational Conditions 1, 2 and 3. Operational Condition 3 specifies a reactor temperature >200°F. The proposed DAEC LCO thus is not quite as tight as the LCO for the STS. However, the DAEC LCO applicability for control of drywell temperature is the same as the DAEC LCO applicability for control of primary containment integrity, which previously has been reviewed by the staff and found acceptable. There is no reason to have tighter controls on drywell temperature applicability than on primary containment integrity. Thus, since this is an added requirement not present in the existing TS; since it conforms closely to the STS requirement, varying only in the specified reactor coolant temperature above which the LCO is applicable; and since its applicability is the same as for the previously approved applicability for primary containment integrity controls, we find the change acceptable.	4
3.7-9	TS section 3.7.C.2 contains the action statement for Specification 3.7.C.1. These actions are identical to the actions required by STS section 3.6.1.7.	This action statement conforms the DAEC TS to the action statement of the STS and is acceptable.	3

TS Page No.	Proposed Change	Evaluation of the Proposed Change	Review Category
3.7-9	The title of TS section 4.7.C has been changed from "Secondary Containment" to "Drywell Average Air Temperature."	This is an administrative change to conform the section title to the title its counterpart Section 3.7.C. It is acceptable.	1
3.7-9	TS section 4.7.C.1 contains drywell average air temperature surveillance requirements similar to the surveillance requirements of STS section 4.6.1.7. The reference to volumetric average contained in the STS surveillance requirement is located in the Bases to DAEC TS section 3.7.C.	The specified surveillance requirements are the same as those contained in the STS except that the locations for sampling to determine volumetric average drywell temperature are not specified. The Bases for Section 3/4.7.C discuss sampling at various elevations in the drywell to obtain a volumetric average. Sample points are not specified, but the stated intent is similar to that of the STS. Since the requirement for a drywell temperature limit is an added requirement that will enhance plant safety, we find this change acceptable.	4
3.7-10	The title of TS section 3.7.D has been changed from "Primary Containment Power Operated Isolation Valves" to "Pressure Suppression Chamber - Reactor Building Vacuum Breakers".	This is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable.	1
3.7-10	TS section 3.7.D.1 contains the LCO for the Pressure Suppression Chamber - Reactor Building Vacuum Breakers previously located in TS section 3.7.A.3.a. This section has been revised to be consistent with STS section 3.6.4.2 "Reactor Building - Suppression Chamber Vacuum Breakers". The specified applicability is the same as required in current TS section 3.7.A.3.a. The setpoint specified in present TS section 3.7.A.3.a has been relocated to surveillance requirement 4.7.D.3. The term, Primary Containment Integrity, has been capitalized.	This LCO has been modified to correspond to the wording of the STS. The intent of the LCO has not changed from that of the existing TS. The change is an improvement and is, therefore, acceptable. The relocation of the setpoint specification to Section 4.7.D.3 is consistent with the organization of the TS and is acceptable. Capitalization of the term "primary Containment integrity" has no affect on plant safety and is acceptable.	1, 3

TS Page No.	Proposed Change	Evaluation of the Proposed Change	Review Category
3.7-10	TS sections 3.7.D.2 - 3.7.D.4 contain the Action Statements for Specification 3.7.D.1. These actions, previously located in TS section 3.7.A.3.b. have been revised to be consistent with the actions of STS section 3.6.4.2. Specifically: • The actions of TS section 3.7.D.2 are identical to action a. of STS section 3.6.4.2. • The actions of TS section 3.7.D.3 are identical to action b. of STS section 3.6.4.2. • The actions of TS section 3.7.D.4 are identical to action c. of STS section 3.6.4.2.	The action statements covered by sections 3.7.D.2 through 3.7.D.4 include all actions of the STS regarding the Pressure Suppression Chamber - Reactor Building Vacuum Breakers. Together, these actions provide for improved controls on the vacuum breakers from that afforded by the existing TS. They are, therefore, acceptable.	3
3.7-10	The title of TS section 4.7.D has been changed from "Primary Containment Power Operated Isolation Valves" to "Pressure Suppression Chamber - Reactor Building Vacuum Breakers."	This is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable.	1
3.7-10	TS section 4.7.D.1 is a new surveillance requirement and is consistent with STS section 4.6.4.2.a.	This new requirement is consistent with the STS, provides for enhanced plant safety, and is acceptable.	3

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3.7-10	TS sections 4.7.D.2 and 4.7.D.3 contain the Pressure Suppression Chamber Reactor Building Vacuum Breakers surveillance requirements previously located in TS section 4.7.A.3.a. These surveillance requirements have been revised as follows: • TS section 4.7.D.2 now specifies that the position indication shall be verified as part of the quarterly cycling test. • TS section 4.7.D.3 now specifies that the opening setpoint of $\leq$ 0.5 psid shall be demonstrated. This setpoint was previously located in TS section 3.7.A.3.a.	The relocation of these surveillance requirements is consistent with the revised organization of TS Section 3/4.7 and is acceptable. The requirement of the new TS Section 4.7.D.2 to verify operability of each vacuum breaker assembly valve once per quarter by cycling the valve through one complete cycle, with concurrent verification of the operability of the valve position indicators, is a more precise way of stating the intent of existing TS 4.7.A.3.a. and is acceptable. The requirement of Section 4.7.D.3 for quarterly verification of the differential opening pressure for the vacuum breakers is an improvement to the requirement of the existing TS Section 3.7.A.3.a and is acceptable. The STS require monthly surveillance intervals for the vacuum breaker valves, while the proposed DAEC TS require quarterly surveillance intervals. Thus, the proposed DAEC requirements do not meet the STS requirements. However, they are the same as required by the existing DAEC TS which the staff has previously reviewed and found to be acceptable. The quarterly surveillance intervals are, therefore, acceptable.	2, 4
3.7-11	The title of TS section 3.7-E has been changed from "Main Steam Isolation Valve Leakage Control System (MSIV-LCS)" to "Drywell - Pressure Suppression Chamber Vacuum Breakers".	This is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable.	1

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3.7-11	TS section 3.7.E.1 contains the LCO for the Drywell - Pressure Suppression Chamber Vacuum Breakers previously located in TS section 3.7.A.4.a. This section has been revised to be consistent with the Suppression Chamber - Drywell Vacuum Breaker LCO contained in STS 3.6.4.1. The specified applicability, however, is the same as required in current TS section 3.7.A.4.a except that the reference to "except during testing" has been deleted. "Primary Containment Integrity" has been capitalized.	This LCO has been modified to correspond to the wording of the STS. The applicability of the LCO is the same as is required by the existing TS which, as discussed earli in the evaluation for Section 3.7.C.1, is nearly identical to the applicability requirements of the STS and is acceptable. Deletion of the words "except during testing" strengthens the LCO and is consistent with the STS which has no such exception. Capitalization of "primary containment integrity" has no affect on plant safety. In sum, this proposed change strengthens the LCO and is acceptable.	3, 4
3.7-11	TS sections 3.7.E.2 - 3.7.E.4 contain the action statements for specifications 3.7.E.1. These actions, previously located in TS sections 3.7.A.4.b - 3.7.A.4.d, have been revised to be consistent with the actions of STS section 3.6.4.1. Specifically: • The actions of TS section 3.7.E.2 are identical to action a. of STS section 3.6.4.1. • The actions of TS section 3.7.E.3 are identical to action b. of STS section 3.6.4.1. • The actions of TS section 3.7.E.4 are identical to action c. of STS section 3.6.4.1 with the following exception. The actions of TS section 3.7.E.4.b are DAEC-specific and were previously located in TS section 3.7.A.4.b. The specified time limits, however, are in accordance with the time limits of action C.2 of STS section 3.6.4.1.	The action statements of TS sections 3.7.E.2 through 3.7.E.4 apply the requirements of the STS to the DAEC LCO with one exception. The STS specifies that with one of the vacuum breaker position indicators inoperable, verification that the vacuum breaker is closed is determined by confirming the ability to maintain a 0.5 psi ΔP across the breaker for one hour without makeup. The DAEC-specific method for verification that the vacuum breaker is closed is to verify that the total drywell to suppression pool bypass area is less than 0.2 ft^2. This is in the existing TS and is applied to the revised TS in lieu of the STS method. This does not represent a reduction in safety from that afforded by the existing TS and is acceptable. Overall, the revised TS regarding the Drywell - Pressure Suppression Pool Vacuum Breakers represents a considerable improvement over the existing TS and is acceptable.	2, 3

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3.7-11	The title of TS section 4.7.E has been changed from "Main Steam Isolation Valve Leakage Control System" to "Drywell - Pressure Suppression Chamber Vacuum Breakers".	This is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable.	1
3.7-11	TS section 4.7.E.1 is a new surveillance requirement and is consistent with STS section 4.6.4.1.a.	This new surveillance requirement, conforming to the STS, represents an improvement from the existing TS and is acceptable.	3

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3.7-11	TS sections 4.7.E.2 - 4.7.E.4 contain the Drywell - Pressure Suppression Chamber Vacuum Breaker surveillance requirements previously located in TS sections 4.7.A.4.a - 4.7.A.4.d. These surveillance requirements have been revised as follows:	Relocation of the surveillance requirements is an administrative change necessitated by the revised organization of Section 3.7, and is acceptable.	1
	• TS section 4.7.E.2 has been reworded for clarity and now specifies that the position indication shall be verified as part of the cycling test. This surveillance requirement was previously located in TS section 4.7.A.4.a.	The rewording of this surveillance requirement is an editorial change to improve clarity, while the addition of the requirement to verify position indication is an improvement to the existing TS. This change is acceptable.	1, 2, 4
	• TS section 4.7.E.3 contains the inspection requirement previously located in TS section 4.7.A.4.c. The requirement to exercise all OPERABLE vacuum breakers upon identification of a vacuum breaker which is inoperable for opening (also located in previous TS section 4.7.A.4.c) has been deleted. The asterisk "*" has also been deleted.	Deletion of the requirement to immediately exercise all OPERABLE vacuum breakers and at 15-day intervals thereafter upon discovery of an inoperable vacuum breaker represents an improvement in safety in that it eliminates the need for unwarranted wear of the OPERABLE equipment and decreases the possibility of human error. The monthly cycle tests of the OPERABLE vacuum breakers is sufficient to assure their continuing operability. Deletion of the asterisk is an editorial change and is acceptable.	1, 4
	• The surveillance requirement for determining Drywell - Pressure Suppression Chamber bypass leakage previously located in TS section 4.7.A.4.b has been deleted. This surveillance requirement is already part of the actions specified in TS section 3.7.E.4.b.	This change in the location of the requirement to monitor bypass leakage does not affect the intent of the existing DAEC TS and is acceptable.	2
	• TS section 4.7.E.4 contains the test requirement previously located in TS section 4.7.A.4.d. The details of this test have been deleted.	Relocation of the test requirement does not affect the intent of the existing DAEC TS and is acceptable. Test details are not needed in the TS.	2

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3.7-11	• The note "*" at the bottom of the page has been deleted.	This footnote explained a previous amendment change. Its deletion is editorial in nature and does not affect the TS requirements. The change is acceptable.	1
3.7-12	The title of TS section 3.7.F has been changed from "Mechanical Vacuum Pump" to "Main Steam Isolation Valve Leakage Control System (MSIV-LCS)".	This is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable.	1
3.7-12	TS section 3.7.F.1 contains the LCO for the MSIV-LCS previously located in TS section 3.7.E.1. The reference to TS section 3.7.E.2. has been revised to 3.7.F.2.	Relocation of the LCO is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable.	1
3.7-12	TS section 3.7.F.2 contains the actions previously located in TS section 3.7.E.2. The statement allowing operation for 30 days after one MSIV-LCS is inoperable "provided all active components of the other MSIV-LCS subsystems are OPERABLE" has been changed to "verified to be OPERABLE".	Relocation of this action statement is an administrative change to accommodate the revised organization of Section 3.7, and is acceptable. The addition of the words "verified to be" allows the operators to rely upon the periodic (monthly) tests of the other MSIV components to verify operability rather than require potentially non-conservative, conditional surveillance testing of these components, which could result in equipment failure due to the testing and introduce the added possibility of human error. This verification of operability is consistent with the requirements of STS Section 4.4.7 for the MSIVs. Elimination of this additional testing does not substantially decrease the assurance of operability of the redundant MSIV components, and is acceptable.	1, 3, 4

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3.7-12	TS section 3.7.F.3 contains the actions previously located in TS section 3.7.E.3. The reference to TS section 3.7.E has been revised to 3.7.F. The requirement to be in COLD SHUTDOWN in 24 hours has been changed to be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN in the following 24 hours.	Relocation of this action statement is an administrative change to accommodate the revised organization of Section 3.7, and s acceptable. The change to the action statement represents a relaxation of the requirement in the existing DAEC TS, but the resulting requirement is consistent with the action statement of the corresponding STS Section 3.4.7.a.2. It allows for a more orderly shutdown of the plant, if needed, and is acceptable.	1, 3, 4
3.7-12	The title of TS sections 4.7.F has been changed from "Mechanical Vacuum Pump" to "Main Steam Isolation Valve Leakage Control System".	This is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable.	1
3.7-12	TS section 4.7.F. contains the testing requirements previously stated in section 4.7.E.1. The footnote in TS sections 4.7.F.1.a and 4.7.F.1.e has been deleted.	Relocation of the testing requirements is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable. The footnote was in explanation of a previous TS change and is no longer needed. Its deletion is acceptable.	1, 4
3.7-12	TS section 4.7.E.2 has been deleted. The operable MSIV-LCS subsystems are now only required to be "verified to be OPERABLE" per TS section 3.7.F.2.	This deletion is consistent with the revised wording of Section 3.7.F.2 and is acceptable.	4
3.7-12	The note "*" at the bottom of the page has been deleted.	The footnote was in explanation of a previous TS change and is no longer needed. Its deletion is acceptable.	4

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3.7-13	TS section 3.7.G, "Suppression Pool Level and Temperature" has been added. This new TS section contains the suppression pool level and temperature requirements previously located in TS section 3.7.A.1. The LCO now specifies that the suppression pool shall be OPERABLE. The applicability of the suppression pool level and temperature limits has been revised to delete references to "work is being done which has the potential to drain the vessel".	This is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable. Addition of the requirement for suppression pool operability to the LCO is an enhancement to the existing DAEC TS and is acceptable. Deletion of the requirement to maintain the limits on suppression pool water volume when "work is being done which has the potential to drain the vessel" has no impact on plant safety during periods of reactor operation since such work is accomplished only during shutdown periods. The revised TS is consistent with the STS which requires suppression pool operability only during Operational Conditions 1, 2 and 3. Existing DAEC TS 3.5.G.4.d requires that, during a refueling outage, operations that have the potential for draining the reactor vessel will be suspended whenever the water level in the suppression chamber falls below the minimum. This is consistent with STS section 3.5.3.b.1. Thus, we find deletion of the words "work is being done which has the potential to drain the vessel" acceptable.	1, 3, 4

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3.7.13	TS section 3.7.G.1, "Suppression Pool Level", contains the LCO for Suppression Pool Level previously located in TS sections 3.7.A.1.a and 3.7.A.1.b. This section has been revised for clarity and to be consistent with STS section 3.6.2.1. Specifically:	Relocation of the suppression pool level requirements is an administrative change to accommodate the revised organization of Section 3.7. The revisions to the wording improve the clarity and conform the wording to the STS format. The changes are acceptable.	1, 3
	• TS section 3.7.G.1.a is consistent with STS section 3.6.2.1.a.1. A reference to indicated suppression pool water level (in percent) has been added.	The wording format now follows the STS wording format. There are no changes to the maximum and minimum specified water levels. Addition of the percentages of suppression pool volume to the water level limits adds information not previously provided. These changes are acceptable.	2, 3
	• TS section 3.7.G.1.b is consistent with action a. of STS section 3.6.2.1. A reference to indicated suppression pool water level has been added.	This action statement is not present in the existing DAEC TS. It is an improvement, consistent with the STS, and is acceptable.	3

TS Page No.	Proposed Change	Evaluation of the Proposed Change	Review Category
3.7-13	TS section 3.7.G.2, "Suppression Pool Temperature" contains the LCOs for suppression pool temperatures previously located in TS section 3.7.A.1.c. This section has been revised for clarity. The specified limits and actions are consistent with STS section 3.6.2.1. Specifically:	Relocation of the suppression pool temperature requirements is an administrative change to accommodate the revised organization of Section 3.7. The revisions to the wording improve the clarity and conform the wording to the STS format. The changes are acceptable.	1, 3
	• TS section 3.7.G.2.a specifies the normal suppression pool temperature limit previously located in TS section 3.7.A.1.c. (1). This temperature limit is consistent with the normal temperature limit specified in STS section 3.6.2.1.a.2.	This is the same temperature limit specified in the existing DAEC TS. It is acceptable.	2
	• TS section 3.7.G.2.b is a new LCO which specifies what actions are to be taken when average suppression pool water temperature is > 95 °F but < 110 °F during operation and not performing testing which adds heat to the pool. These actions are consistent with action b. of STS section 3.6.2.1 and STS section 4.6.2.1.b.2.a.	This LCO is not present in the existing DAEC TS, which are silent regarding actions to be taken when the pool temperature is >95°F but <110°F and testing is not in progress which adds heat to the pool. The 110°F upper limit is allowed by the existing DAEC TS. This change is an improvement to the TS and is acceptable.	2, 4
	• TS section 3.7.G.2.c contains the suppression pool water temperature limits during the performance of testing which adds heat to the suppression pool previously located in TS section 3.7.A.1.c. (2). This limit and the specified actions are consistent with STS section 3.6.2.1.a.2. (a) and action b.1. of STS section 3.6.2.1. The requirement to verify temperature is < 110 °F once/hr has been added as an additional conservatism and is consistent with STS section 4.6.2.1.b.2.a.	This LCO amplifies the requirements stated in the existing DAEC TS by specifying the actions to be taken if the temperature exceeds 105°F during testing which adds heat to the pool. The actions are consistent with the STS and are acceptable.	2, 4

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3.7-13	TS section 4.7.G, "Suppression Pool Level and Temperature", has been added. This new TS section contains the suppression pool surveillance requirements previously located in TS section 4.7.A.1.	Relocation of the suppression pool surveillance requirements is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable.	1
3.7-13	TS sections 4.7.G.1, "Suppression Pool Level", and 4.7.G.1.a contain the suppression pool level surveillance requirements previously located in TS section 4.7.A.1.a. This surveillance is consistent with STS section 4.6.2.1.a.	This requirement is the same as in the existing DAEC TS, but is reformatted to conform to the STS wording. It is acceptable.	2, 3
3.7-13	TS sections 4.7.G.2, "Suppression Pool Temperature" and 4.7.G.2.a contain the suppression pool temperature surveillance requirements previously located in TS section 4.7.A.1.a. This surveillance is consistent with STS section 4.6.2.1.b.	This requirement is the same as in the existing DAEC TS. It has been reformatted to conform to the pattern of the STS wording. It is acceptable.	2, 3
3.7-13	TS section 4.7.G.2.b contains the suppression pool temperature surveillance requirement previously located in TS section 4.7.A.1.b. The requirement to verify suppression pool water temperature every 5 minutes when there is indication of relief valve operation has been deleted. This surveillance is consistent with STS section 4.6.2.1.b.1. The details of this verification (monitoring) have been deleted.	This requirement is the same as in the existing DAEC TS but the 105°F limit has been added. It has been reformatted to conform to the pattern of the STS wording. The requirement for surveillance when there is an indication of relief valve operation has been deleted and is now incorporated into new TS 4.7.G.2.c. The change is an improvement and is acceptable	2, 3, 4

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3.7-14	TS section 3.7.G.2.d contains the suppression pool water temperature limits and actions previously located in TS section 3.7.A.1.c. (3). These limits and actions are consistent with action b.2 of STS section 3.6.2.1. The requirement for resuming power operation previously included in TS section 3.7.A.1.c. (3) is adequately covered by revised TS section 3.7.G.2.a and has been deleted.	The requirement to scram the reactor if the suppression pool average water temperature exceeds 110°F is the same as in the existing DAEC TS and is acceptable. The condition for resumption of power operation (pool temperature equal to or less than 95°F) is adequately stated in TS Section 3.7.G.2.a. Therefore, deletion of this portion of the requirements from the existing TS is acceptable.	2, 4
3.7-14	TS section 3.7.G.2.e contains the suppression pool water temperature limit and actions previously located in TS section 3.7.A.1.c.(4). This limit and specified action have been reworded to be consistent with action b.3 of STS section 3.6.2.1.	Relocation of the suppression pool water temperature limit requirement is an administrative change to accommodate the revised organization of Section 3.7. The revisions to the wording improve the clarity and conform the wording to the STS format. These changes are acceptable. The deletion of the phrase, "during reactor isolation conditions," make the requirement more restrictive and is therefore acceptable.	1, 3, 4
3.7-15	TS section 3.7.H, "Containment Atmosphere Dilution," has been added. This new TS section contains the containment atmosphere dilution requirements previously located in TS section 3.7.A.6.	Relocation of the containment atmosphere dilution requirements is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable.	1

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3.7-15	TS section 3.7.H.1 contains the requirements previously located in TS section 3.7.A.6.a. The applicability has been revised to specify that the containment atmosphere dilution system is only required to be operable when the primary containment is required to be inerted. The term "operable" has been capitalized. Additionally, the requirement to take the reactor "out of power operation" has been revised to "be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN in the following 24 hours" to be consistent with the other shutdown requirements of this chapter.	The revision of the applicability statement to specify that the CAD system need be OPERABLE only when the containment is required to be inerted is a relaxation of the requirement in the existing DAEC TS, but the resulting applicability is consistent with STS 3.6.6.2 for when the containment must be inerted. There is no need for the CAD system to be operable when the containment is not required to be inerted. Therefore, this relaxation of the requirement in the existing TS is acceptable. Capitalization of "operable" is an editorial change and is acceptable. Replacement of the vague requirement to "take the reactor out of power operation" with the specified times to be in HOT SHUTDOWN and COLD SHUTDOWN will provide the operators with firm guidance as to what is required. The time requirements are consistent with terminology used in the STS and with other action statements related to the primary containment and are acceptable.	1, 3, 4
3.7-15	TS section 3.7.H.2 contains the requirements previously located in TS section 3.7.A.6.b. The requirement to take the reactor "out of power operation" has been revised to "be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN in the following 24 hours" to be consistent with the other shutdown requirements of this chapter.	The basic requirement for the minimum volume of N_2 to be available, and the need to restore this volume within 7 days if the specification cannot be met, are unchanged from the existing TS and are acceptable. Replacement of the vague requirement to "take the reactor out of power operation" with the specified times to be in HOT SHUTDOWN and COLD SHUTDOWN will provide the operators with firm guidance as to what is required. The time requirements are consistent with terminology used in the STS and with other action statements related to the primary containment and are acceptable.	2, 4

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3.7-15	TS section 3.7.H.3 contains the requirements previously located in TS section 3.7.A.6.c.	This proposed change is identical to the wording of the existing DAEC TS and is acceptable.	2
3.7-15	TS section 4.7.H., "Containment Atmosphere Dilution", has been added. This new TS section contains the containment atmosphere dilution surveillance requirements previously located in TS section 4.7.A.6.	This is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable.	1
3.7-15	TS section 4.7.H.1 contains the surveillance requirements previously located in TS section 4.7.A.6.a.	This surveillance requirement is unchanged from the requirement in the existing DAEC TS and is acceptable.	2
3.7-15	TS section 4.7.H.2 contains the surveillance requirements previously located in TS section 4.7.A.6.b.	This surveillance requirement is unchanged from the requirement in the existing DAEC TS and is acceptable.	2
3.7-15	TS section 4.7.H.3 contains the surveillance requirements previously located in TS section 4.7.A.6.c. The reference to TS section 4.7.A.6.a has been revised to 4.7.H.1.	This surveillance requirement is unchanged from the requirement in the existing DAEC TS and is acceptable. Revision of the reference is an administrative change to accommodate the revised organization of Section 3.7 and is acceptable.	1, 2

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3.7-16	TS section 4.7.I, "Oxygen Concentration," has been added. This new TS section contains the oxygen concentration surveillance requirements previously located in TS section 4.7.A.5. The frequency of the surveillance however, has been revised to be consistent with STS section 4.6.6.2.	The movement of the requirement to the new location is an administrative change and is acceptable. Addition of the requirement to verify the oxygen concentration within 24 hours after placing the mode switch in RUN is a new requirement, comparable to the STS requirement, and consistent with the proposed new LCO 3.7.I.2. It is acceptable. The change in surveillance frequency from twice weekly to once every 7 days is consistent with the STS surveillance requirement, and is acceptable.	1, 3, 4
3.7-17	TS section 3.7.J, "Secondary Containment" has been added. This new TS section contains requirements previously located in TS section 3.7.C.	This is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable.	1
3.7-17	TS section 3.7.J.2 contains the requirements previously located in TS section 3.7.C.2. The reference to TS section 3.7.C.1 has been revised to 3.7.J.1.	This is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable.	1
3.7-17	TS section 3.7.J.2.c has been revised to be consistent with the other shutdown requirements of this chapter.	The change in the shutdown requirement provides for a more orderly shutdown, is consistent with the shutdown requirements elsewhere in this section, and is acceptable.	4
3.7-17	TS section 4.7.J, "Secondary Containment" has been added. This new TS section contains surveillance requirements previously located in TS section 4.7.C.	This is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable.	1

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3.7-17	TS section 4.7.J.1 has been revised to delete all mention of historical testing previously described in TS sections 4.7.C.1.a and 4.7.C.1.b.	This is an administrative change, deleting material that no longer is applicable to the operation of the DAEC. It does not affect current plant operations and is acceptable.	1, 4
3.7-17	TS section 4.7.J.1.a contains the surveillance requirements previously located in TS section 4.7.C.1.c. The reference to calm wind conditions as < 5 mph has been revised to < 15 mph. A new discussion of what constitutes calm wind conditions has been added to the Bases to TS section 3.7.L.	The relocation of the surveillance requirement is an administrative change and is acceptable. The change in the definition of calm wind speed from 5 mph to 15 mph is based upon the licensee's engineering evaluation of the effect of wind speeds on the secondary containment manometer readings. The change is conservative in that it requires that secondary containment vacuum be maintained over a wider range of wind speeds. It is, therefore, acceptable.	1, 4
3.7-17	Previous TS section 4.7.C.1.d has been deleted. This type of verification testing is not required by STS.	Deletion of this requirement does not materially affect plant safety and it eliminates unnecessary operation of the SGTS. Most violations of secondary containment are temporary and minor in nature (open doors, penetrations, etc.) and are readily correctable after identification. They do not demand a test to verify the capability of maintaining a vacuum after the violation has been corrected. Any modifications to the secondary containment boundary which could result in a change in secondary containment operability are subjected to post-modification testing which would confirm the operability of the secondary containment. The change is, therefore, acceptable.	4

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3.7-17		(Continued) While there is no requirement in the STS to verify secondary containment integrity after identification of a violation, STS Section 4.6.5.1 does require periodic verification of reactor building negative pressure and closure of secondary containment penetrations, doors, hatches, and blowout panels. DAEC FSAR Section 6.5 states that the secondary containment is maintained at a 1/4-inch of water vacuum during normal operation, but neither the existing or the proposed DAEC TS require periodic verification of the secondary containment pressure. While this requested TS change does not justify adding such a requirement to the TS, this is a shortcoming that the licensee should be urged to correct. Neither the existing nor the proposed TS state a time limit for the SGTS to reduce the secondary containment pressure to a negative 1/4 inch of water. Since the negative pressure in secondary containment is maintained during normal operation, this should not be a problem. However, upon loss of offsite power and transfer to emergency power, this could become a factor. The STS allow a 375 second time period to establish the negative pressure based upon the results of accident analyses. While this requested TS change does not provide justification for adding a time limit for operation of the SGTS to reduce the secondary containment pressure to the negative 1/4-inch of water, this is a shortcoming that the licensee should be urged to correct.	

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3.7-18	TS section 3.7.K., "Secondary Containment Automatic Isolation Dampers", has been added. These requirements were not previously included in the DAEC TS. The requirements of TS section 3.7.K.1 are consistent with the requirements of STS section 3.6.5.2. Specifically:	This is a new requirement, not present in the existing DAEC TS. As such, it is an improvement to the existing TS, consistent with the STS, and is acceptable.	3
	• TS section 3.7.K.1 is consistent with STS 3.6.5.2. The applicability, however, is DAEC-specific and the list of applicable valves/dampers is not included in the TS but will be incorporated into an administrative procedure.	This requirement is consistent with the intent of the STS, but is tailored to be specific to DAEC. It is an improvement to the existing TS and is acceptable.	3, 4
	• TS section 3.7.K.2 (including sub items a., b., and c) are consistent with actions a., b., and c., of STS sections 3.6.5.2. A note, however, has been added to TS section 3.7.K.2.c. This note is consistent with the note for closed/isolated primary containment isolation valves.	This new requirement, not present in the existing DAEC TS, conforms the revised TS to the STS requirements with the exception of the note allowing intermittent opening of the isolated penetrations under administrative control. This exception is the same as is allowed for the primary containment isolation valves in existing DAEC TS 3.7.D.2 and in proposed revised DAEC TS 3.7.B.2. There is no reason for this new requirement regarding secondary containment isolation dampers to be more restrictive than the requirement for the primary containment isolation valves. It is, therefore, acceptable.	3,4
3.7-18	TS section 3.7.K.3 is consistent with the shutdown action statement of STS section 3.6.5.2. The requirement to suspend reactor building fuel cask and irradiated fuel movement is consistent with TS section 3.7.J.2.a.	This action statement follows the format of, but is not as all-inclusive as, the STS in that it does not address suspension of core alterations and operations with the potential for draining the reactor vessel. However, it is a considerable improvement to the existing DAEC TS, and it is acceptable.	3, 4

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3.7-18	TS section 4.7.K "Secondary Containment Automatic Isolation Dampers", has been added. This surveillance requirement was not previously required by DAEC TS. The surveillance requirement of TS section 4.7.K is consistent with the surveillance requirements of STS section 4.6.5.2 except that surveillance requirements a. and c. of STS section 4.6.5.2 are not included. (Post-maintenance testing has not historically been included in DAEC TS.) Specifically:	This is a new surveillance requirement, not present in the existing DAEC TS. It is consistent with the STS requirement for testing at least once per operating cycle, but omits the STS requirement for verifying isolation damper operating times and the requirement for post-maintenance testing of the dampers. Still, it is an improvement over the existing TS and is acceptable.	3, 4
	• TS section 4.7.K.1 is consistent with STS section 4.6.5.2.b.	This statement is partially true, but the proposed surveillance requirement calls only for simulated initiation of the dampers, while the STS require verification that the dampers actuate to their isolated position on the test signal. It is, however, an improvement over the existing TS and is acceptable.	4
3.7-19	TS section 3.7.L, "Standby Gas Treatment System," has been added. This new TS section contains the standby gas treatment system requirements previously located in TS section 3.7.B. The reference to TS section 3.7.B.3 has been revised to 3.7.L.3.	This is an administrative change to accommodate the revised organization of Section 3.7. It is, acceptable.	1
3.7-19	TS section 4.7.L Standby Gas Treatment System", has been added. This new TS section contains the standby gas treatment system surveillance requirements previously located in TS section 4.7.B.	This is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable.	1

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3.7-19	TS section 4.7.L.1.b has been revised to require that the inlet heaters of each train be capable of an output of at least 22 kw. The previous TS requirement was 11 kw.	The 22 kw requirement in the revised TS is more restrictive than the 11 kw requirement in the existing TS. The licensee states that the 22 kw minimum is needed to assure that the assumed initial conditions for inlet to the SGTS can be met, but that the temperature rise across the heaters at this heat rate still would be less than the 20°F maximum differential temperature specified in the UFSAR Section 6.5.3.3. The change, therefore, is acceptable.	4
3.7-19	TS section 4.7.L.1.c has been revised to require that the air distribution demonstration be performed "after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing." This test was previously required to be performed annually.	The requirement to demonstrate air distribution after "each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing" is an improvement to the present requirement for annual demonstration. The requirement is consistent with ASME N510-1989, "Testing of Nuclear Air Treatment Systems." Any changes to the flow distribution would be expected to occur after changes are made to the filters or filter housing rather than on a time-dependent basis. The change, therefore, is acceptable.	4
3.7-19	In TS section 4.7.L.1.g, the reference to TS section 3.7.B.2.b. has been revised to 3.7.L.2.b.	This is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable.	1
3.7-19	In TS section 4.7.L.2.a, the reference to TS section 3.7.B.2 has been revised to 3.7.L.2.	This is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable.	1

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3.7-20	TS section 3.7.L.3 contains the requirements previously located in TS section 3.7.B.3. The wording has been revised for consistency. Specifically:	This is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable.	1
	• The wording "HOT SHUTDOWN within 12 hours" has been revised to 'HOT SHUTDOWN within the next 12 hours".	This is an editorial change which clarifies the intent of the action statement. It is acceptable.	1
	• The wording "suspend fuel handling operations" has been revised to "suspend reactor building fuel cask and irradiated fuel movement." This wording is consistent with TS sections 3.7.J and 3.7.K.	This change makes the action statement for an inoperable SGTS comparable to the action statements for lack of secondary containment integrity or for inoperability of the secondary containment isolation dampers. The revised words are more explicit, but may not be as all-inclusive as the existing words, e.g., they could allow movement of fresh fuel to continue. However, since the principal danger of radioactive release stems from movement of irradiated fuel or movement of the fuel cask, the net effect of the change with regard to safety is minimal. Since the safety impact is minimal and since the change conforms the action statement to the comparable statements for secondary containment integrity and for inoperability of the secondary containment isolation dampers, the change is acceptable.	4
3.7-21	TS section 3.7.M, "Mechanical Vacuum Pump" has been added. This new TS section contains the mechanical vacuum pump requirements previously located in TS section 3.7.F.	This is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable.	1
3.7-21	In TS section 3.7.M.3, the references to TS sections 3.7.F.1 and 3.7.F.2 have been revised to 3.7.M.1 and 3.7.M.2.	This is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable.	1

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3.7-21	TS section 4.7.M, "Mechanical Vacuum Pump" has been added. This new TS section contains surveillance requirements previously located in TS section 4.7.F.	This is an administrative change to accommodate the revised organization of Section 3.7. It is acceptable.	1
3.7-22	The Bases for TS sections 3.7.A and 4.7.A has been changed from "Primary Containment" to "Primary Containment Integrity." A discussion of the requirement to restore primary containment within 1 hour in the event primary containment is inoperable has been added. The Bases information has been reorganized for clarity.	This is an editorial change necessary to accommodate the revised organization of Section 3.7. The new Bases for TS Sections 3.7.A and 4.7.A incorporate those portions of the existing TS Bases that relate to primary containment integrity and leak rate testing. The added discussion regarding the need to restore primary containment integrity within 1 hour is an improvement. The changes are acceptable.	1, 2, 4
3.7-24	The Bases for TS sections 3.7.B and 4.7.B, "Primary Containment Power Operated Isolation Valves", contains the information previously located in Bases section 3/4.7.A.8. This information has been reorganized for clarity.	This is an administrative change to account for the revised organization of Section 3.7, and is acceptable.	1
3.7-26	A discussion of the actions to be taken in the event that one or more primary containment isolation valves are inoperable has been added to the Bases of section 3.7.B. and 4.7.B.	This discussion regarding the need for isolation and the method of isolation of inoperable PCIVs is generally an improvement. However, the words relating to use of "a check valve inside primary containment with flow through the valve secured" as an acceptable isolation barrier is not consistent with the TS 3.7.B and should be corrected.	4
3.7-26	The Bases for TS sections 3.7.C and 4.7.C, "Drywell Average Air Temperature" has been added to provide additional information on this new specification.	This is a new Bases section supporting the new TS Section 3/4 7.C. It provides the rationale for selecting the related LCO limits and surveillance frequency, and explains the need for Drywell temperature control.	4

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3.7-27	The Bases for TS sections 3.7.D and 4.7.D, "Pressure Suppression Chamber - Reactor Building Vacuum Breakers" contains the information previously located in Bases section 3/4.7.A.3. This section has been expanded to provide additional information specific to these vacuum breakers.	This is an administrative change to account for the revised organization of Section 3.7, and is acceptable. The added discussion regarding the need to restore operability of inoperable vacuum breakers explains the need for the action statements of TS Section 3.7.D and is a considerable improvement over the existing wording in the present Bases. It is acceptable.	1, 4
3.7-28	The Bases for TS sections 3.7.E and 4.7.E, "Drywell-Pressure Suppression Chamber Vacuum Breakers" contains the information previously located in Bases section 3/4.7.A.3. This section has been expanded to provide additional information specific to these vacuum breakers.	This is an administrative change to account for the revised organization of Section 3.7, and is acceptable. The added discussion regarding the need to restore operability of inoperable vacuum breakers explains the need for the action statements of TS Section 3.7.E and is a considerable improvement over the existing wording in the present Bases. It is acceptable.	1, 4
3.7-29	The Bases for TS section 3.7.F and 4.7.F, "Main Steam Isolation Valve Leakage Control System (MSIV-LCS)" contains the information previously located in Bases section 3/4.7.E.	This is an administrative change to account for the revised organization of Section 3.7, and is acceptable.	1
3.7-30	The Bases for TS sections 3.7.G and 4.7.G, "Suppression Pool Level and Temperature" contains the information previously located in Bases sections 3/4.7.A.1, 3/4.7.A.5, and 3/4.7.A.8. This information has been reorganized for clarity. Additional details regarding the bases for the maximum suppression pool volume and equivalent indicated levels has been added.	This is an administrative change to account for the revised organization of Section 3.7, and is acceptable. The added discussion regarding the need for a maximum water volume in the suppression pool is an improvement and is acceptable.	1, 4

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3.7-32	The requirement to maintain the suppression pool temperature "below" the normal operating limit of 95 °F has been changed to "at or below" the normal operating limit of 95 °F. This change is consistent with TS section 3.7.G.2.a.	This is an editorial change to conform the Bases words to the requirements of the TS. It does not change the intent of the wording in the present BASES and is acceptable.	1
3.7-33	The reference to Bases section 3.7.A.1 (previously located in the first paragraph of previous TS page 3.7-48a) has been deleted.	This is an editorial change. The deleted reference is no longer needed since the words from previous Bases Section 3.7.A.1 are now incorporated in the new Bases section. The change is acceptable.	1
3.7-33	The discussion of the daily suppression pool level and temperature surveillance previously located in the first paragraph of the Bases section 3/4.7.A.4, "Leak Rate Testing" (but applicable to suppression pool level and temperature) has been deleted. This information is redundant to the surveillance discussion already included in the Bases for section 3.7.G and 4.7.G.	The deleted discussion regarding daily volume and temperature checks of the suppression pool water are adequately covered in the revised Bases for Sections 3.7.G and 4.7.G. Inclusion of this discussion would be redundant. Therefore, this change is acceptable.	4
3.7-34	The Bases for TS section 3.7.H and 4.7.H, "Containment Atmosphere Dilution" contains the information previously located in the Bases for sections 3/4.7.A.6.	This is an administrative change to account for the revised organization of Section 3.7, and is acceptable.	1
3.7-35	The Bases for TS sections 3.7.I and 4.7.I, "Oxygen Concentration", contains the information previously located in Bases section 3/4.7.A.2.	This is an administrative change to account for the revised organization of Section 3.7, and is acceptable.	1

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3.7-36	Additional information has been added to the Bases for TS section 3.7.I and 4.7.I stating that the CAD system is not required to be operable during drywell inspections and when the containment is not inerted. This note is consistent with the note added to TS section 3.7.I.	This additional information is consistent with the requirement of revised TS Section 3.7.H (which refers to TS Section 3.7 I.1) and is acceptable. Contrary to the licensee's statement, there is no note added to TS Section 3.7.I. Rather, the information apparently was incorporated in the text of the TS. In any event, the additional information provided in the Bases supports the requirements of the TS and is acceptable.	4
3.7-36	The discussion of oxygen monitoring in the last paragraph of the Bases for TS sections 3.7.I and 4.7.I has been changed from "twice a week" to "once per week". This corresponds to revised surveillance requirement 4.7.I.1.	This revision conforms the Bases' wording to the TS requirement, which was previously found acceptable. Therefore, this change is acceptable.	4
3.7-36	The Bases for TS sections 3.7.J and 4.7.J, "Secondary Containment" contains the information previously located in Bases section 3/4.7.A.7.	This is an administrative change to account for the revised organization of Section 3.7, and is acceptable.	1
3.7-37	The Bases for TS sections 3.7.K and 4.7.K, "Secondary Containment Automatic Isolation Dampers", has been added to provide additional information on this new specification.	This additional information explains the need for controls on the operability of the secondary containment automatic isolation dampers, supporting the new TS on this equipment. It is acceptable.	4
3.7-38	The Bases for TS sections 3.7.L and 4.7.L, "Standby Gas Treatment System" contains the information previously contained in Bases section 3/4.7.A.7. The previous reference to TS section 3.7.B.3 has been changed to TS section 3.7.L.3.	This is an administrative change to account for the revised organization of Section 3.7, and is acceptable.	1

TS Page No.	Proposed Change	Evaluation of the Proposed Change	Review Category
3.7-39	The Bases for TS section 3.7.L and 4.7.L, "Standby Gas Treatment System", has been revised to clarify the in-place and laboratory tests performed on the system. Specifically, the previous reference to "less than 1 percent bypass leakage" for the charcoal absorbers has been changed to "$\leq$ 0.1 percent bypass leakage" for the charcoal absorbers. This change is consistent with the requirements of TS sections 3.7.L.2.a. The discussion of laboratory carbon sample test results previously described a radioactive methyl iodide removal efficiency of "at least 99.9 percent for expected accident conditions." This has been changed to "at least 99% for expected accident conditions." This change is consistent with the requirements of TS section 3.7.L.2.b.	These revisions conform the discussion in the Bases to the requirements of the proposed revised TS and are acceptable. The corresponding numbers presented in the present Bases Section 3/4.7.A.7 are in error.	4
3.7.39	None	The Bases for Section 3.7.L and 4.7.L, carried forward from the Bases of the existing DAEC TS, state that "Heater capability, pressure drop, and <u>air distribution</u> should be determined <u>annually</u> to show system performance capability." However, proposed revised TS 4.7.L.1.c now requires an air distibution demonstration to be performed "after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing," in lieu of the requirement in existing DAEC TS 4.7.B.1.c for an annual demonstration of the air flow distribution. The proposed revised Bases thus needs to be modified to state that air flow distribution is determined following complete or partial replacement of the HEPA filter bank and after any structural maintenance on the system housing instead of the annual determination.	4

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TS Page No.	Proposed Change	Evaluation of the Proposed Change	Review Category
3.7-40	The description of standby gas treatment system (SGTS) inlet heater capacity in Bases section 3.7.L and 4.7.L has been revised from 11 kw to 22 kw. This change is consistent with revised TS section 4.7.L.1.b.	The change conforms the wording in the Bases to the requirement of the revised TS 4.7.L.1.b and is acceptable.	4
3.7-40	A specific discussion of the engineering evaluation regarding the effects of differing wind speeds on SGTS testing has been added to the Bases of TS sections 3.7.L and 4.7.L.	New TS 4.7.J.1.a requires the capability of maintaining a 1/4-inch water vacuum in the secondary containment with wind speeds up to 15 mph. This additional discussion explains how the averaged manometer readings compensate for wind speeds up to 15 mph. The change adds clarification and is acceptable.	4
3.7-41	The Bases for TS sections 3.7.M and 4.7.M "Mechanical Vacuum Pump", contains the information previously located in Bases sections 3.7.F and 4.7.F.	This is an administrative change to account for the revised organization of Section 3.7, and is acceptable.	1
3.7-42	Previous TS page number 3.7-49 has been renumbered to page 3.7-42.	This is an administrative change to account for the revised organization of Section 3.7, and is acceptable.	1
3.7-43	Previous TS page number 3.7-50 has been renumbered to page 3.7-43.	This is an administrative change to account for the revised organization of Section 3.7, and is acceptable.	1
3.7-43	Previous TS page 3.7-20 has been deleted.	The deleted page, inserted by Amendment 181, noted that a series of tables had been deleted from the TS, which resulted in a number of pages no longer being used. With this revision to the TS, this note is no longer needed. This change, therefore, is acceptable.	4