Docket No.: 50-293 January 21, 1988

Boston Edison Company
ATTN: Mr. Ralph E. Bird
Senior Vice President - Nuclear
800 Boylston Street
Boston, Massachusetts 02199

SUBJECT: ISSUANCE OF AMENDMENT NO. 113 TO FACILITY OPERATING LICENSE NO. DPR-35 (TAC NO. 59190) PILGRIM NUCLEAR POWER STATION

Dear Mr. Bird:

The Commission has issued the enclosed Amendment No. 113 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment consists of changes to the Technical Specifications in response to your application dated June 4, 1987, as supplemented by letters dated August 13, September 21, and December 8, 1987.

This amendment concerns technical specifications regarding primary containment isolation valves. The changes include (1) revising the definition of frequently used terms, (2) modifying Table 3.2-B to add instrumentation for the two new High Pressure Coolant Injection System vacuum breakers, (3) revising containment isolation valve listing in Table 3.7.1, and (4) making editorial changes and adding clarifications to facilitate the interpretation of the Technical Specifications.

A copy of our Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

/S/

Richard H. Wessman, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 113 to DPR-35
2. Safety Evaluation

cc w/enclosures:
See next page

Distribution:
See next page

Concurrence subject to modifications noted on SE and question raised in my note dated 12/22/87

OFC	PDI-3	PDI-3	OGC-Bethesda	ACTDIR	PDI-3		
NAME	RWessman:lm	MRushbrook	SHLewis	VRooney	RWessman		
DATE	12/14/87	12/15/87	12/22/87	12/21/87	1/21/88		

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

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See next page

Mr. Ralph G. Bird
Boston Edison Company

Pilgrim Nuclear Power Station

cc:

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

BOSTON EDISON COMPANY

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 113
License No. DPR-35

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

   A. The application for amendment by Boston Edison Company (the licensee) dated June 4, 1987, as supplemented by letters dated August 13, September 21, and December 8, 1987, 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 3.B of Facility Operating License No. DPR-35 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 113, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective 30 days after the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard H. Wessman, Acting Director
Project Directorate I-3
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 21, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 113

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are provided to maintain document completeness.

Remove Pages	Insert Pages
ii	ii
3	3
5b	5b
48	48
68	68
69	69
152	152
152a	---
152b	---
153-155	153-155
---	155a
---	155b
157a	157a
160-171a	160-171a
175	175
225	225

valve closure, are bypassed when reactor pressure is less than 600 psig, the low pressure main steam line isolation valve closure trip is bypassed, the reactor protection system is energized with IRM neutron monitoring system trips and control rod withdrawal interlocks in service.

2. Run Mode - In this mode the reactor system pressure is at or above 880 psig and the reactor protection system is energized with APRM protection and RBM interlocks in service.

3. Shutdown Mode - The reactor is in the shutdown mode when the reactor mode switch is in the shutdown mode position and no core alterations are being performed.

   a. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212°F.

   b. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F.

4. Refuel Mode - The reactor is in the refuel mode when the mode switch is in the refuel mode position. When the mode switch is in the refuel position, the refueling interlocks are in service.

L. Design Power - Design power means a steady-state power level of 1998 thermal megawatts.

M. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

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

3. All blind flanges and manways are closed.

4. All automatic primary containment isolation valves are operable or at least one containment isolation valve in each line having an inoperable valve shall be deactivated in the isolated condition.

5. All containment isolation check valves are operable or at least one containment valve in each line having an inoperable valve is secured in the isolated position.

N. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1.0 DEFINITIONS (Continued)

AA. Action - Action shall be that part of a specification which prescribes remedial measures required under designated conditions.

BB. Member(s) of the Public[1] - Member(s) of the public shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the site.

CC. Site Boundary[1] - The site boundary is shown in Figure 1.6-1 in the FSAR.

DD. Radwaste Treatment System

1. Gaseous Radwaste Treatment System - The gaseous radwaste treatment system is that system identified in Figure 4.8-2.

2. Liquid Radwaste Treatment System - The liquid radwaste treatment system is that system identified in Figure 4.8-1.

EE. Automatic Primary Containment Isolation Valves - Are primary containment isolation valves which receive an automatic primary containment group isolation signal.

[1] See FSAR Figure 1.6-1

TABLE 3.2-B (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum # of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Remarks
2	High Drywell Pressure	≤2.5 psig	1. Initiates Core Spray; LPCI; HPCI. 2. In conjunction with Low-Low Reactor Water Level, 120 second time delay and LPCI or Core Spray pump running, initiates Auto Blowdown (ADS) 3. Initiates starting of Diesel Generators. 4. In conjunction with Reactor Low Pressure initiates closure of HPCI vacuum breaker containment isolation valves.
1	Reactor Low Pressure	400 psig ± 25	Permissive for Opening Core Spray and LPCI Admission valves.
1	Reactor Low Pressure	≤110 psig	In conjunction with PCIS signal permits closure of RHR (LPCI) injection valves.
1	Reactor Low Pressure	400 psig ± 25	In conjunction with Low-Low Reactor Water Level initiates Core Spray and LPCI.
2	Reactor Low Pressure	900 psig ± 25	Prevents actuation of LPCI break detection circuit.
2	Reactor Low Pressure	100>P>50 psig	In conjunction with High Drywell Pressure initiates closure of HPCI vacuum breaker containment isolation valves.

BASES:

3.2 In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The set points of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at 128.26 inches above the top of the active fuel closes all isolation valves except those in Groups 1, 4 and 5. This trip setting is adequate to prevent core uncovery in the case of a break in the largest line assuming a 60 second valve closing time. Required closing times are less than this.

The low low reactor water level instrumentation is set to trip when reactor water level is 77.26 inches above the top of the active fuel (-49" on the instrument). This trip closes Main Steam Line Isolation

Valves, Main Steam Drain Valves, Recirc Sample Valves (Group 1) activates the CSCS subsystems, starts the emergency diesel generators and trips the recirculation pumps. This trip setting level was chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation and primary system isolation so that no fuel damage will occur and so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation and primary system isolation are initiated in time to meet the above criteria.

The high drywell pressure instrumentation is a diverse signal to the water level instrumentation and in addition to initiating CSCS, it causes isolation of Group 2 isolation valves. For the breaks discussed above, this instrumentation will initiate CSCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here also. The low low water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes isolation of Group 1 isolation valves.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140% of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperatures remain approximately 1000°F and release of radioactivity to the environs is well below 10 CFR 100 guidelines.

Temperature monitoring instrumentation is provided in the main steam line tunnel and the turbine basement to detect leaks in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. The setting of 170°F for the main steam line tunnel detector is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop acci-

3.7 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

Suppression Pool

1. At any time that the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water volume and temperature shall be maintained within the following limits except as specified in 3.7.A.2 and 3.7.A.3.

   a. Minimum water volume - 84,000 ft^3

   b. Maximum water volume - 94,000 ft^3

   c. Maximum suppression pool bulk temperature during normal continuous power operation shall be $\leq$80°F, except as specified in 3.7.A.1.e.

   d. Maximum suppression pool bulk temperature during RCIC, HPCI or ADS operation shall be $\leq$90°F, except as specified in 3.7.A.1.e.

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment

Suppression Pool

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

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

   c. Whenever there is indication of relief valve operation with the bulk temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

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

3.7 CONTAINMENT SYSTEMS (Con't)

e. In order to continue reactor power operation, the suppression chamber pool bulk temperature must be reduced to ≤80°F within 24 hours.

f. If the suppression pool bulk temperature exceeds the limits of Specification 3.7.A.1.d, RCIC, HPCI or ADS testing shall be terminated and suppression pool cooling shall be initiated.

g. If the suppression pool bulk temperature during reactor power operation exceeds 110°F, the reactor shall be scrammed.

h. During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cool down rates if the pool bulk temperature reaches 120°F.

i. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.17 psid, except as specified in j and k.

j. The differential pressure shall be established within 24 hours of placing the reactor in the run mode following a shutdown. The differential pressure may be reduced to less than 1.17 psid 24 hours prior to a scheduled shutdown.

k. The differential pressure may be reduced to less than 1.17 psid for a maximum of four (4) hours for maintenance activities on the differential pressure control system and during required operability testing of the HPCI system, the relief valves, the RCIC system and the drywell-suppression chamber vacuum breakers.

4.7 CONTAINMENT SYSTEMS (Con't)

e. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

f. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift when the differential pressure is required.

g. Suppression chamber water level shall be recorded at least once each shift when the differential pressure is required.

3.7 CONTAINMENT SYSTEMS (Con't)

l. If the specifications of Item i, above, cannot be met, and the differential pressure cannot be restored within the subsequent (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition in twenty-four (24) hours.

m. Suppression chamber water level shall be maintained between -6 to -3 inches on torus level instrument which corresponds to a downcomer submergence of 3.00 and 3.25 feet respectively.

n. The suppression chamber can be drained if the conditions as specified in Sections 3.5.F.3 and 3.5.F.5 of this Technical Specification are adhered to.

4.7 CONTAINMENT SYSTEMS (Cont'd)

3.7.A Primary Containment (Con't)

Primary Containment Integrity

2.a Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics test at power levels not to exceed 5 Mw(t).

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

(1) All manual containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

(2) At least one door in each airlock is closed and sealed.

(3) All blind flanges and manways are closed.

(4) All automatic primary containment isolation valves and all instrument line flow check valves are operable except as specified in 3.7.A.2.b.

(5) All containment isolation check valves are operable or at least one containment isolation valve in each line having an inoperable valve is secured in the isolated position.

4.7.A Primary Containment (Con't)

Primary Containment Integrity

2.a The primary containment integrity shall be demonstrated by performing Primary Containment Leak Tests in accordance with 10 CFR 50 Appendix J, as amended thru Sept. 22, 1980, with exemptions as approved by the NRC and exceptions as follows:

(1) The main steam line isolation valves shall be tested at a pressure $\geq$23 psig, and normalized to a value equivalent to 45 psig each operating cycle.

(2) Personnel air lock door seals shall be tested at a pressure $\geq$10 psig each operating cycle. Results shall be normalized to a value equivalent to 45 psig.

If the total leakage rates listed below are exceeded, repairs and retests shall be performed to correct the conditions.

(1) All double-gasketed seals: 10% L_t (x)

(2) All testable penetrations and isolation valves: 60% L_a (x)

(3) Any one penetration or isolation valve except main steam line isolation valves: 5% L_t (x)

(4) Any one main steam line isolation valve: 11.5 scf/hr @23 psig.

where x = 45 psig
L_t = .75 L_a
L_a = 1.0% by weight of the contained air @ 45 psig for 24 hrs.

3.7.A Primary Containment (Con't)

Primary Containment Isolation Valves

2.b. In the event any Primary Containment Isolation Valve that receives an automatic isolation signal listed in Table 3.7-1 becomes inoperable, at least one containment isolation valve in each line having an inoperable valve shall be deactivated in the isolated condition. (This requirement may be satisfied by deactivating the inoperable valve in the isolated condition. Deactivation means to electrically or pneumatically disarm, or otherwise secure the valve.)*

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

4.7.A Primary Containment (Con't)

Primary Containment Isolation Valves

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

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

b. At least once per quarter:

1. All normally open power operated isolation valves (except for the main steam line power operated isolation valves) shall be fully closed and reopened.

2. Trip the main steam isolation valves individually and verify closure time.

c. At least twice per week the main steam line power operated isolation valves shall be exercised by partial closure and subsequent reopening.

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

2.b.2 Whenever a primary containment isolation valve, that receives an automatic isolation signal, listed in Table 3.7-1 is inoperable, the position of the isolated valve in each line having an inoperable valve shall be recorded daily.

3.7.A Primary Containment (Con't)

4.7.A Primary Containment (Con't)

2.c Continuous Leak Rate Monitor

When the primary containment is inerted, the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

2.d Drywell Surfaces

The interior surfaces of the drywell and torus above the water line shall be visually inspected every refueling outage for evidence of deterioration.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.A Primary Containment	4.7.A Primary Containment
5.b. Within the 24-hour period subsequent to placing the reactor in the Run mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. De-inerting may commence 24 hours prior to a shutdown.	
6. If the specifications of 3.7.A.1 thru 3.7.A.5 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in Cold Shutdown condition within 24 hours.	

TABLE 3.7-1

PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION VALVES

GROUP	POWER OPERATED VALVE #	SYSTEM & DESCRIPTION	IPC/OPC	PENETRATION NUMBER	MAXIMUM OPERATING TIME (SEC)	NORMAL POSITION	ISOLATION POSITION
1	AO-203-1A	Main Steam Line "A" Isolation Valve	IPC	X-7A	3 ≤t≤ 5	Open	Closed
1	AO-203-2A	Main Steam Line "A" Isolation Valve	OPC	X-7A	3 ≤t≤ 5	Open	Closed
1	AO-203-1B	Main Steam Line "B" Isolation Valve	IPC	X-7B	3 ≤t≤ 5	Open	Closed
1	AO-203-2B	Main Steam Line "B" Isolation Valve	OPC	X-7B	3 ≤t≤ 5	Open	Closed
1	AO-203-1C	Main Steam Line "C" Isolation Valve	IPC	X-7C	3 ≤t≤ 5	Open	Closed
1	AO-203-2C	Main Steam Line "C" Isolation Valve	OPC	X-7C	3 ≤t≤ 5	Open	Closed
1	AO-203-1D	Main Steam Line "D" Isolation Valve	IPC	X-7D	3 ≤t≤ 5	Open	Closed
1	AO-203-2D	Main Steam Line "D" Isolation Valve	OPC	X-7D	3 ≤t≤ 5	Open	Closed
1	MO-220-1	Main Steam Drain Isolation Valve	IPC	X-8	30	Closed	Closed
1	MO-220-2	Main Steam Drain Isolation Valve	OPC	X-8	30	Closed	Closed
1[1]	AO-220-44	Reactor Water Sample Line Valve	IPC	X-41A	10	Open	Closed
1[1]	AO-220-45	Reactor Water Sample Line Valve	OPC	X-41A	10	Open	Closed
2[3,5]	AO-5033A	Drywell Purge/Makeup	OPC	X-26	10	Closed	Closed
2[5]	AO-5033B	Drywell Purge/Makeup	OPC	X-26	10	Closed	Closed
2[3,5]	AO-5033C	Torus Makeup	OPC	X-205	10	Closed	Closed
2[5]	AO-5035A	Drywell Purge/Makeup	OPC	X-26	5	Closed	Closed
2[5]	AO-5035B	Drywell Purge/Makeup	OPC	X-26	5	Closed	Closed
2[5]	AO-5036A	Torus Purge Inlet	OPC	X-205	5	Closed	Closed
2[5]	AO-5036B	Torus Purge Inlet	OPC	X-205	5	Closed	Closed
2[3,5]	AO-5041A	Torus Exhaust Bypass	OPC	X-227	10	Closed	Closed
2[3,5]	AO-5041B	Torus Exhaust Bypass	OPC	X-227	10	Closed	Closed
2[5]	AO-5042A	Torus Main Exhaust	OPC	X-227	5	Closed	Closed
2[5]	AO-5042B	Torus Main Exhaust	OPC	X-227	5	Closed	Closed
2[3,5]	AO-5043A	Drywell 2" Exhaust Bypass	OPC	X-25	10	Closed	Closed
2[3,5]	AO-5043B	Drywell 2" Exhaust Bypass	OPC	X-25	10	Closed	Closed
2[5]	AO-5044A	Drywell Purge Exhaust	OPC	X-25	5	Closed	Closed
2[5]	AO-5044B	Drywell Purge Exhaust	OPC	X-25	5	Closed	Closed
2[4]	A	TIP Ball - Ball Solenoid Valve	OPC	X-35	5	Closed	Closed
2[4]	B	TIP Ball - Ball Solenoid Valve	OPC	X-35	5	Closed	Closed
2[4]	C	TIP Ball - Ball Solenoid Valve	OPC	X-35	5	Closed	Closed
2[4]	D	TIP Ball - Ball Solenoid Valve	OPC	X-35	5	Closed	Closed

TABLE 3.7-1 (con't)

PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION VALVES

GROUP	POWER OPERATED VALVE #	SYSTEM & DESCRIPTION	IPC/OPC	PENETRATION NUMBER	MAXIMUM OPERATING TIME (SEC)	NORMAL POSITION	ISOLATION POSITION
2[6]	SV-5065-11A	H_2/O_2 Analyzer Supply	OPC	X-228J	2	Closed	Closed
2[6]	SV-5065-13B	H_2/O_2 Analyzer and Leak Detection Supply	OPC	X-50A-d	2	Open	Closed
2[6]	SV-5065-14A	H_2/O_2 Analyzer and Leak Detection Supply	OPC	X-106A-b	2	Open	Closed
2[6]	SV-5065-15B	H_2/O_2 Analyzer Supply	OPC	X-228C	2	Closed	Closed
2[6]	SV-5065-18A	H_2/O_2 Analyzer Supply	OPC	X-228J	2	Closed	Closed
2[6]	SV-5065-20B	H_2/O_2 Analyzer and Leak Detection Supply	OPC	X-50A-d	2	Open	Closed
2[6]	SV-5065-21A	H_2/O_2 Analyzer and Leak Detection Supply	OPC	X-106A-b	2	Open	Closed
2[6]	SV-5065-22B	H_2/O_2 Analyzer Sample	OPC	X-228C	2	Closed	Closed
2[6]	SV-5065-24A	H_2/O_2 and PASS Sample Return	OPC	X-46F	2	Open	Closed
2[6]	SV-5065-25B	H_2/O_2 Analyzer Return	OPC	X-228K	2	Closed	Closed
2[6]	SV-5065-26A	H_2/O_2 and PASS Sample Return	OPC	X-46F	2	Open	Closed
2[6]	SV-5065-27B	H_2/O_2 Analyzer Return	OPC	X-228K	2	Closed	Closed
2[6]	SV-5065-31B	H_2/O_2 Analyzer Supply	OPC	X-15E	2	Closed	Closed
2[6]	SV-5065-33A	H_2/O_2 Analyzer and PASS Supply	OPC	X-29E	2	Open	Closed
2[6]	SV-5065-35B	H_2/O_2 Analyzer Supply	OPC	X-15E	2	Closed	Closed
2[6]	SV-5065-37A	H_2/O_2 Analyzer and PASS Supply	OPC	X-29E	2	Open	Closed
2[6]	SV-5065-63	PASS Reactor Sample Jet Pump #15	OPC	X-40A-a	2	Closed	Closed
2[6]	SV-5065-64	PASS Reactor Sample Jet Pump #15	OPC	X-40A-a	2	Closed	Closed
2[6]	SV-5065-71	PASS Liquid Sample Return	OPC	X-228H	2	Closed	Closed
2[6]	SV-5065-72	PASS Liquid Sample Return	OPC	X-228H	2	Closed	Closed
2[6]	SV-5065-77	PASS Liquid Sample Return	OPC	X-228G	2	Closed	Closed
2[6]	SV-5065-78	PASS Liquid Sample Return	OPC	X-228G	2	Closed	Closed
2[6]	SV-5065-85	PASS Reactor Sample Jet Pump #5	OPC	X-40D-c	2	Closed	Closed
2[6]	SV-5065-86	PASS Reactor Sample Jet Pump #5	OPC	X-40D-c	2	Closed	Closed
2	CV-5065-91	Leak Detection and O_2 Analyzer Return	OPC	X-32A	5	Open	Closed
2	CV-5065-92	Leak Detection and O_2 Analyzer Return	OPC	X-32A	5	Open	Closed
2	AO-7011A	R/W Collection D/W Equip. Sump	OPC	X-19	20	Closed	Closed
2	AO-7011B	R/W Collection D/W Equip. Sump	OPC	X-19	20	Closed	Closed
2	AO-7017A	R/W Collection D/W Floor Sump	OPC	X-18	20	Closed	Closed
2	AO-7017B	R/W Collection D/W Floor Sump	OPC	X-18	20	Closed	Closed
2	MO-1001-21	RHR Discharge to Radwaste	OPC	None	20	Closed	Closed
2	MO-1001-32	RHR Discharge to Radwaste	OPC	None	20	Closed	Closed

TABLE 3.7-1 (con't)

PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION VALVES

GROUP	POWER OPERATED VALVE #	SYSTEM & DESCRIPTION	IPC/OPC	PENETRATION NUMBER	MAXIMUM OPERATING TIME (SEC)	NORMAL POSITION	ISOLATION POSITION
3[2]	MO-1001-29A	RHR Injection "A" Loop	OPC	X-51A	30	Closed	Closed
3[2]	MO-1001-29B	RHR Injection "B" Loop	OPC	X-51B	30	Closed	Closed
3	MO-1001-47	RHR S/D Cooling Suction Valve	OPC	X-12	30	Closed	Closed
3	MO-1001-50	RHR S/D Cooling Suction Valve	IPC	X-12	30	Closed	Closed
3	MO-1001-60	Reactor Vessel Head Spray	OPC	X-17	30	Closed	Closed
3	MO-1001-63	Reactor Vessel Head Spray	IPC	X-17	30	Closed	Closed
4	MO-2301-4	HPCI Steam to Turbine	IPC	X-52	25	Open	Closed
4	MO-2301-5	HPCI Steam to Turbine	OPC	X-52	25	Open	Closed
5	MO-1301-16	RCIC Steam to Turbine	IPC	X-53	20	Open	Closed
5	MO-1301-17	RCIC Steam to Turbine	OPC	X-53	20	Open	Closed
6	MO-1201-2	RWCU Suction	IPC	X-14	25	Open	Closed
6	MO-1201-5	RWCU Suction	OPC	X-14	25	Open	Closed
6	MO-1201-80	RWCU Return	OPC	X-9A	30	Open	Closed
7	MO-2301-33	HPCI Vacuum Breaker Isolation	OPC	X-219	30	Open	Closed
7	MO-2301-34	HPCI Vacuum Breaker Isolation	OPC	X-219	30	Open	Closed
	6-58A	Feedwater Line A Check Valve	IPC	X-9A	-	Open	Process
	6-58B	Feedwater Line B Check Valve	IPC	X-9B	-	Open	Process
	6-62A	Feedwater Line A Check Valve	OPC	X-9A	-	Open	Process
	6-62B	Feedwater Line B Check Valve	OPC	X-9B	-	Open	Process
	1101-15	SBLC Injection Check Valve	IPC	X-42	-	Closed	Process
	1101-16	SBLC Injection Check Valve	OPC	X-42	-	Closed	Process

NOTES FOR TABLE 3.7-1

Key: IPC - Inside Primary Containment
OPC - Outside Primary Containment

ISOLATION GROUPINGS

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

1. Reactor low-low water level
2. Main Steam Line high radiation
3. Main Steam Line high flow
4. Main Steam Line tunnel high temperature
5. Main Steam Line low pressure (in run mode only)
6. Reactor high water level (not in run mode, below 880 psig)

Group 2: The valves in this group are closed upon any one of the following conditions.

1. Reactor low water level
2. High drywell pressure

Group 3: The valves in this group are closed upon any one of the following conditions.

1. Reactor low water level
2. High reactor pressure
3. High drywell pressure

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

1. HPCI steam line high flow
2. HPCI steam line area high temperature
3. Low reactor pressure

Group 5: The valves in this group are closed upon any one of the following conditions.

1. RCIC steam line high flow
2. RCIC steam line area high temperature
3. Low reactor pressure

Group 6: The valves in this group are closed upon any one of the following conditions.

1. Reactor low water level
2. Cleanup area high temperature
3. Cleanup inlet high flow

Group 7: The valves in this group are closed on the following conditions:

1. Reactor Low Pressure and High Drywell Pressure

FOOTNOTES:

1 The Reactor Water Sample Line Isolation Valves initiate on a Group 1 or a Group 2 isolation signal.

2 MO-1001-29A&B Isolate on reactor low water level OR high drywell pressure if MO-1001-50 and MO-1001-47 are not fully closed AND reactor pressure not high (i.e., not >110 psig).

3 In addition to Group 2 isolation, these valves also receive a reactor low-low water level isolation which cannot be bypassed by utilizing the valves emergency open feature.

4 Reactor vessel low water level or high drywell pressure causes automatic withdrawal of TIP probe. When probe is withdrawn beyond these ball valves, these valves automatically close within 5 seconds.

5 In addition to Group 2 isolation, these valves also receive a Refueling Floor High Radiation isolation.

6 Isolation signals are overridden with the keylocked Control Switch in the "Override" position.

BASES:

3.7.A & 4.7.A Primary Containment

The integrity of the primary containment and operation of the core standby cooling system in combination limit the off-site 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 was made to this requirement during initial core loading and while the low power test program was being conducted and ready access to the reactor vessel was required. There was no pressure on the system at this time, thus greatly reducing the chances of a pipe break. Should this type of testing be necessary in the future, the reactor may be taken critical; however, restrictive operating procedures would be in effect again to minimize the probability of an accident. 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 secondary containment and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep off-site doses well below 10 CFR 100 limits.

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 1035 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 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 45 psig which is below the maximum of 62 psig. Maximum water volume of 94,000 ft^3 results in a downcomer submergency of 4'-0" and the minimum volume of 84,000 ft^3 results in a submergence approximately 12-inches less. Mark I Containment Long Term Program Quarter Scale Test Facility (QSTF) testing at a downcomer submergency of 3.25 feet and 1.17 psi wetwell to drywell pressure differential shows a significant suppression chamber load reduction and Long Term Program analysis and modifications are based on the above submergence and differential pressure.

Should it be necessary to drain the suppression chamber, provision will be made to maintain those requirements as described in Section 3.5.F BASES of this Technical Specification.

BASES:

3.7.A & 4.7.A Primary Containment

Experimental data indicates that excessive steam condensing loads can be avoided if the peak local temperature of the pressure suppression pool is maintained below 200°F during any period of relief-valve operation with sonic conditions at the discharge exit. Analysis has been performed to verify that the local pool temperature will stay below 200°F and the bulk temperature will stay below 160°F for all SRV transients. 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 pressure 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.

If a loss-of-coolant accident were to occur when the reactor water temperature is below approximately 330°F, the containment pressure will not exceed the 62psig code permissible pressure, even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperature above 212°F provides additional margin above that available at 330°F.

BASES:

3.7.A & 4.7.A Primary Containment

Primary Containment Testing

The primary containment pre-operational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The calculated peak drywell pressure is about 45 psig which would rapidly reduce to 27 psig following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig, equalizes with drywell pressure and therefore rapidly decays with the drywell pressure decay. The design pressure of the drywell and suppression chamber is 56 psig. The design leak rate is 0.5%/day at a pressure of 56 psig. Based on the calculated containment pressure response discussed above, the primary containment pre-operational test pressures were chosen. Also, 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 at the primary containment maximum allowable accident leak rate of 1.25%/day at 45 psig. Calculations made by the AEC staff with this leak rate and a standby gas treatment system filter efficiency of 95% for halogens and assuming the fission product release fractions stated in TID 14844, show that the maximum total whole body passing cloud dose is about 13 REM and the maximum total thyroid dose is about 110 REM at the site boundary over an exposure duration of two hours. The resultant doses that would occur for the duration of the accident at the low population zone distance of 4.3 miles are about 3 REM total whole body and 70 REM total thyroid. Thus, the doses reported are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site dose and 10 CFR 100 guidelines.

The maximum allowable test leak rate is 1.0%/day at a pressure of 45 psig. This value for the test condition was derived from the maximum allowable accident leak rate of 1.25%/day when corrected for the effects of containment environment under accident and test conditions. In the accident case, the containment atmosphere initially would be composed of steam and hot air whereas under test conditions the test medium would be air at ambient conditions. Considering the differences in mixture composition and temperatures, the appropriate correction factor applied was 0.8 as determined from the guide on containment testing.

Establishing the test limit of 1.0%/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly from the containment design value to take advantage of the design leak-tightness capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as built" condition is achieved by establishing the allowable operational leak rate. The allowable operational leak rate is derived by multiplying the maximum allowable leak rate or the allowable test leak rate by 0.75 thereby providing a 25% margin to allow for leakage deterioration which may occur during the period between leak rate tests.

BASES:

3.7.A & 4.7.A Primary Containment

The primary containment leak rate test frequency is based on maintaining adequate assurance that the leak rate remains within the specification. The leak rate test frequency is in accordance with 10 CFR 50 App. J as amended through Sept. 22, 1980.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a bolted double-gasketed penetration is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized. The personnel air lock is tested at 10 psig, because the inboard door is not designed to shut in the opposite direction.

Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss of coolant accident.

Group 1 - process lines are isolated by reactor vessel low-low water level in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in group 1 are also closed when process instrumentation detects excessive main steam line flow, high radiation,, low pressure, main steam space high temperature, or reactor vessel high water level.

Group 2 - isolation valves are closed by reactor vessel low water level or high drywell pressure. The group 2 isolation signal also "isolates" the reactor building and starts the standby gas treatment system. It is not desirable to actuate the group 2 isolation signal by a transient or spurious signal.

Group 3 - isolation valves can only be opened when the reactor is at low pressure and the core standby cooling systems are not required. Also, since the reactor vessel could potentially be drained through these process lines, these valves are closed by low water level.

Group 4 and 5 - process lines are designed to remain operable and mitigate the consequences of an accident which results in the isolation of other process lines. The signals which initiate isolation of group 4 and 5 process lines are therefore indicative of a condition which would render them inoperable.

BASES:

3.7.A & 4.7.A Primary Containment

Group 6 - process lines are normally in use and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from non-safety related causes. To protect the reactor from a possible pipe break in the system, isolation is provided by high temperature in the cleanup system area or high flow through the inlet to the cleanup system. Also, since the vessel could potentially be drained through the cleanup system, a low level isolation is provided.

Group 7 - The HPCI vacuum breaker line is designed to remain operable when the HPCI system is required. The signals which initiate isolation of the HPCI vacuum breaker line are indicative of a break inside containment and reactor pressure below that at which HPCI can operate.

The maximum closure time for the automatic isolation valves of the primary containment and reactor vessel isolation control system have been selected in consideration of the design intent to prevent core uncovering following pipe breaks outside the primary containment and the need to contain released fission products following pipe breaks inside the primary containment.

In satisfying this design intent an additional margin has been included in specifying maximum closure times. This margin permits identification of degraded valve performance, prior to exceeding the design closure times.

In order to assure that the doses that may result from a steam line break do not exceed the 10CFR100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds.

These valves are highly reliable, have low service requirements and most are normally closed. The initiating sensors and associated trip channels are also checked to demonstrate the capability for automatic isolation. The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate. More frequent testing for valve operability results in a greater assurance that the valve will be operable when needed.

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

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25 inch restricting orifice inside the primary containment. A program for periodic testing and examination of the excess flow check valves is in place.

Primary Containment Painting

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, approximately every 18 months, assures the paint is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.

BASES:

3.7.A & 4.7.A Primary Containment

Vacuum Relief

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and reactor building so that the structural integrity of the containment is maintained. The vacuum relief system from the pressure suppression chamber to reactor building consists of two 100% vacuum relief breakers (2 parallel sets of 2 valves in series). Operation of either system will maintain the pressure differential less than 2 psig; the external design pressure. One valve may be out of service for repairs for a period of seven days. If repairs cannot be completed within seven days, the reactor coolant system is brought to a condition where vacuum relief is no longer required.

The capacity of the 10 drywell vacuum relief valves is sized to limit the pressure differential between the suppression chamber and drywell during post-accident drywell cooling to the design limit of 2 psig. 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 5 psig vacuum; therefore, with two vacuum relief valves secured in the closed position and eight operable valves, containment integrity is not impaired.

Reactor operation is permissible if the bypass area between the primary containment drywell and suppression chamber does not exceed an allowable area. The allowable bypass area is based upon analysis considering primary system break area, suppression chamber effectiveness, and containment design pressure. Analyses show that the maximum allowable bypass area is 0.2 ft^2, which is equivalent to all vacuum breakers open 3/32". (See letters from Boston Edison to the Directorate of Licensing, dated May 15, 1973 and October 22, 1974)

Reactor operation is not permitted if differential pressure decay rate is demonstrated to exceed 25% of allowable, thus providing a margin of safety for the primary containment in the event of a small break in the primary system.

Each drywell suppression chamber vacuum breaker is equipped with three switches. One switch provides full open indication only. Another switch provides closed indication and an alarm on Panel C-7 should any vacuum breaker come off its closed seat by greater than 3/32". The third switch provides a separate and redundant alarm on Panel 905 should any vacuum breaker come off its closed seat by greater than 3/32". The two alarms above are those referred to in Section 3.7.A.4.a(3) and 3.7.A.4.d.

The water in the suppression chamber is used only for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

BASES:

3.7.A & 4.7.A Primary Containment

Inerting

The relatively small containment volume inherent in the GE-BWR pressure suppression containment and the large amount of zirconium in the core are such that the occurrence of a very limited (a percent or so) reaction of the zirconium and steam during a loss-of-coolant accident could lead to the liberation of hydrogen combined with an air atmosphere to result in a flammable concentration in the containment. If a sufficient amount of hydrogen is generated and oxygen is available in stoichiometric quantities, the subsequent ignition of the hydrogen in rapid recombination rate could lead to failure of the containment to maintain a low leakage integrity. The 4% oxygen concentration minimizes the possibility of hydrogen combustion following a loss-of-coolant.

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 primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least twice a week the oxygen concentration will be determined as added assurance. Mark I Containment Long Term Program testing showed that maintaining a drywell to wetwell pressure differential to keep the suppression chamber downcomer legs clear of water significantly reduced suppression chamber post LOCA hydrodynamic loads. A pressure of 1.17 psid is required to sufficiently clear the water legs of the downcomers without bubbling nitrogen into the suppression chamber at the 3.00 ft. downcomer submergence which corresponds to approximately 84,000 ft.^{3} of water. Maximum downcomer submergence is 3.25 ft. at operating suppression chamber water level. The above pressure differential and submergence number are used in the Pilgrim I Plant Unique Analysis.

BASES:

3.7.A & 4.7.A Primary Containment (Cont'd)

Post LOCA 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. A minimum of 1500 gallons of liquid N_2 in the storage tank assures that a three-day supply of N_2 for post-LOCA containment inerting is available. Since the inerting makeup system is continually functioning, no periodic testing of the system is required.

The Post-LOCA Containment Atmospheric Dilution (CAD) System is designed to meet the requirements of AEC Regulatory Guides 1.3, 1.7 and 1.29, ASME Section III, Class 2 (except for code stamping) and seismic Class I as defined in the PNPS FSAR.

In summary, the limiting criteria are:

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

2. Limit the buildup in the containment pressure due to nitrogen addition to less than 28 psig.

3. To limit the offsite dose due to containment venting (for pressure control) to less than 300 Rem to the thyroid.

By maintaining at least a 3-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.(1) The system design contains sufficient redundancy to ensure its reliability. Thus, it is sufficient to test the operability of the whole system once per operating cycle. The H_2 analyzers will provide redundancy for the drywell i.e., there are two H_2 analyzers for the Unit. By permitting reactor operation for 7 days with one of the two H_2 analyzers inoperable, redundancy of analyzing capability will be maintained while not imposing an immediate interruption in plant operation. Monthly testing of the analyzers using H_2 will be adequate to ensure the system's readiness because of the design. Since the analyzers are normally not in operation there will be little deterioration due to use. In order to determine H_2 concentration, the analyzers must be warmed up 6 hours prior to putting into service. This time frame is acceptable for accident conditions because a 4% H_2 level will not be reached in the drywell until 16 hours following the accident. Due to nitrogen addition, the pressure in the containment after a LOCA will increase with time. Under the worst expected conditions the containment pressure will reach 28 psig in approximately 45 days. If and when that pressure is reached, venting from the containment shall be manually initiated per the requirements of 10CFR50.44. The venting path will be through the Standby Gas Treatment system in order to minimize the off site dose.

(1) As listed in Pilgrim Nuclear Power Station Procedure No. 5.4.6 "Post Accident Venting".

BASES:

3.7.C - 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 shutdown 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.

Initiating reactor building isolation and operation of the standby gas treatment system to maintain at least a 1/4 inch of water negative pressure within the secondary containment provides an adequate test of the operation of the reactor building isolation valves, leak tightness of the reactor building and performance of the standby gas treatment system. Functionally testing the initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Performing these tests prior to refueling will demonstrate secondary containment capability prior to the time the primary containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

(next pg. is 177)

TABLE 6.9.1

	Area	Reference	Submittal Date
a.	Secondary Containment Leak Rate Testing (1)	4.7.C.1.c	Upon completion of each test (2)
b.	(Deleted)		
c.	(Deleted)		
d.	Gross Gaseous Release 0.05 Ci/sec for 48 Hours	4.8.B	Ten days after the release occurs
e.	Standby Liquid Control solution enrichment out of specification	3.4.C.3	Fourteen days after receipt of a non-complying enrichment report or lack of receipt of such a report within the required thirty days, if enrichment compliance cannot be achieved within seven days.

NOTES:

1. Each integrated leak rate test of the secondary containment shall be the subject of a summary technical report. This report shall include data on the wind speed, wind direction, outside and inside temperatures during the test, concurrent reactor building pressure, and emergency ventilation flow rate. The report shall also include analyses and interpretations of those data which demonstrate compliance with the specified leak rate limits.

2. The report shall be submitted approximately 90 days after completion of each test. Test periods shall be based on the commercial service date as the starting point.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 113

TO FACILITY OPERATING LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 BACKGROUND

The NRC, by letter dated July 2, 1984, requested the licensee (the Boston Edison Company) to update the Pilgrim Technical Specifications (TS) related to Primary containment testing for compliance with the requirements of Appendix J to 10 CFR 50 (Appendix J). By letter dated June 21, 1985, the licensee responded by submitting a proposed amendment to Appendix A of Operating License No. DPR-35 to modify a portion of the TS concerning primary containment tests. The staff reviewed the licensee's submittal and in a letter dated October 1, 1985, recommended approval of the TS changes except for the proposed deletion of Table 3.7-1 which lists major primary containment isolation valves.

As a result of the October 1, 1985 evaluation, the licensee reinstated Table 3.7-1 in the TS and also made additional changes to the TS to reflect amendments and plant modifications that had been made to comply with NRC'S TMI Action Plan and Multi-Plant Action (MPA) B-24 requirements. The revised TS was submitted by the licensee in a letter dated June 4, 1987, which completely superseded the unapproved portions of the licensee's submittal of June 21, 1985. Further revisions to the June 4, 1987 submittal were made in response to staff questions by letter dated August 13, 1987. In letters dated September 21 and December 8, 1987, the licensee provided five corrected pages to TS pages submitted on August 13, 1987. These corrections, of a nontechnical nature, resolved typographical errors and made the text consistent with recently approved license amendments.

The licensee's proposed TS changes include (1) revising the definition of frequently used terms, (2) modifying Table 3.2-B to add instrumentation for the two new HPCI vacuum breakers, (3) revising containment isolation valve listing in Table 3.7-1, and (4) making editorial changes and adding clarifications to facilitate the interpretation of the TS. The changes to the TS sections are itemized below, followed by the staff's assessment of each change.

2.0 EVALUATION

2.1 TS page 3, Definition M "Primary Containment Integrity":

The licensee proposes to add condition 5 in Definition M to further define containment interity and rewrite condition 4. Condition 4 is revised by adding an additional statement to require that at least one valve on each line having an inoperable valve shall be deactivated in the isolated condition. The added condition 5 states that all containment isolation check valves are operable or at least one containment isolation valve in each line having an inoperable valve is secured in the isolated position. The staff finds the proposed revision to definition M acceptable since both conditions 4 and 5 further define containment integrity in considering an inoperable valve in a line.

2.2 TS page 5b, Definition EE "Automatic Primary Containment Isolation Valves":

The licensee proposes to add Definition EE to the TS which states that automatic primary containment isolation valves are the isolation valves which receive an automatic primary conatinment group isolation signal. The staff finds the proposed definition acceptable since it defines the meaning of the automatic isolation valves.

2.3 TS page 48, Table 3.2-B "Instrumentation that Initiates or Controls the Core and Containment Cooling Systems":

The licensee proposes to add to Table 3.2-B two instrument channels per trip system for closure of High Pressure Coolant Injection (HPCI) vacuum breaker isolation valves due to HPCI modification. The HPCI system modification added two new vacuum breakers and a vacuum relief line to the HPCI system turbine exhaust pipe during refueling outage No. 7 (during shutdown period which commenced April 12, 1986). The new vacuum relief line was routed from the torus hydrogen recombiner tie-in pipe to the high point of the HPCI turbine exhaust pipe. The new line includes two vacuum relief check valves and two normally open motor operated containment isolation valves. The newly installed containment isolation valves have control and indication in the main control room, and are automatically closed on a combination of high drywell pressure and low reactor pressure signal (Group 7 isolation condition).

The purpose of the HPCI modification is to reduce hydrodynamic transients (water hammer) in the HPCI turbine exhaust pipe. Steam left in the exhaust pipe cools and condenses, thereby creating a vacuum in the pipe. The vacuum draws water from the torus into the exhaust line. When the HPCI turbine is restarted, a water slug travels down the pipe to the torus causing the water hammer transient. Implementation of vacuum breaker will correct the problem by not allowing significant vacuum (with respect to torus air pressure) to be developed in the HPCI turbine exhaust line. The licensee has previously submitted information to the NRC on design and installation of the HPCI vacuum breakers.

The staff has reviewed available information on the HPCI vacuum breaker modification, and discussed this modification with the licensee. The staff finds that the automatic closure trip system is needed to automatically close the vacuum breaker isolation valves. Consequently, the proposed revision to Table 3.2-B is acceptable.

2.4 TS pages 155, 155a, and 157a, Section 3.7.A "Primary Containment":

The licensee proposes to rewrite Subsection 3.7.A.2 by adding Subsections 3.7.A.2.a and 3.7.A.2.b, and revise Subsections 3.7.A.5 and 3.7.A.6.

Subsection 3.7.A.2.a, entitled "Primary Containment Integrity", is a revision of Subsection 3.7.A.2 with the additional five conditions added to define containment integrity. These conditions are (1) all manual isolation valves, which are not required to be open during accident conditions, are closed; (2) at least one door in each airlock is closed and sealed; (3) all blind flanges and manways are closed; (4) all automatic isolation valves and instrument line flow check valves are operable; and (5) all containment isolation check valves are operable or at least one containment isolation valve in each line having an inoperable valve is secured in the isolation position. The staff finds that the proposed Subsection 3.7.A.2.a provides the minimum requirements on containment isolation to maintain containment integrity which comply with the requirements specified in the Standard TS for BWRs, and therefore, is acceptable. (However, the condition for airlock isolation with one door is only applicable to limiting condition for operation when the airlock is being used for normal transit entry and exit through the containment.)

Subsection 3.7.A.2.b, entitled "Primary Containment Isolation Valves", is added to Section 3.7.A.2 by moving the existing TS Section 3.7.A.D.2 with additional requirements on valve deactivation in the isolation condition that must be applied in any line having an inoperable isolation valve. The subsection also specifies that closed isolation valves may be reopened on an intermittent basis under Operations Review Committee (ORC) administrative controls. The staff finds the proposed Subsection 3.7.A.2.b acceptable since it is a relocation of Subsection 3.7.D.2 with additional limitations and restrictions on the inoperable isolation valves which are not currently included in the TS.

The existing Subsection 3.7.5b is renumbered to Subsection 3.7.A.5b, and the existing Subsection 3.7.6 is renumbered to 3.7.A.6. The staff finds the proposed Subsections 3.7.A.5b and 3.7.A.6 acceptable since the changes are editorial to make the TS consistent.

2.5 TS pages 155 and 155a, Section 4.7.A "Primary Containment Integrity":

The licensee proposes to rewrite Subsection 4.7.A.2 for surveillance requirements by adding paragraphs 2.a, 2.b.1, 2.b.2, 2.c, 2.d, and deleting leakrate testing methods A and B which were related to tests prior to initial unit operation.

Paragraph 4.7.A.2.a, entitled "Primary Containment", is added to the Subsection. It requires that the primary containment be leakrate tested in accordance with Appendix J, with exceptions as approved by the NRC. The paragraph also addresses total leakage rates for: (1) double gasketed seals, (2) all testable penetrations and isolation valves, (3) any one penetration or isolation valve, and (4) any one main steam line isolation valve. These leakage criteria were relocated from the existing Subsection 4.7.A.f to the proposed Paragraph 4.7.A.2.a. The staff finds the proposed paragraph 4.7.A.2.a acceptable since the changes are editorial for clarity and consistency. Certain leakrate test provisions, such as main steam line isolation valve and personnel airlock testing at reduced pressures, were previously approved by the NRC in a letter dated July 2, 1984.

Paragraph 4.7.A.2.b, entitled "Primary Containment Isolation Valves", is added to the Subsection by using the content of existing Section 4.7.D. The new Subsection has two paragraphs, i.e., 2.b.1 and 2.b.2. Paragraph 2.b.1 uses the wording in Subsection 4.7.D.1, which addresses primary containment isolation valve surveillance requirements. Paragraph 2.b.2 uses the wording in Subsection 4.7.D.2, which requires that the position of the automatic isolation valve in each line having an inoperable valve be recorded daily. The staff finds the proposed Subsection 4.7.A.2.b acceptable since the changes are editorial to make the TS consistent.

Paragraph 4.7.A.2.c, entitled "Continuous Leakrate Monitors", is added to the Subsection by using the content in existing Subsection 4.7.A.2.g, which requires the primary containment to be continuously monitored for gross leakage while inerted. The change is editorial in nature for consistency, and therefore, is acceptable.

Paragraph 4.7.A.2.d, entitled "Drywell Surfaces", is added to the Subsection by using the content in existing Subsection 4.7.A.2.h which requires the interior of the drywell and torus above the water line to be visually inspected every refueling outage. The change is editorial in nature for consistency, and therefore, is acceptable.

2.6 TS pages 160 thru 164, Table 3.7-1 "Primary Containment and Reactor Vessel Isolation Valves":

The licensee has revised Table 3.7-1 to include all the changes resulting from previous staff reviews and plant modifications. The staff's assessment is addressed in each of the changes.

(1) The licensee proposes to change the title of the table from "Primary Containment Isolation Valves" to "Primary Containment and Reactor Vessel Isolation Valves". The licensee stated that certain valves in the table are also used to isolate the reactor vessel boundary lines. The staff finds the change clarifies the content of the table, and therefore, is acceptable.

(2) The licensee proposes to add "valve number", "penetration number", and "system description" to the table identifications. The staff finds these added columns of information to the table acceptable since it makes the table more complete.

(3) The licensee proposes to change valve position for the reactor water sample valves (MO-220-44/45) from normally closed to normally open. The safety function of these valves is to isolate the line upon receipt of Group 1 containment isolation signal. This line is used for continuous sampling of reactor coolant for crack arrest verification (intergranular stress corrosion cracking) and should stay open during modes 1, 2, and 3. The isolation signal for these valves during accident conditions remains unchanged. The staff finds the proposed change to valve position acceptable since the change would ensure the operability of the reactor water sampling system during normal operation and would not impact the operability of these isolation valves.

(4) The licensee proposes to change maximum closing time for valves AO-5035A/B (drywell purge makeup), AO-5036A/B (torus purge inlet), AO-5042A/B (torus main exhaust), and AO-5044A/B (drywell purge exhaust) from 15 seconds to 5 seconds.

The proposed change of the valve closing time is due to a purge and vent system modification made during refueling outage No. 6 (1984). The original purge and vent system was redesigned to comply with the requirements of NUREG-0737, Item II.E.4.2 and the guidance developed as part of the MPA B-24 concerning containment isolation dependability. The change of valve closing time reflects the modified purge and vent system design. The original 20-inch Rockwell butterfly valves have been replaced by the 8-inch Clow tricentric valve since the Clow valves can close against the buildup of containment pressure in the event of a LOCA. The closure time was 15 seconds for the 20-inch valves and is seconds for the new 8-inch valves. The purge and vent system modification was previously reviewed and approved by the NRC in a letter dated September 24, 1984.

In accordance with SRP 6.2.4, Branch Technical Position (BTP) 6.4, purge valve closure time should not exceed 5 seconds to assure compliance with 10 CFR 100 regarding offsite radiological consequences. Consequently, the staff finds the proposed change to the TS acceptable since the reduction of purge valve closure time meets the guidance of SRP 6.2.4, BTP 6.4. Furthermore, the faster closure feature of the valve to avoid debris causing improper valve closing would improve containment isolation reliability during accident conditions.

(5) The licensee proposes to change valve position for valve AO-5033B, AO-5035B, AND AO-5036A/B from normally open to normally closed. The change corrects errors in the current description of the normal valve positions and will not affect the operability of these valves. Therefore, the change is acceptable.

(6) The licensee proposes to add a total of 26 containment isolation valves (SV-5065-11A thru SV-5065-86, and CV-5065-91/92) to the H2/O2 system, post-accident sampling system (PASS), and leak detection system. These valves were installed to the systems during refueling outage No. 6 (1984) to meet TMI Action Plant (NUREG-0737) requirements.

The H2/O2 system was designed to comply with the requirements of NUREG-0737, Item II.F.1(6) to provide the capability for "on-line" monitoring of the hydrogen and oxygen concentrations within the inerted primary containment atmosphere, and for potential combustion gas mixture monitoring within the containment following a LOCA. The PASS was designed to comply with the requirements of NUREG-0737, Item II.B.3 to provide the capability of collecting diluted or undiluted liquid reactor coolant and gaseous grab samples from RHR and jet pumps. These system modifications include a reduction in the total number of primary containment isolation valves (several penetrations were cut and capped). The original 16 valves in the oxygen analyzer system were removed and deleted from the TS. The system modifications installed 24 faster closing solenoid globe valves to the PASS and H2/O2 system. The leak detection system uses the existing penetration and valves. All these valves will close upon receiving Group 2 containment isolation signal. The implementation and design of the post-accident sampling, monitoring, and analysis systems were previously reviewed by the NRC against the criteria identified in Item II.B.3 of NUREG-0737, and were approved by the staff in a letter dated July 1, 1985. The system installations were inspected by the NRC (Inspection Report No. 50-293/85-27, dated November 13, 1985).

The staff has reviewed available documents on these system modifications and finds the 26 isolation valves acceptable. Because some of these valves are upgraded replacements of the original gate valves that were accepted by the NRC during the licensing review whose isolation signal to close the valve remains unchanged, and some are the new solenoid valves that receive the same isolation signal, the use of higher quality solenoid globe valves would improve containment isolation dependability.

(7) The licensee proposes to list four Traveling Incore Probe (TIP) system ball valves (solenoid valves-A,B,C,D) in Table 3.7-1. These are the existing valves in the TIP system but have never been listed in the TS. The licensee had previously requested to exempt these ball valves from Type C test requirements but the request was denied by the staff in a letter (NRC/FCR SER) dated July 2, 1984. The staff finds the proposed TS change acceptable since the inclusion of these valves in the table would avoid missed TS surveillance requirements on these valves.

(8) The licensee proposes to add two vacuum breaker isolation valves (MO-2301-33/34) to the table to reflect the HPCI system modifications. The vacuum breaker modification was recommended by General Electric Company (GE) for all BWR plants (SIL No. 30, October 31, 1973), and has been the subject of NRC safety system functional inspection report 50-293/85-30, dated January 28, 1986 which identified the need for this modification. The system isolation logic was recommended by GE in Study EDE DRF #E41-00018-3, dated October 31, 1986, and is similar to the systems at other BWR plants. The proposed vacuum relief line and valves were approved by GE (GE letter G-HK-6-100, dated April 9, 1986). The two new motor operated containment isolation valves are normally open and will close within 30 seconds upon receiving a Group 7 isolation signal. The licensee stated that the closure time was assigned in considering line

size, class of the valve, and comparison with similar valves. Reopening the valve after isolation will require operator action from the control room. The staff has reviewed the available information on the vacuum breaker isolation valves and finds that the design of these valves meets the requirements of General Design Criterion (GDC) 55 and SRP Section 6.2.4. As required by GDC 55, these valves are both automatic isolation valves, and are located just outside containment. They meet the instrumentation, control, environmental, seismic and quality criteria appropriate to containment isolation valves. Therefore, these valves are acceptable.

(9) The licensee proposes to remove the two HPCI torus suction isolation valves (MO-2301-35/36) from Table 3.7-1 because the line they isolate terminates below the free water surface of the suppression pool and will remain so throughout the duration of any accident. The staff previously approved the licensee's request to exempt these valves from Type C testing (NRC letter of April 28, 1981). Consequently, the staff finds the proposed deleting of the HPCI torus suction isolation valves from the table acceptable.

(10) The licensee proposes to modify the notes of the table to separate Group 1 isolation condition 5 (main steam line low pressure or reactor vessel high water level) into condition 5 (main steam line low pressure) and condition 6 (reactor high water level). The staff finds that the proposed change (into conitions 5 and 6) is editorial to avoid misinterpretation of Group 1 isolation signal, and is acceptable.

(11) The licensee proposes to modify the notes by adding Group 7 isolation condition (high drywell pressure and low reactor pressure). The added Group 7 isolation condition is due to the HPCI vacuum breaker modification, which has been addressed in Section 2.3 and 2.6.(8) of this safety evaluation. Therefore, the proposed Group 7 isolation condition is acceptable.

(12) The licensee proposes to add footnotes No. 1 thru No. 6 at the end of the table. These added footnotes are for clarity of the table, and therefore, are acceptable.

2.7 SUMMARY

The staff has completed its review of the licensee proposed TS changes pertaining to containment systems. In the updated submittal, the licensee has revised or reinstated certain items in response to staff questions made during this review. Based on the review of the licensee's submittals and supporting documents, the staff has concluded that the proposed TS changes are acceptable.

3.0 ENVIRONMENTAL CONSIDERATIONS

This amendment involves a change in the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no signficant 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 published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. 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.

4.0 CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: J. Guo

Dated: January 21, 1988

AMENDMENT NO. 113 TO FACILITY OPERATING LICENSE DPR-35
PILGRIM NUCLEAR POWER STATION

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