UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-206

SOUTHERN CALIFORNIA EDISON COMPANY

AND

SAN DIEGO GAS & ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 58 to Provisional Operating License No. DPR-13, issued to Southern California Edison Company and San Diego Gas and Electric Company (the licensees), which revised the Technical Specifications for operation of the San Onofre Nuclear Generating Station Unit No. 1 (the facility) located in San Diego County, California. The amendment is effective 30 days from its date of issuance.

The amendment approves changes to the Appendix A Technical Specifications which incorporate certain of the TMI-2 Lessons Learned Category "A" requirements. These requirements concern (1) Emergency Power Supply Requirements, (2) Valve Position Indication, (3) Instrumentation for Inadequate Core Cooling, (4) Containment Isolation, (5) Shift Technical Advisor, and (6) Containment Sphere Hydrogen Detection and Control. Note that these Technical Specifications are to be implemented within 30 days.

The application for amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact that pursuant to 10 CFR §51.5(d)(4), an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated May 7, 1981, (2) Amendment No. 58 to License No. DPR-13, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Mission Viejo Branch Library, 24851 Chrisanta Drive, Mission Viejo, California. A single copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 6th day of November, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas V. Wambach

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TABLE 3.5.7-1

AUXILIARY FEEDWATER INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
a) Stm. Gen. Water Level-Low					
i. Start Motor Driven Pumps	3	2	2	1, 2, 3, 4	F
ii. Start Turbine-Driven Pump	3	2	2	1, 2, 3, 4	F

ACTION F - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required CHANNEL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION G - With more than one channel inoperable, an operator shall assume continuous surveillance and actuate manual initiation of auxiliary feedwater, if necessary. Restore the system to no more than one channel inoperable within 7 days, or be in HOT STANDBY within the following 6 hours and in COLD SHUTDOWN within the following 30 hours.

Amendment No. 58

TABLE 3.5.7-2

AUXILIARY FEEDWATER INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
a) Steam Generator Water Level-Low	≥ 5% of narrow range instrument span each steam generator	≥ 0% of narrow range instrument span each steam generator

3.6 Containment Systems

3.6.1 Containment Sphere

Applicability: Applies to the operating status of the containment sphere.

Objective: To ensure containment integrity.

Specification: A. Leakage

The reactor coolant system temperature shall not be increased above 200°F if the containment leakage exceeds the maximum acceptable values specified in Surveillance Standard 4.3.

B. Access to Containment

(1) Containment integrity shall not be violated unless the reactor coolant system is below 500 psig and a shutdown margin greater than 1% Δk/k with all rods inserted is maintained for the most reactive temperature.

(2) Containment integrity shall not be violated when the reactor coolant system is open to the containment atmosphere unless a shutdown margin greater than 5% Δ k/k is maintained with all control rods inserted.

(3) Positive reactivity changes shall not be made by rod drive motion or boron dilution whenever the containment integrity is not intact.

C. Internal Pressure

The reactor shall not be made critical, nor be allowed to remain critical, if the containment sphere internal pressure exceeds 0.4 psig, or the internal vacuum exceeds 2.0 psig.

Basis: The bases for the shutdown margins and 500 psig pressure are as follows:

Δk/k	Event	Basis for Adequacy
1% (Below 500 psig)	Violation of Containment	Safety injection system disarmed; no credible automatic or operator action could cause return to criticality.
5%	Open reactor coolant system	Provides adequate margin so that maintenance activities can be carried out with the reactor head removed.(1)

TABLE 3.6.2-1

POWER OPERATED OR AUTOMATIC CONTAINMENT ISOLATION VALVE SUMMARY

	DESCRIPTION	INSIDE SPHERE	ALIGNMENT*	OUTSIDE SPHERE	ALIGNMENT*
1.	Sphere Sump Discharge	CV-102 (SV-108)	B	CV-103 (SV-109)	A
2.	RCS Dr Tk Discharge	CV-104 (SV-110)	B	CV-105 (SV-111)	A
3.	RCS Dr Tk Vent	CV-106 (SV-112)	B	CV-107 (SV-113)	A
4.	N_2 to RCS Drain Tank and PRT	CV-536	A	CV-535	B
5.	ORMS 1211/1212 Sphere Sample Supply	CV-147 (SV-1212-7)	B	SV-1212-9	A
6.	ORMS 1211/1212 Sphere Sample Return	CV-146 (SV-1212-6)	B	SV-1212-8	A
7.	A Stm. Gen. Stm. Sample	None		SV-119	A
8.	B Stm. Gen. Stm. Sample	None		SV-120	A
9.	C Stm. Gen. Stm. Sample	None		SV-121	A
10.	A Stm. Gen. Blowdown Sample	None		SV-123	A
11.	B Stm. Gen. Blowdown Sample	None		SV-122	A
12	C Stm. Gen. Blowdown Sample	None		SV-124	A
13.	Service Water to Sphere	CV-537	A	CV-115 (SV-126)	B
14.	Service Air to Sphere	Check Valve		SV-125	A
15.	SI Loop C Vent	SV-702B	A	SV-702A	B
16.	SI Loop B Vent	SV-702D	A	SV-702C	B
17.	PRT Gas Sample	CV-948**	A	CV-949 (SV-949)	B
18.	RC Loop Sample	(CV-955, CV-956, CV-962)**	A	CV-957 (SV-957)	B
19.	Pressurizer Sample	(CV-951, CV-953)**	A	CV-992 (SV-992)	B
20.	Sphere Purge Air Supply	-		POV-9 (SV-29)	A
21.	Sphere Purge Air Outlet	-		POV-10 (SV-30)	A
22.	Sphere Equalizing/Sphere Vent	CV-116 (SV-27)	B	CV-10 (SV-28)	A
	Inst. Air Vent	CV-40 (SV-19)	B		
23.	Primary Makeup to Press Rlf. Tk	CV-533	A	CV-534	B
24.	Cont. Cooling Out	-	-	CV-515**	A
25.	Cont. Cooling In	-	-	CV-516**	B
26.	N_2 Supply to PORV	CV-532**	B	Check Valve	-
27.	Letdown	CV-525**	A	CV-526**	B
28.	Seal Water Return	CV-527**	A	CV-528**	B
29.	Hydrogen Monitoring System	SV-3004	B	SV-2004	A

* Logic Nest C, Train A is aligned to power train F; Logic Nest D, Train B is aligned to power train G.

** These valves do not receive an automatic containment isolation signal. They are operated by remote manual switch (RMS).

4.1.3 RADIOACTIVE GASEOUS PROCESS AND EFFLUENT MONITORING INSTRUMENTATION

(The applicable Technical Specifications remain under review and will be issued at a later date.)

4.1.4 CONTAINMENT ISOLATION INSTRUMENTATION

Applicability: Applies to instrumentation which actuates the containment sphere isolation valves, containment sphere purge and exhaust valves, and containment sphere instrumentation vent header valves.

Objective: To ensure reliability of the containment sphere isolation provisions.

Specification:

A. Each instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL TEST operations for the MODES and at the frequencies shown in Table 4.1.4-1.

B. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

Basis: The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

References:

(1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

TABLE 4.1.4-1

CONTAINMENT ISOLATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
Containment Isolation (Valves listed in Table 3.6.2-1)				
a) Manual	N.A.	N.A.	M(1)	1, 2, 3, 4
b) Containment Pressure-High	N.A.	R	M(2)	1, 2, 3
c) Sequencer Subchannels	N.A.	N.A.	M	1, 2, 3, 4
d) Safety Injection				
1) Containment Pressure-High	N.A.	R	M(2)	1, 2, 3
2) Pressurizer Pressure-Low	N.A.	R	M	1, 2, 3, 4
Purge and Exhaust Isolation (POV-9, POV-10, CV-10, CV-40, CV-116)				
a) Manual	N.A.	N.A.	M(1)	1, 2, 3, 4
b) Containment Radioactivity-High	S	R	M	1, 2, 3, 4

TABLE 4.1.4-1 (Continued)

TABLE NOTATION

(1) Manual actuation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual safeguards actuation shall receive a CHANNEL TEST at least once per 31 days.

(2) The CHANNEL TEST shall include exercising the transmitter by applying either a vacuum or pressure to the appropriate side of the transmitter.

4.1.5 Accident Monitoring Instrumentation

Applicability: Applies to the accident monitoring instruments shown in Table 4.1.5-1 for MODES 1, 2 and 3.

Objectives: To ensure the reliability of the accident monitoring instrumentation.

Specification: A. Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.1.5-1.

Basis: The surveillance requirements specified for these systems ensure that the overall functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

References: (1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

TABLE 4.1.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
Pressurizer Water Level	M	R
Auxiliary Feedwater Flow Indication*	M	R
Reactor Coolant System Subcooling Margin Monitor	M	R
PORV Position Indicator	M	R
PORV Block Valve Position Indicator	M	R
Safety Valve Position Indicator	M	R

* See footnote of Table 3.5.6-1.

4.1.6 Pressurizer Relief Valves

Applicability: Applies to the power operated relief valves (PORVs) and their associated block valves for MODES 1, 2 and 3.

Objective: To ensure the reliability of the PORVs and block valves.

Specification: A. Each PORV shall be demonstrated OPERABLE:

1. At least once per 31 days by performance of a CHANNEL TEST, which may include valve operation, and
2. At least once per 18 months by performance of a CHANNEL CALIBRATION.

B. Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel.

C. The backup nitrogen supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by transferring motive power from the normal air supply to the nitrogen supply and operating the valves through a complete cycle of full travel.

Basis: The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The air supply for both the relief valves and the block valves is capable of being supplied from a backup passive nitrogen source to ensure the ability to seal this possible RCS leakage path.

References:

(1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

4.1.7 Pressurizer

Applicability: Applies to pressurizer heaters and pressurizer water level for MODES 1, 2 and 3.

Objective: To ensure proper pressurizer water volume and to ensure the capability to energize the pressurizer heaters from the emergency diesel generator.

Specification:

A. The pressurizer water level shall be determined to be between 5% and 70% at least once per 12 hours.

B. The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by transferring power from the normal supply to the emergency diesel generator and energizing the heaters.

Basis: The requirement that the pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency diesel generator provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT STANDBY.

References:

(1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

4.1.8 Auxiliary Feedwater Instrumentation

Applicability: Applies to the instruments shown in Table 4.1.8-1.

Objective: To ensure reliability of automatic initiation of the auxiliary feedwater pumps.

Specification: A. Each instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL TEST operations for the MODES and at the frequencies shown in Table 4.1.8-1.

Basis: The surveillance requirements specified for this instrumentation ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

References: (1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

TABLE 4.1.8-1

AUXILIARY FEEDWATER INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
a) Steam Generator Water Level-Low	S	R	M	1, 2, 3, 4

4.3 Containment Systems

4.3.1 Containment Testing

Applicability: Applies to containment leakage.

Objective: To verify that leakage from the containment sphere is maintained within specified values.

Specifications: I. Integrated Leakage Rate Tests, Type A

A. Test Pressure

Peak pressure tests are conducted at a test pressure greater than or equal to 49.4 psig, and reduced pressure tests are conducted at a test pressure greater than or equal to 24.7 psig.

B. Acceptance Criteria

For the peak pressure test program the containment sphere leakage rate measured is less than 0.090 wt%/24 hours of the initial content of the containment air at the calculated peak pressure of 49.4 psig. For the reduced pressure test program to be conducted at 24.7 psig, the measured leakage rate shall be less than 0.064 wt%/24 hours of the initial content of the containment atmosphere at the calculated peak pressure of 49.4 psig.

The accuracy of each Type A test is verified by a supplemental test which (1) confirms the accuracy of the Type A test by verifying that the difference between supplemental and Type A test data is within 25% of 0.12 wt%/24 hours for the peak pressure test or 0.085 wt%/24 hours for the reduced pressure test, and (2) requires the quantity of air injected into the containment during the supplemental test to be equivalent to at least 75 percent of the total allowable leakage rate at 49.4 psig.

C. Frequency

(1) An integrated leak rate test shall be performed as follows:

(a) Within 24 months from the date of initial criticality.

(b) Within 26 months from the date of the test in "a" above.

(c) Within 39 months from the date of the test in "b" above.

(d) Within every 39 months from the date of the previous test

The intervals specified in a, b, c, and d may be varied within an allowance of plus-4 months and minus-8 months to coincide with planned shutdown. In the event it is determined during any one test that the containment leakage rate does not meet the acceptability limit specified in "B" above, the condition shall be corrected, a retest made, and the testing frequency shall revert back to item "a" of the above schedule.

II. Penetration Testing

The combined leakage rate of all penetrations and all containment isolation valves subject to leakage rate tests shall not exceed 0.072 wt%/24 hours off the initial content of the containment atmosphere at the calculated peak pressure of 49.4 psig.

A. Types D, E, and Electrical Penetrations

(1) Tests

Leakage tests of types D, E, and electrical penetrations through the containment sphere shall be performed at an initial pressure (beginning of test) of 49.4 psig.

(2) Frequency

For Type D penetration, testing shall be accomplished during shutdown when the reactor is depressurized if the test has not been performed within the previous 6 months but in no case at intervals greater than 2 years.

Type E and all electrical penetrations shall be tested at a frequency of at least every 6 months.

B. Personnel Air Locks

(1) Test

Leakage tests of personnel air locks shall be performed at an initial pressure (beginning of test) of approximately 10 psig.

(2) Frequency

During operation, personnel air locks shall be tested at a frequency of at least every 6 months.

C. Isolation Valve Testing

(1) Tests

All isolation valves shall be tested for leak rate characteristics. Isolation valves normally operating with pressure less than 50 psig shall be tested at an initial pressure (beginning of test) of 49.4 psig.

4.3.2 CONTAINMENT ISOLATION VALVES

Application: Applies to the containment isolation valves listed in Table 3.6.2-1 for MODES 1, 2, 3 and 4.

Objective: To ensure reliability of containment isolation valves.

Specification:

A. The isolation valves specified in Table 3.6.2-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test.

B. Each isolation valve specified in Table 3.6.2-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

1. Verifying that on containment isolation test signal, each automatic isolation valve actuates to its isolation position.

2. Verifying that on a containment radiation-high test signal, each purge supply and purge outlet automatic valve actuates to its isolation position.

Basis: The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

References: (1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

4.3.3 Hydrogen Monitors and Hydrogen Recombiners

Application: Applies to containment sphere hydrogen monitors and hydrogen recombiners for MODES 1 and 2.

Objective: To ensure reliability of the equipment and systems required for the detection and control of hydrogen gas.

Specification:

A. Each hydrogen monitor shall be demonstrated OPERABLE at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gases containing:

1. Two volume percent hydrogen, balance nitrogen.

2. Six volume percent hydrogen, balance nitrogen.

B. Each hydrogen recombiner system shall be demonstrated OPERABLE at least once per 6 months by verifying that the minimum heater sheath temperature increases to greater than or equal to 700°F within 90 minutes. Upon reaching 700°F, increase the power setting to maximum power for 2 minutes and verify that the power meter reads greater than or equal to 60 Kw.

C. Each hydrogen recombiner system shall be demonstrated OPERABLE at least once per 18 months by:

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

2. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure (i.e., loose wiring or structural connections, deposits or foreign materials, etc.), and

3. Verifying the integrity of all heater electrical circuits by performing a resistance to ground test following the test in Specification B above. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

Basis: The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit (or the purge system) is capable of controlling the expected hydrogen generation associated with radiolytic decomposition of water and corrosion of metals within containment. (Cumulative operation of the purge system with the heaters on for 10 hours over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters). These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March, 1971.

References: (1) Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March, 1971.

TABLE 6.2.2.2

MINIMUM SHIFT CREW COMPOSITION#

LICENSE CATEGORY QUALIFICATIONS	APPLICABLE MODES 1, 2, 3 & 4	5 & 6
SRO	1	1*
RO	2	1
Non-Licensed Auxiliary Operator	1	1
Shift Technical Advisor	1	None Required

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator limited to Fuel Handling, supervising CORE OPERATIONS.

#Shift crew composition may be one less than the minimum requirements for a period of time not to exceed 2 hours in order to accomodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2.2.2. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

6.3 Unit Staff Qualifications

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Health Physics Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 and (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

6.4 Training

6.4.1 A retraining and replacement training program for the faciity staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements of Section 27 of the National Fire Protection Association Code - 1976."

f. Records of in-service inspection performed pursuant to these Technical Specifications.

g. Records of Quality Assurance activities as required by the QA Manual.

h. Records of reviews performed for changes made to procedures or equipment or reviews or tests and experiments pursuant to 10 CFR 50.59.

i. Records of meetings of the OSRC and the NARC.

j. Records for Environmental Qualification which are covered under the provisions of paragraph 6.12.

6.10.3 The following records shall be retained for two years:

a. Records of facility radiation and contamination surveys.

b. Records of training of facility personnel.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 ENVIRONMENTAL QUALIFICATIONS

A. By no later than June 20, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License DPR-13 dated October 24, 1980.

B. By no later than December 1, 1980, complete and auditible records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.13 SYSTEMS INTEGRITY

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventative maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

6.14 IODINE MONITORING

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

6.15 BACKUP METHOD FOR DETERMINING SUBCOOLING MARGIN

The licensee shall implement a program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

1. Training of personnel, and

2. Procedures for monitoring.