BEFORE THE UNITED STATES NUCLEAR REGULATORY COMMISSION

Application of SOUTHERN CALIFORNIA EDISON COMPANY and SAN DIEGO GAS & ELECTRIC COMPANY for a Class 104(b) License to Acquire, Possess, and Use a Utilization Facility as Part of Unit No. 1 of the San Onofre Nuclear Generating Station	DOCKET NO. 50-206 Amendment No. 98

SOUTHERN CALIFORNIA EDISON COMPANY and SAN DIEGO GAS & ELECTRIC COMPANY, pursuant to 10 CFR 50.90, hereby submit Amendment No. 98.

This amendment consists of Proposed Change Nos. 103 and 104 to the Technical Specifications incorporated in Provisional Operating License No. DPR-13 as Appendices A and B. Proposed Change No. 103 is a request to revise Table 3.13.1 of Technical Specification 3.13 by the addition of one mechanical snubber in accordance with Technical Specification 3.13.E. Proposed Change No. 104 is a request to revise Technical Specification 4.16 to define an acceptance criterion for the imperfection depth at which plugging of sleeved tubes is required as a result of inservice inspection.

In the event of conflict, the information in this Amendment No. 98 supersedes the information previously submitted.

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Accordingly, it is concluded that (1) the proposed changes do not involve an unreviewed safety question as defined in 10 CFR 50.59, nor do they present significant hazards considerations not described or implicit in the Final Safety Analysis, and (2) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed change.

The addition of one mechanical snubber as requested by Proposed Change No. 103, submitted as part of Amendment No. 98, corrects the inadvertent omission of this item from Table 3.13.1. Since additions of this type are allowed by Technical Specification 3.13.E, it has been determined that no fee is required for submittal of this request.

As a result of discussions with the NRC staff and pursuant to 10 CFR 170.22, Proposed Change No. 104, submitted as part of Amendment No. 98, is to be included as part of the overall review of the steam generator repair program at San Onofre Unit 1 and is therefore determined to be a Class IV change. The basis for this determination is that the changes involve more than one safety and environmental issue.

Accordingly, a fee of $12,300 corresponding to this determination is required by 10 CFR 170.22.

Subscribed on this 5th day of May 1981

Respectfully submitted,

SOUTHERN CALIFORNIA EDISON COMPANY

By ______________________

L. T. Papay
Vice President

Subscribed and sworn to before me this

5th day of May 1981.

____

Notary Public in and for the County of
Los Angeles, State of California

My Commission Expires: Aug. 27, 1982

OFFICIAL SEAL
AGNES CRABTREE
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Charles R. Kocher
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California Edison Company

By ______________________
James A. Beoletto

Subscribed on this 29TH day of APRIL, 1981.

Respectfully submitted,

SAN DIEGO GAS & ELECTRIC COMPANY

By D.W. Gilman

D.W. Gilman
Vice President

Subscribed and sworn to before me this

29th day of April 1981.

Anne R. Schmidt
Notary Public in and for the County of
San Diego, State of California
Anne R. Schmidt
My Commission Expires: Oct. 11, 1983

OFFICIAL SEAL
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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of SOUTHERN CALIFORNIA EDISON COMPANY and SAN DIEGO GAS & ELECTRIC COMPANY (San Onofre Nuclear Generating Station Unit No. 1)	)	Docket No. 50-206

CERTIFICATE OF SERVICE

I hereby certify that a copy of Amendment No. 98 was served on the following by deposit in the United States Mail, postage prepaid, on the 5th day of May, 1981.

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Description of Proposed Change and Safety Analysis
Proposed Change No. 103 to the Technical Specifications
Provisional Operating License DPR-13

This request would revise Table 3.13.1 of the Technical Specifications for San Onofre Unit 1.

Reason for Proposed Change

While reviewing mechanical snubber inspection data for response to IE Bulletin 81-01, Mechanical Snubbers, it was discovered that snubber 1-734-SS-003 had been inadvertently omitted from Table 3.13.1 of the Technical Specification.

Existing Specifications

The existing Specifications are constituted in Section 3.13 of Technical Specifications for Provisional Operating License DPR-13.

Proposed Specifications

Technical Specification 3.13 would be revised to modify Table 3.13.1 by the addition shown on the following page. This change will rectify the error of omission made in the current Technical Specification.

Safety Analysis

The proposed revision to the Technical Specifications requires surveillance of an additional mechanical snubber. This surveillance activity is intended to decrease the probability that equipment failure will go undetected.

In accordance with IE Bulletin 81-01, snubber 1-734-SS-003 will be inspected prior to startup from this outage. Subsequent inspections would be in accordance with the Technical Specification, as modified by this change.

Accordingly, it is concluded that (1) the proposed change does not involve an unreviewed safety question as defined in 10CFR50.59, nor does it present significant hazards considerations not described or implicit in the Final Safety Analysis, and (2) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed change.

Proposed addition to Table 3.13.1

Snubber No.	Location	Line No.	Snubber in High Radiation Area During Shutdown*	Snubbers Especially Difficult To Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
**1-734-SS-003	Sphere Spray	734		X	X	

DESCRIPTION OF PROPOSED CHANGE AND SAFETY ANALYSIS
PROPOSED CHANGE NO. 104 TO THE TECHNICAL SPECIFICATIONS
PROVISIONAL OPERATING LICENSE DPR-13

This is a request to revise Appendix A Technical Specification 4.16, "INSERVICE INSPECTION OF STEAM GENERATOR TUBING."

Reason for Proposed Change

As a result of the steam generator tube sleeving modifications, an acceptance criterion must be defined for the imperfection depth at which plugging of sleeved tubes is required, when inservice inspection is performed.

Existing Specification

(1) Technical Specification 4.16.E.1.c currently reads:

"c. Plugging limit means the imperfection depth at or beyond which plugging of the tube must be performed. The plugging limit is equal to or greater than 50% of the nominal tube wall thickness."

(2) The Basis for Technical Specification 4.16 currently reads, in part:

"Wastage-type imperfections are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 50% of the tube nominal wall thickness. A plugging limit of 50% ensures that tube defects will not occur between inspection intervals."

Proposed Specification

(1) Technical Specification 4.16.E.1.C would be revised to read:

"c. Plugging limit means the imperfection depth at or beyond which plugging of the tube must be performed. The plugging limit is equal to or greater than 50% of the nominal tube wall thickness, except where sleeves are installed, in which case the plugging limit is equal to or greater than 54% of the nominal sleeve wall thickness."

(2) The Basis for Technical Specification 4.16 would be revised to read, in part:

"If a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 50% of the tube nominal wall thickness, except where sleeves are installed, in which case the plugging limit is 54% of the nominal sleeve wall thickness. A plugging limit of 50% for tubes and 54% for sleeves ensures that defects will not occur between inspection intervals."

The remainder of Technical Specification 4.16 would remain as constituted in Appendix A to Provisional Operating License No. DPR-13.

Safety Analysis

Steam generator tube and sleeve inspection is one of the measures which will be used following resumption of power operation to monitor the continued effectiveness of the steam generator repair. As a result of the steam generator repair and the type of tube degradation that may occur at the top of the tubesheet, the plugging limits for steam generator tubes remains the same (i.e., 50% or greater), except where sleeves are installed, the plugging limit will be 54% or greater as applied to the sleeve. As indicated in the document entitled, "Steam Generator Repair Report, Revision 1, San Onofre Nuclear Generating Station, Unit 1, March, 1981," which was submitted to the NRC on March 20, 1981, the plugging limit of 54% for the sleeve is based on structural requirements for maintaining sleeve integrity during accident conditions. The minimum wall thickness required during an accident is 36% of nominal wall thickness. A safety margin of 10% is added to account for the uncertainty in the inspection technique and possible degradation between inspection intervals, resulting in a minimum wall thickness of 46% and an imperfection depth of 54%.

Accordingly, it is concluded that, (1) the proposed change does not involve an unreviewed safety question as defined in 10CFR50.59, nor does it present significant hazard considerations not described or implicit in the Final Safety Analysis, and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change.

NRC-SONGS

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Accession	8105110326

DESCRIPTION OF PROPOSED CHANGE AND SAFETY ANALYSIS
PROPOSED CHANGE NO. 102 TO THE TECHNICAL SPECIFICATIONS
PROVISIONAL OPERATING LICENSE DPR-13

This is a request to revise Appendix A Technical Specifications and to amend the license by adding three license conditions.

Reason for Proposed Change

Subsequent to the accident at Three Mile Island, the NRC issued NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," July 1979, which required certain actions by light water reactor licensees, including modifications and additions of equipment. By letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees, in order to provide reasonable assurance that operation is maintained within the limits determined acceptable following the implementation of the TMI-2 Lessons Learned Category "A" items, the NRC transmitted model Technical Specifications. Proposed Change No. 102 is in compliance with the guidance and scope of the July 2, 1980, NRC letter.

Existing Specifications and License Conditions

The existing specifications are as constituted in Appendix A Technical Specifications. The existing license conditions are as constituted in section 3 of Provisional Operating License No. DPR-13.

Proposed Specifications and License Conditions

The existing specifications and license conditions would be revised as indicated in the Enclosure to this Proposed Change. The added or revised portions are identified by a bar in the margin.

Safety Analysis

The Technical Specification changes discussed in the enclosure provide specifications for certain TMI-2 Lessons Learned Category "A" items in the areas of equipment and administrative requirements, including actions considered appropriate if a limiting condition for operation cannot be met.

Proposed Change No. 102 involves:

1. Emergency Power Supply Requirements

   The pressurizer water level indicators, pressurizer relief and block valves, and pressurizer heaters are important in a post-accident situation. Adequate emergency power supplies ensure post-accident functioning of these components. The enclosed specifications will satisfy these requirements.

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2. Valve Position Indication

The installed system for indication of valve position is a diagnostic aid to the operator. Although the indicating system provides no automatic action, this system should be operable and periodic surveillance should be performed.

3. Instrumentation for Inadequate Core Cooling

4. Containment Isolation

These specifications include a Table of Containment Isolation Valves which reflect the diverse isolation signals which the design currently provides. Limiting conditions for operation and associated surveillance are included.

5. Auxiliary Feedwater Systems

This proposed change treats both initiation and indication of auxiliary feedwater flow.

6. Shift Technical Advisor

The specification related to minimum shift manning is revised to reflect the augmentation of a Shift Technical Advisor.

7. Containment Sphere Hydrogen Detection and Control

This proposed change includes specifications which govern hydrogen monitors and hydrogen recombiners. In a post-LOCA situation, these systems monitor hydrogen levels and maintain hydrogen concentration within containment below its flammable limit.

The limiting conditions for operation and the surveillance standards of Proposed Change No. 102 will provide assurance of the reliability and availability of equipment which could be required to mitigate the consequences of an accident.

The Bases of the Proposed Change provide greater detail to support and supplement this Safety Analysis.

The proposed license conditions are related to a system integrity measurements program, an improved iodine measurements capability, and a backup method for determining the subcooling margin of the reactor coolant system. These programs represent a continuing commitment to safe operation of the facility.

Accordingly, it is concluded that (1) the proposed change does not involve an unreviewed safety question as defined in 10CFR50.59, nor does it present significant hazard considerations not described or implicit in the Final Safety Analysis, and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change.

ENCLOSURE 1 OF PROPOSED CHANGE NO. 102

Portions of Proposed Change No. 83, submitted as Amendment No. 84 on September 12, 1979, are included herein to provide continuity in the numbering system of the Technical Specifications, and to submit an addition to Table 1.1, FREQUENCY NOTATION.

TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY(1)
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
BW	At least once per 14 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
P	Completed prior to each release.
N.A.	Not applicable.

(1) For each frequency, the allowable extension is 25%. The total allowable extension for three consecutive frequency intervals is 3.25 times the interval.

This page was submitted as part of Proposed Change No. 83.

The notation for "BW" is added here.

To Appendix A, Specification 3.5, Instrumentation and Control, the following Specifications 3.5.3 and 3.5.4 will be added:

3.5.3 RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION

3.5.4 RADIOACTIVE GASEOUS PROCESS AND EFFLUENT MONITORING INSTRUMENTATION

In Appendix A, Specification 4.1, Operational Safety Items, will be revised as follows:

4.1 EQUIPMENT, INSTRUMENTATION AND SAMPLING

4.1.1 Operational Safety Items

Tables 4.1.1 and 4.1.2 will be changed to tables 4.1.1.1 and 4.1.1.2 Specifications 4.1.2, Radioactive Liquid Effluent Instrumentation, and 4.1.3, Radioactive Gaseous Process and Effluent Monitoring Instrumentation, will be added as follows:

FOR REFERENCE ONLY

This page contains excerpts from Proposed Change No. 83.

Items 6, 7 and 9 of Table 3.5.1 will be deleted.

TABLE 3.5.1
INSTRUMENT OPERATING CONDITIONS

	Functional Unit	COLUMN I Minimum Operational Channels	COLUMN II Minimum Redundancy* Required	COLUMN III Required Operating Action if Column I or Column II Cannot be Met
1.	Nuclear Power-Critical	3	For 3-Channel Operation —1 For 4 Channel Operation —2	Maintain hot standby conditions.
	-Subcritical	3	1	Maintain hot standby if at least one source and one intermediate channel are available; otherwise maintain 10% Δk/k shutdown margin.
2.	Pressurizer Variable Low Pressure	2	1	Maintain load below 10% F. P.
3.	Pressurizer Fixed High Pressure	2	1	Maintain hot standby conditions.
4.	Pressurizer High Level	2	1	Maintain hot standby conditions.
5.	Reactor Coolant Flow — 3-Loop Operation	3	1**/2***	Maintain load below 10% F. P.
6.	Deleted.			

* Redundancy is defined as N-M, where N is the number of channels in operation, and M is the number of channels in operation which, when tripped, will cause an automatic shutdown.

** For operation at <50% of full power
*** For operation at >50% of full power.

TABLE 3.5.1 (continued)

INSTRUMENT OPERATING CONDITIONS

Functional Unit	COLUMN I Minimum Operational Channels	COLUMN II Minimum Redundancy* Required	COLUMN III Required Operating Action if Column I or Column II Cannot be Met
7. Deleted.			
8. Manual Trip	1		Maintain hot standby conditions.
9. Deleted.			
10. Steam Feed-Water Flow Mismatch	3	1	Operator shall assume continuous surveillance and actuate manual scram if required.

* Redundancy is defined as N-M where N is the number of channels in operation, and M is the number of channels in operation which, when tripped, will cause an automatic shutdown.

3.5.5 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. The instrumentation channels shown in Table 3.5.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.5.5-2.

B. With an instrumentation channel trip setpoint less conservative than the Allowable Values column of Table 3.5.5-2, declare the channel inoperable and apply the applicable Action requirement of Table 3.5.5-1 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint Value.

C. With an instrumentation channel inoperable, take the action shown in Table 3.5.5-1.

Basis: The operability of these instrumentation systems ensure that 1) the associated action will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available from diverse parameters.

The operability of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

References:

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

TABLE 3.5.5-1

CONTAINMENT ISOLATION INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
Containment Isolation (Valves listed in Table 3.6.2-1)					
a) Manual	2	1	2	1, 2, 3, 4	D
b) Containment Pressure-High	3/train	2/train	2/train	1, 2, 3	B
c) Sequencer Subchannels	2/sequencer	2/sequencer	2/sequencer	1, 2, 3, 4	A
d) Safety Injection					
1) Containment Pressure-High	3/train	2/train	2/train	1, 2, 3	E
2) Pressurizer Pressure-Low	3/train	2/train	2/train	1, 2, 3	E
Purge and Exhaust Isolation (POV-9, POV-10, CV-10, CV-40, CV-116)					
a) Manual	1	1	1	1, 2, 3, 4	C
b) Containment Radioactivity-High	1	1	1	1, 2, 3, 4	C

TABLE 3.5.5-1 (Continued)

TABLE NOTATION

ACTION STATEMENTS

ACTION A - With the number of OPERABLE channels one less than the Total Number of Channels, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.1.4.

ACTION B - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is demonstrated within 4 hours; one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.1.4.

ACTION C - With less than the Minimum Channels OPERABLE, operation may continue provided the containment purge and exhaust valves (POV-9 & POV-10) are maintained closed.

ACTION D - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION E - With the number of OPERABLE Channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 4 hours.

b. The Minimum Channels OPERABLE requirements are met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.1.4.

TABLE 3.5.5-2

CONTAINMENT ISOLATION INSTRUMENTATION TRIP SET POINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
Containment Isolation		
a) Manual	Not Applicable	Not Applicable
b) Containment Pressure-High	≤1.4 psig	≤2.0 psig
c) Sequencer Subchannels	Not Applicable	Not Applicable
d) Safety Injection		
1) Containment Pressure-High	≤1.4 psig	≤2.0 psig
2) Pressurizer Pressure-Low	≥1685 psig	≥1675 psig
Purge and Exhaust Isolation		
a) Manual	Not Applicable	Not Applicable
b) Containment Radioactivity-High	≤2 x inservice reading	≤2.5 x inservice reading

Item 17 of this Table will be deleted

TABLE 4.1.1 (continued)

Channels	Action	Minimum Frequency	
13. Boric Acid Tank Level	Calibration	At each refueling shutdown	17 12/20/74
	Test	Once per month during operation	
14. Residual Heat Pump Flow	Calibration	At each refueling shutdown	17 12/20/74
15. Refueling Tank Level	Calibration	At each refueling shutdown	17 12/20/74
	Test	Once per month during operation	
16. Volume Control Tank Level	Calibration	At each refueling shutdown	17 12/20/74
	Test	Once per month during operation	
17. DELETED			
18. Area Radiation Monitors	Calibration	Once per month	17 12/20/74
	Test	Once per day	
19. Hydrazine Tank level	Calibration	At each refueling shut-down	34 4/1/77
	Test	One per month during operation	

NOTE FOR REFERENCE: This table was renumbered Table 4.1.1.1 in Proposed Change No. 83.

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.

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.

3.5.6 Accident Monitoring Instrumentation

Applicability: Applies to the accident monitoring instruments shown in Table 3.5.6-1 for MODES 1, 2 and 3.

Objective: To ensure reliability of the accident monitoring instrumentation.

Specification:

A. The accident monitoring instrumentation channels shown in Table 3.5.6-1 shall be OPERABLE.

B. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.5.6-1, either restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

C. With the number of OPERABLE accident monitoring instrumentation channels less than the MINIMUM CHANNELS OPERABLE requirements of Tabel 3.5.6-1, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

Basis: The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessions Learned Task Force Status Report and Short-Term Recommendations."

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

TABLE 3.5.6-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
Pressurizer Water Level	3	2
Auxiliary Feedwater Flow Indication*	2 per steam generator	1 per steam generator
Reactor Coolant System Subcooling Margin Monitor	2	1
PORV Position Indicator	1/valve	1/valve
PORV Block Valve Position Indicator	1/valve	1/valve
Safety Valve Position Indicator	1/valve	1/valve

* Auxiliary feedwater flow indication for each steam generator is provided by one channel of steam generator level and one channel of auxiliary feedwater flow rate. These comprise the two channels of auxiliary feedwater flow indication for each steam generator.

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.

3.1.5 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 reliability of the PORVs and block valves.

Specification:

A. Two PORVs and their associated block valves shall be OPERABLE.

B. With one or more PORV(s) inoperable, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

C. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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.

4.1.6 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.

3.1.6 Pressurizer

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

Objective: To ensure that pressurizer heaters are available during a loss of offsite power condition.

Specification:

A. The pressurizer shall be OPERABLE with at least 125 kilowatts of pressurizer heaters and a water level between 5 percent and 70 percent.

B. With the pressurizer inoperable due to the loss of capability to energize the pressurizer heaters from an emergency diesel generator, either restore the capability to energize the pressurizer heaters from an emergency diesel generator within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

Basis: The requirement that 125 kw of 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.

(2) SCE letter dated October 17, 1979, from J. H. Drake to D. G. Eisenhut, "Responses to NRC Requirements Related to the Three Mile Island Accident," Item 2.1.1 of the enclosure.

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.

Appendix A, Specification 3.6, CONTAINMENT, will be retitled and subdivided as follows:

3.6 Containment Systems

3.6.1 Containment Sphere

Appendix A, Specification 4.3, CONTAINMENT TESTING, wil be retitled and subdivided as follows:

4.3 Containment Systems

4.3.1 Containment Testing

Specification 4.3.II.C.1 of the existing Technical Specifications reads:

C. Isolation Valve Testing

1. Tests

All isolation valves shall be tested for operability and 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.

Specification 4.3.II.C.1 will be changed to:

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.

3.6.2 Containment Isolation Valves

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

Objective: To provide assurance that containment isolation will function when initiated by appropriate sensors.

Specification:

A. The containment isolation valves specified in Table 3.6.2-1 shall be OPERABLE.

B. With one or more of the isolation valve(s) specified in Table 3.6.2-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

1. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or
2. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or
3. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or
4. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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.

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).

4.3.2 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.

C. Each power operated or automatic valve of Table 3.6.2-1 shall be determined to be OPERABLE when tested in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

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.

3.5.7 Auxiliary Feedwater Instrumentation

Applicability: Applies to automatic initiation of the auxiliary feedwater pumps.

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

Specification:

A. The instrumentation channels shown in Table 3.5.7-1 shall be OPERABLE with their trip setpoints set consistent with the Trip Setpoint column of Table 3.5.7-2.

B. With an instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.5.7-2, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.5.7-1 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint Value.

C. With one instrumentation channel inoperable, take the action shown in Table 3.5.7-1.

D. With more than one channel inoperable, take ACTION G of Table 3.5.7-1.

Basis: The OPERABILITY of the auxiliary feedwater instrumentation ensures that 1) the associated action will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of this instrumentation is required to provide the overall reliability, redundancy, and diversity assumed available for the protection and mitigation of accident and transient conditions. The operation of this instrumentation is consistent with the assumptions used in the accident analyses.

References:

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

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 4 hours.

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.

TABLE 3.5.7-2

AUXILIARY FEEDWATER INSTRUMENTATION TRIP SETPOINTS

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

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

On the following pages, copies of Proposed Change No. 87 and Proposed Change No. 90 are provided. Both of these proposed changes were submitted as part of SCE Amendment 88 which was transmitted on February 8, 1980 by letter from R. Dietch to H. R. Denton.

The existing technical specification 4.4.E should be deleted and Proposed Change No. 87 should be cancelled, as both are superseded by new technical specification 4.1.9 which follows.

Item 20, page 1, and paragraph 2, page 3, of Proposed Change No. 90 should be deleted, as both are superseded by new technical specification 4.1.8 which follows.

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Description of Proposed Change and Safety Analysis
Proposed Change No. 87 to the Technical Specifications
Provisional Operating License DPR-13

This request would revise Section 4.4.E. of the Appendix A Technical Specifications for San Onofre Unit 1.

Reason for Proposed Change

These changes are submitted in response to NRC Staff requests contained in Enclosure 1 to a November 15, 1979 letter from D. G. Eisenhut to J. H. Drake. These requests were based on the NRR Bulletins and Orders Task Force review of operating reactors in light of the accident at Three Mile Island, Unit 2.

Existing Specifications

The existing Specifications are constituted in Section 4.4 of Appendix A Technical Specifications for Provisional Operating License DPR-13.

Proposed Specifications

Technical Specification 4.4.E. would be revised to read as follows:

"E. Auxiliary Feedwater System

1. At least every fourteen (14) days when the reactor coolant system pressure is greater than 500 psig, the auxiliary feedwater pumps shall be started to demonstrate satisfactory operation.

2. At least once every thirty-one (31) days when the reactor coolant system pressure is greater than 500 psig, the electric driven auxiliary feedwater pump shall be started and the normal flow path motor operated discharge valve shall be opened using the automatic and the remote manual actuation circuitry. This testing may be done in conjunction with 4.4.E.1 above.

3. When the reactor coolant system pressure remains less than 500 psig for a period longer than fourteen (14) days, a flow test shall be performed to verify the normal flow path from the condensate storage tank to each steam generator using the motor driven auxiliary feedwater pump prior to increasing reactor coolant system pressure above 500 psig. The flow test shall be conducted with the auxiliary feedwater system valves in their normal alignment. As soon as steam becomes available, the steam driven auxiliary feedwater pump shall be started to demonstrate satisfactory operation.

4. At least once every thirty-one (31) days when the auxiliary feedwater system is required to be operable, an inspection shall be made to verify that manual valves in the auxiliary feedwater system suction piping and the normal path from the auxiliary feedwater pumps to the main feedwater header that could interrupt all AFW flow are locked open.

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5. At least once every eighteen months, all normally closed manual valves in the alternate auxiliary feedwater system suction line and in the emergency flow path from the auxiliary feedwater pumps to the steam generator feedwater lines shall be demonstrated operable."

Safety Analysis

The proposed revisions to the Technical Specifications require additional surveillance verifications of AFW system valve positions and component operability. This surveillance activity is intended to decrease the probability that equipment failure or system misalignment will go undetected while the AFW system is required to be in standby status.

Accordingly, it is concluded that (1) the proposed change does not involve an unreviewed safety question as defined in 10CFR50.59, nor does it present significant hazards considerations not described or implicit in the Final Safety Analysis, and (2) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed change.

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DESCRIPTION OF PROPOSED CHANGE AND SAFETY ANALYSIS PROPOSED CHANGE NO. 90 TO THE TECHNICAL SPECIFICATIONS PROVISIONAL OPERATING LICENSE DPR-13

This is a request to (1) revise the definition of Containment Integrity as contained in Section 1.0; (2) add requirements for calibration and testing of Auxiliary Feedwater Flow and Condensate Tank level instrumentation in Table 4.1.1 of Section 4.1, and; (3) revise the requirements for the containment isolation system design in Section 5.2 of the Appendix A, Technical Specifications for San Onofre Unit 1.

Reason for Proposed Change

These changes are submitted to provide revisions and additions to reflect station modifications being performed during the January, 1980 outage to complete the Category A NRC TMI Short Term Lessons Learned Requirements as described in our letters dated January 17 and 23, 1980.

Existing Specifications

The existing specifications are as constituted in Sections 1, 4, and 5 of the Appendix A, Technical Specifications for Provisional Operating License DPR-13.

Proposed Specifications

The definition of "Containment Integrity" in Technical Specifiation 1.0 would be revised to read:

"Containment Integrity":

Containment Integrity means that all of the conditions below are satisfied:

(1) All manual containment isolation valves (or blind flanges) are closed.

(2) The equipment door is properly closed.

(3) At least one door in each personnel air lock is properly closed.

(4) All automatic and remote manual containment isolation valves are operable."

Technical Specification 4.1 would be revised by adding items 20 and 21 to Table 4.1.1 to read:

	Channels	Action	Minimum Frequency
"20.	Auxiliary Feedwater flow	Test	Once per month during operation

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21.	Condensate Tank level	Calibration	At each refueling shutdown
		Test	Once per month during operation"

The last paragraph of Technical Specification 5.2 would be revised to read:

"The automatically actuated containment isolation valves shall be designed to close upon high-pressure in the containment (set point no higher than 5 psig) or upon safety injection initiation. In addition, design provisions shall prevent automatic reopening of any isolation valves upon reset of the containment isolation signal. The actuation system shall be designed such that no single component failure will prevent containment isolation if required."

Safety Analysis

Each of the proposed Technical Specification revisions discussed above is required as part of the implementation of the NRC TMI Short Term Lessons Learned Requirements. The basis for each revision is discussed below:

1. By letters dated December 17, 1979 and January 17, 1980, the results of the Essential/Non-essential study of the containment isolation systems were provided to the NRC as required by NUREG-0578, Section 2.1.4. Based on the results, two automatically isolated systems which were previously identified as non-essential systems have been identified as essential systems (i.e., those required to mitigate an accident or which, if unavailable, could increase the magnitude of the event). These systems are:

   a. The turbine plant cooling water supply and return lines may be required to support extended operation of the reactor coolant pumps since they supply cooling water to the reactor coolant pump enclosure air conditioning units, and

   b. The nitrogen supply line to the Pressurizer Relief Tank currently provides a redundant source of pneumatic motive power to the power-operated relief valves and will provide a similar function to their associated block valves.

   As discussed in our January 17, 1980 letter, these two systems will be modified to provide remote-manual containment isolation capability consistent with other essential systems. The revision of the "Containment Integrity" definition requires that the remote-manual containment isolation valves be operable, and allows them to be open during operating conditions which require "Containment Integrity."

   Based on our review of the Technical Specifications which might be affected by the implementation of the NRC TMI Short Term Lessons Learned Requirements, we have determined that the definition of "Containment Integrity" should have been revised as part of our thorough review of the San Onofre Unit 1 containment isolation design completed in

April, 1976 for the Sphere Enclosure Project and assessment of compliance with 10CFR50, Appendix J. As a result of this review, the containment isolation design was modified to improve the leak tightness capability of the containment in the event of an accident requiring containment isolation by installation of new automatic and remote-manual valves. These new valves, as well as existing remote-manual valves, were designated as containment isolation valves. The remote-manual valves provide the ability to isolate essential systems, if necessary, following an accident to improve the leak tightness of containment. However, the definition of "Containment Integrity" was not revised to reflect the improved containment isolation design utilizing both automatic and remote-manual valves as containment isolation valves.

2. The auxiliary feedwater flow test requirement and condensate tank level calibration and test requirements added to Table 4.1.1 specify the minimum frequency and type of surveillance to be applied to the instrumentation. These calibration and test requirements are consistent with existing surveillance requirements for instrumentation installed in other systems.

3. The revision to Containment Design features (Technical Specification 5.2) accurately describes the containment isolation design.

Based on the above, it is concluded that (1) the proposed change does not involve an unreviewed safety question as defined in 10CFR50.59, nor does it present significant hazard considerations not described or implicit in the Final Safety Analysis, and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change.

3.4.3 Auxiliary Feedwater System

Applicability: Applies to the motor driven auxiliary feedwater pump and the turbine driven auxiliary feedwater pump for MODES 1, 2 and 3.

Objective: To ensure the availability of auxiliary feedwater to remove decay heat.

Specification: A. Both steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE as follows:

1. One auxiliary feedwater pump capable of being powered from an emergency electrical power source, and

2. One auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

B. With one auxiliary feedwater pump inoperable, restore both auxiliary feedwater pumps (one capable of being powered from an emergency electrical power source and one capable of being powered by an OPERABLE steam supply system) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Basis: The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

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

4.1.9 Auxiliary Feedwater Pumps and Valves

Applicability: Applies to the motor driven auxiliary feedwater pump, the turbine driven auxiliary feedwater pump, and auxiliary feedwater valves for MODES 1, 2 and 3.

Objective: To ensure the reliability of the auxiliary feedwater system.

Specification:

A. Each auxiliary feedwater pump shall be demonstrated OPERABLE by testing each pump in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i).

B. At least once per 31 days, verify that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

C. Each auxiliary feedwater pump shall be demonstrated OPERABLE at least once per 18 months during shutdown by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of each auxiliary feedwater actuation test signal.

2. Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of each auxiliary feedwater actuation test signal.

Basis: The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

The electric driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 208 gpm at a pressure of 1100 psig to the entrance of the steam generators. The steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 260 gpm at a pressure of 1100 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

References:

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

3.6.3 Hydrogen Monitors and Hydrogen Recombiners

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

Objective: To ensure the capability to maintain the hydrogen concentration within the containment sphere below its flammable limit during post-LOCA conditions.

Specification:

A. Two independent containment hydrogen monitors shall be OPERABLE.

B. With one hydrogen monitor inoperable, restore the inoperable monitor to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

C. Two independent containment hydrogen recombiner systems shall be OPERABLE.

D. With one hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

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.

The hydrogen mixing systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

References:

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

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. One volume percent hydrogen, balance nitrogen.

2. Four 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 will be revised as follows:

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.

The following license conditions will be added to Provisional Operating License No. DPR-13, Section 3:

I. 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.

J. 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.

K. 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.