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February 17, 1989

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Subject: Docket No. 50-206
Supplement to Amendment Application No. 158
San Onofre Nuclear Generating Station
Unit 1

By letter dated December 8, 1988, SCE submitted Amendment Application No. 158 regarding installation of a third auxiliary feedwater (AFW) pump at San Onofre Unit 1. While implementing associated plant modifications, it was identified that the design requirements of NUREG-0737 (post-TMI upgrades), Item II.E.1.2, Part 2, had not been fully implemented in the design of the Steam Generator Wide-Range Level (SGWRL) indication system. The SGWRL indication provides a backup means for verifying Auxiliary Feedwater flow during post accident conditions. SCE committed to upgrade the SGWRL instrumentation in 1981-1982 to meet post-TMI design requirements by: 1) providing environmental qualification of appropriate components, and 2) providing a power supply having a battery backup. As indicated in SCE's letter dated January 9, 1989 submitting Licensee Event Report No. 88-020, the SGWRL indicators were not upgraded in accordance with these commitments.

In resolution of this, SCE is converting two existing environmentally qualified safety related trains of narrow range steam generator level to wide range. Enclosure 1 to this letter provides design details for this modification. Enclosure 2 to this letter provides a supplement to Amendment Application No. 158 to include technical specification changes associated with the conversion of the narrow range transmitters. SCE considers these additional changes to be consistent with our intent to provide a fully qualified, single failure proof AFW system. The safety analysis provided with Amendment Application No. 158, as supplemented in Enclosure 2, bounds these additional changes.

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In addition to the supplement to Amendment Application No. 158 discussed above, conversations with the NRC staff indicated the need for additional information regarding the proposed change to Technical Specification 3.4.4 which specifies the minimum volume of water in the Auxiliary Feedwater Storage Tank. The change requested in Amendment Application No. 158 is to increase the minimum volume of water required in the AFW tank from 150,000 gallons to 190,000 gallons. The basis for the request was to account for water usage not currently included in the design calculation, i.e., (1) spillage that occurs during a main feedwater line break with delayed isolation of the broken line due to a single failure, and (2) loss of cooling water via the AFW pump bearings. The NRC staff requested additional information regarding the quantity of water consumed by these items, and to expand on the basis for the assumptions used to quantify the water usage. Specific information was requested relating to the utilization of reactor coolant system loop delta-temperature indication to identify the broken loop, the basis for the assumed worst case duration for isolation of the broken loop (i.e., less than one hour), and the basis for the assumed quantity of water lost through cooling of the AFW pump bearings.

With respect to the quantity of water lost through a broken loop, including flow rate and duration of spillage prior to isolating the broken loop, the most limiting conditions for break location and assumed single failure were postulated. As discussed below, these conditions resulted in a maximum spillage rate of less than 190 gpm for a period not exceeding one hour. It is noted that under certain circumstances the SGWRL indicators provide a backup indication of AFW flow to the steam generators and also a backup method for identification of the broken loop for purposes of isolating the broken loop. However, these conditions do not result in the most limiting transient with respect to loss of AFW through the broken loop. As discussed below, the most limiting transient with respect to loss of water through the break was evaluated as Case D provided in Enclosure 2 of SCE to NRC letter dated November 20, 1987. The specifics of the Case D transient are briefly summarized below. In this transient the operators will utilize the reactor coolant system (RCS) loop delta-temperature indicators to identify the broken loop. The methodology for utilization of these instruments, and the basis for not using the SGWRL indicators are provided below.

The Case D transient assumed a main feedwater line break upstream of the in-containment check valve from 100% reactor power. This assumed break location will maintain the steam generators in a pressurized condition. A reactor trip will occur from a steam/feedwater flow mismatch signal. The acceptance criterion for this transient is that the AFW system provide at least 125 gpm to the intact steam generators within 30 minutes. For purposes of identification of the broken loop, the primary instruments would be the AFW flow transmitters due to the disparity of flows in the three loops. The broken loop would exhibit much greater flow than the two intact loops because of the pressure in the steam generators versus atmospheric pressure in the broken loop. Assuming a failure of one of the AFW flow transmitters, the backup instruments that will assist in locating, and isolating the broken loop are the RCS loop delta-T indicators.

Based on system response during the Case D transient, the water level in the steam generators may not recover into the wide range band for up to four hours. Therefore, these instruments will not provide indication of which steam generators are receiving AFW for up to four hours. The RCS loop delta-T indicators, however, will begin to exhibit temperature differentials on the two intact loops in less than one hour. As illustrated by the reactor coolant system hot and cold leg temperatures shown in Figures 23, 24, and 25 in the November 20, 1987 submittal, a differential between hot leg and cold leg temperatures becomes clearly evident on the two intact loops (Figures 24 and 25) prior to one hour into the event, while the broken loop (Figure 23) shows the convergence of the hot and cold leg temperatures at that point. Thus, utilization of the RCS Loop Delta-T indication will allow identification of the broken loop within one hour.

Prior to isolating the broken loop, it will be spilling at less than 190 gpm while the AFW system is operating, depending on break location and assumed single failure. The spillage rate is bounded by system design features that include flow restricting orifices. The design calculation conservatively assumes a spillage rate of 200 gpm for one hour. It is noted that notwithstanding break location or assumed single failure, the new AFW system provides sufficient water to the intact steam generators to meet the acceptance criteria as indicated in the Summary Table of Hydraulic Calculation Results provided in the San Onofre ESF Single Failure Report.

The revised minimum AFW tank volume requirement includes an allowance for the cooling of the AFW pump G10S bearings. The rate of cooling water flow to the bearings has been measured at less than 11 gpm. In addition, the nominal value recommended by the pump vendor, Dresser Industries Inc., is 4 to 6 gpm. Based on these values, the 15 gpm assumed in the AFW Tank volume calculation is conservative.

In summation, the revised AFW tank water volume requirement is based on providing the following emergency service demands necessary for plant cooldown to 350°F:

1. Decay Heat for 32 hours -	128,500 gallons
2. Cooldown to RHR cut-in (350°F) at 32 hours -	14,500
3. AFW Pump G10S Bearings (15 gpm for 32 hours) -	28,800
4. AFW Spillage (200 gpm for 1 hour) -	12,000
Total -	183,800 gallons

Based on these service demands, the proposed volume requirement of 190,000 gallons contains an additional margin of 6,200 gallons above the required volume. Also, a low level alarm will be provided at 198,000 gallons to alert operators to refill the tank before reaching the Technical Specification minimum volume requirement.

In reference to implementation of the third AFW pump, SCE will have all normal, abnormal, and emergency operating instructions revised to reflect operation of the new system and operators trained prior to startup from the current refueling outage. These procedure revisions will include the utilization of the RCS loop delta-temperature indicators as a backup means for identification of a broken feedwater line. It is noted that a minor design change in the AFW system has been implemented to allow flow adjustments through the new venturis. A small bypass line will be provided on the three downstream venturis to permit minor flow adjustments to maintain minimum flow requirements without exceeding water hammer limits.

In a separate but related issue regarding the new AFW system, SCE indicated in Proposed Change No. 184 that we would be conducting an overall AFW system test to verify system response. This testing would be performed in addition to the surveillances required by the technical specifications. The Train A test was to be conducted in Mode 1 below 10 percent power due to the use of the steam driven AFW pump and associated cooling effects on the reactor coolant and secondary coolant systems. It is now SCE's intent to conduct this Train A test in Mode 1 below 25 percent power due to the higher stability of plant systems during and subsequent to the test. Below 10% power the turbine and main generator are not connected to the grid. Thus, a reactor trip at 10 percent power is in effect. The cooling effect on the reactor coolant system will result in a slight power increase, potentially up to the 10 percent trip set point, depending on initial plant conditions and the duration of the test. Therefore, to avoid a potential inadvertent reactor trip during this test, it will now be performed below 25 percent power with the turbine and main generator connected to the grid. Additional information regarding this test is provided in Enclosure 3.

If you have any questions or require additional information, please let me know.

Subscribed on this 17th day of February, 1989.

Respectfully submitted,

SOUTHERN CALIFORNIA EDISON COMPANY

By: Kenneth P. Baskin
Kenneth P. Baskin
Vice President

Subscribed and sworn to before me this 17th day of February, 1989,

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

OFFICIAL SEAL
AGNES CRABTREE
Notary Public-California
LOS ANGELES COUNTY
My Comm. Exp. Sep. 14, 1990

cc: J. B. Martin, Regional Administrator, NRC Region V
F. R. Huey, NRC Senior Resident Inspector, San Onofre Units 1, 2 and 3
J. H. Hickman, California Department of Health Services

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 the Supplement to Amendment Application No. 158 was served on the following by deposit in the United States Mail, postage prepaid, on the 17th day of February, 1989.

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James A. Beoletto

CONVERSION OF STEAM GENERATOR NARROW RANGE LEVEL TRANSMITTERS TO WIDE RANGE

EXISTING CONFIGURATION

Each steam generator is currently provided with three channels of narrow range steam generator level and two channels of wide range level. Two of the narrow range channels are safety grade and environmentally qualified. These two channels provide the low level input signal to the two trains of Auxiliary Feedwater Auto Initiation Circuitry. The third narrow range channel provides a signal to the main feedwater control system. One of the wide range channels provides a high level signal to the main feedwater control system. The three narrow range channels and this wide range channel all provide control room indication. The remaining wide range channel provides level indication at the remote shutdown panel.

DESIGN DESCRIPTION

Existing Steam Generator narrow range level transmitters LT-2400 A, B, and C, and LT-3400 A, B, and C will be converted to provide wide range level indication. These transmitters will continue to provide input to the AFW bistable logic for the purpose of initiating an Auxiliary Feedwater Actuating Signal (AFWAS). These new wide range level indicators will provide an alternate safety related AFW flow indication in the Control Room. All existing wide range indicators in the Control Room, at panel C-38 and the AFW panel will be rescaled to 0-318" so that all wide range indications will be consistent.

DESIGN DETAILS

The proposed design change will convert existing narrow range steam generator level transmitters LT-2400 A, B, and C, and LT-3400 A, B, and C to wide range. Level transmitters LT-2400 A, B, and C will provide a safety grade environmentally qualified backup means of verification of AFW flow to the Steam Generators. This conversion will be implemented by modifying the transmitters, the instrument tubing configuration to the wide range instrument taps, and relocating the transmitters. The control logic functions of these instrument loops will not be changed. However, the AFWAS setpoint for these transmitters required recalculation to provide an equivalent initiation setpoint of 238" as was provided at the narrow range 5% level.

Presently, level transmitters LT-2400 A, B, C (Train A), and LT-3400 A, B, C (Train B) provide input to the two out of three steam generator low level logic for initiation of AFW Actuation Signal (AFWAS) and also provide narrow range indication in the Control Room via LI-2400 A, B, C, and LI-3400 A, B, and C. Design changes will convert these level transmitters to wide range for the purpose of providing safety related wide range level indication in the Control Room in addition to continuously providing input to the two out of three low steam generator level logic for initiation of AFWAS. This conversion requires an AFWAS setpoint change since the converted indicators will have scales in inches of level, not % of scale. The following is a representation of the equivalency of the two scales at their calibrated conditions. Conversions of the existing narrow range transmitters LT-2400 A, B, and C, and LT-3400 A, B, and C into wide range transmitters will introduce some additional instrument error. This factor has been taken into account in developing the new AFWAS setpoint.

Steam Generator Equivalent Levels:

Narrow Range		Wide Range
100%	Upper Level Tap	318"
85%	Turbine Trip	305"
70%	High Level Alarm	293"
50%	Hot Zero Power Level	275"
30%	Normal Operating Level	259"
26%	Top of Feedring - Low Level Alarm	256"
10%	Administrative Action Level	242"
5%	AFW Initiation	238"
0%	Lower N.R. Level Tap	233"
	Top of Tube Bundle	227"
	Lower W.R. Level Tap	0"

A new setpoint of 238" (hot calibration) has been selected as it is approximately equal to the existing 5% narrow range setpoint. The "Lead-Lag" train initiation aspects of the original design are not affected by this change.

To preclude operator confusion, all Control Room wide range indication (LI-2400 A, B, and C; LI-3400 A, B, and C; LI-450A, LI-451A and LI-452A; and LI-450C, LI-451C and LI-452C) will be provided with the same scale range of 0 to 318 inches. LI-450C, LI-451C and LI-452C currently have 0 - 400" scales. This change will provide 0 - 318" scales for all wide range instruments. Wide range transmitters LT-450A, LT-450C, LT-451A, LT-451C, LT-452A, LT-452C will be cold calibrated, while wide range transmitters LT-2400 A, B, and C, and LT-3400 A, B, and C will be hot calibrated. This configuration provides hot and cold redundant indications for operator use as channel checks.

The conversion of LT-2400 A, B, and C, and LT-3400 A, B, and C to wide range will result in changes to both the normal and emergency operating instructions. Revision of these procedures and training of personnel will be implemented prior to return to service from the Cycle X refueling outage.

Converting LT-2400 A, B, and C, and LT-3400 A, B, and C from narrow range steam generator level indication to wide range level indication provides redundant AFW flow indication required by the NRC SER on TMI Item No. II.E.1.2. Technical Specification Tables 3.5.6-1 and 4.1.5-1 require narrow range steam generator water level indication be operable for accident monitoring instrumentation. Converting these transmitters also affects Technical Specifications 3.1.2 and 4.1.2 which currently require narrow range steam generator level for MODE 1 thru 4 (and some MODE 5) plant operation configurations. The proposed change to convert the narrow range instruments to wide range instruments does not impact plant safety as the Safety Analysis takes credit for only wide range indication since for most events the steam generator water level is no longer within the narrow range band.

Converting LT-2400 A, B, and C, and LT-3400 A, B, and C from narrow range to wide range will also affect Technical Specification 3.5.7 which establishes the AFWAS initiation setpoints. Currently set at 5% of narrow range scale, the setpoint will be at 238" (hot calibration) on the wide range scale. The Technical Specifications will be revised to reflect the existing setpoints in inches.

The new wide range level AFWAS setpoint was calculated to account for the conversion from NR to WR (greater span) including instrument accuracy. The setpoint calculation does not consider allowance for adverse environmental instrument error due to heatup of the wide range level transmitter reference leg from a secondary side break (FWLB or SLB) inside containment. Reference leg heatup adds a positive error to the indicated wide range level causing AFW actuation to occur at a lower actual level. This is acceptable since the safety analysis for the limiting FWLB inside containment assumes AFW is initiated at 15 minutes when the steam generator level is approaching dryout and the indicated WR level is well below the AFWAS setpoint. Hence, decay heat removal is not affected and acceptance criteria are met. The conversion from NR to WR level instruments does not affect SLB analysis to determine environmental qualification parameters for equipment outside containment because of the normal instrument accuracy for WR level is bounded by the previous analysis value (4").

The WR level AFWAS nominal setpoint is 238" (equivalent to the existing NR level setpoint of 5% NR). Normal operating Steam Generator level is 30% NR level or 259" WR level. Thus there is 21" margin between the normal operating level and the AFWAS setpoint level. Normal WR level instrument accuracy is ± 2". Hence, 19" margin is available (considering maximum instrument error) which is sufficient to prevent spurious actuation.

LT-2400 A, B, and C, and LT-3400 A, B, and C previously did not share the same instrument taps. Upon converting these transmitters from narrow to wide range capability, they will now share a common lower tap, while having separate upper taps. A common lower tap does not introduce an unacceptable common mode single failure. Failure of one common tap renders WR level indication inoperable on only one steam generator. AFW actuation of both

AFW trains will still occur on low steam generator level in the remaining two steam generators (2 out of 3 logic). The safety analysis assumes AFW actuation at 15 minutes for a feedline break transient and 30 minutes for a loss of feedwater transient following manual restoration of power to the 4kV Bus (loss of offsite power assumed). If automatic AFW actuation does not occur, operator action can be credited to initiate AFW at 15 minutes. WR level instrumentation is also credited for post-accident monitoring, specifically to verify AFW flow to the steam generators and enable operator action to isolate AFW to a broken feedline. However, the normal AFW flow indicators provide the primary indication of AFW flow for these purposes, and WR level is considered a backup means.

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DESCRIPTION OF SUPPLEMENTAL CHANGES TO PROPOSED CHANGE NO. 184 TO THE TECHNICAL SPECIFICATIONS PROVISIONAL OPERATING LICENSE NO. DPR-13

The following is a supplemental request to revise Sections 3.1.2, "Operational Components," 3.5.6, "Accident Monitoring Instrumentation," 3.5.7, "Auxiliary Feedwater Instrumentation," 4.1.1, "Operational Safety Items," and 4.1.5, "Accident Monitoring Instrumentation" of the Appendix A Technical Specifications for the San Onofre Nuclear Generating Station, Unit 1.

DESCRIPTION OF SUPPLEMENTAL CHANGES

Subsequent to SCE's submittal of Proposed Change No. 184 (PCN 184) by letter dated December 8, 1988, several additional changes relating to the installation of the third auxiliary feedwater pump have been identified. Thus, the following Technical Specification changes are proposed in addition to those proposed by PCN 184:

1. Revise Specification 3.1.2, "Operational Components," and Table 4.1.2, "Minimum Equipment Check and Sampling Frequency" to remove references to narrow range steam generator level. These changes do not impact the level of water required in the steam generators, and are proposed to eliminate potential confusion in that the narrow range level transmitters will not be used to measure steam generator level.

2. Revise Technical Specification 3.5.6, "Accident Monitoring Instrumentation," to incorporate appropriate ACTION requirements for the Steam Generator Water Level instruments. Due to the function of these instruments in relation to the AFW system operation, these instruments will be subject to the more restrictive ACTION requirements of the AFW system. ACTION B of this specification has been revised to consolidate the 72 hour action requirement for the three types of instrumentation providing indication of AFW flow. ACTION C has been revised to ensure that compensatory actions are taken in the event that channels from different types of flow indication are declared inoperable at the same time.

3. Revise Table 3.5.6-1, "Auxiliary Feedwater Instrumentation Setpoints," to consolidate the three types of Auxiliary Feedwater Flow Indications, namely the AFW flow rate, the steam generator water level, and the RCS loop delta-T indication into one location in the table. As proposed herein, the table reflects the use of these three different types of instruments which are used to verify flow to the steam generators and identify a broken loop.

4. Revise Table 3.5.7-2, "Auxiliary Feedwater Istrumentation Setpoints," to change the Setpoint and Allowable Value for the Steam Generator Water Level-low instrument channel. This change is proposed due to the utilization of wide range steam generator level transmitters to initiate auxiliary feedwater. The level in the steam generators at which auxiliary feedwater is initiated is not being revised. By using the wide range transmitters, however, the reference point for initiation will change due to the different instrument span.

5. Revise Table 4.1.5-1, "Accident Monitoring Instrumentation Surveillance Requirements," to consolidate the three types of AFW flow indication into one location in the table. This includes incorporating surveillance requirements for Reactor Coolant System Loop Delta-T Indication. Limiting Conditions for Operation for this channel were proposed as part of PCN 184 and included in Table 3.5.6-1. Associated surveillance requirements for this channel were inadvertently omitted from PCN 184 and are therefore proposed by this supplement. In addition, this change will specify utilization of the wide range level transmitters for monitoring steam generator level in lieu of the existing narrow range transmitters. These changes are proposed to be consistent with the changes to Table 3.5.6-1.

EXISTING TECHNICAL SPECIFICATIONS

See Attachment 1.

PROPOSED TECHNICAL SPECIFICATIONS

See Attachment 2. It is noted that Attachment 2 contains the changes proposed herein, in addition to those changes proposed by PCN 184. The previous changes are included, but change bars are used only to identify the additional changes proposed herein.

SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

The analysis provided by PCN 184, submitted as Amendment Application No. 158, bounds the additional changes proposed herein. The installation of the third auxiliary feedwater pump and associated system modifications are being implemented to eliminate single failure susceptibilities and provide a fully qualified auxiliary feedwater system. During implementation of plant modifications, SCE determined that the design requirements of NUREG-0737 had not been fully implemented in the design of the steam generator wide range level indication system. The existing instrumentation was not upgraded to provide environmental qualification and battery backup as committed by SCE

in 1981-1982 in response to post-TMI upgrades. Therefore, to satisfy this commitment, and to provide a fully qualified auxiliary feedwater system, it is necessary to provide safety related, environmentally qualified Train A wide range steam generator level indication with battery backup as backup indication of auxiliary feedwater flow. Therefore, plant modifications to convert both Trains of safety related, environmentally qualified narrow range level transmitters to wide range will be implemented prior to startup from the Cycle 10 refueling outage. Associated technical specification changes to reflect these plant modifications are proposed herein. The changes are consistent with the intent of PCN 184 to provide a fully qualified auxiliary feedwater system and thus are bounded by the significant hazards consideration analysis provided therein.

SAFETY AND SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the Safety Evaluation provided in Amendment Application No. 158, and the information provided above, it is concluded that: (1) the supplemental changes to Proposed Change No. 184 do not involve a significant hazards consideration as defined by 10CFR50.92; and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change.

Attachment 1 - Existing Technical Specifications

Attachment 2 - Supplemental Changes to Proposed Change No. 184.

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Attachment 1

Existing Technical Specifications

3.1.2 OPERATIONAL COMPONENTS

APPLICABILITY: Applies to the operating status of the reactor coolant system equipment and related equipment. For the applicable surveillance requirements, see Table 4.1.2. 80 10/4/84

OBJECTIVE: To identify those conditions of the reactor coolant system necessary to ensure safe reactor operation.

SPECIFICATIONS:

A. At least one pressurizer safety valve shall be operable or open when the reactor head is on the vessel, except for hydrostatic tests.

B. The reactor shall not be made critical or maintained critical unless both pressurizer safety valves are operable.

C. During Modes 1 and 2 and in Mode 3 with reactor trip breakers closed, all three reactor coolant loops and their associated steam generators and reactor coolant pumps shall be in operation. With less than the above required coolant loops in operation, be in at least Hot Standby with reactor trip breakers open within 1 hour, except as modified by Specification D below. 104 6/9/88

D. The limitations of Specification C may be suspended as follows:

1. During Modes 1 and 2, operation may be conducted with 0, 1, 2 or 3 reactor coolant pumps operating at less than 5% of full power for purposes of conducting low power physics testing.

2. During Modes 1 and 2 and in Mode 3 with reactor trip breakers closed, operation may be conducted for less than 24 consecutive hours with one or two reactor coolant pumps operating if reactor power is less than 10% of full power.

E. During Mode 3 with the reactor trip breakers open, the following specifications shall apply:

1. At least two of the reactor coolant loops listed below shall be operable: 80 10/4/84

a. Reactor coolant loop A and its associated steam generator and reactor coolant pump.

b. Reactor coolant loop B and its associated steam generator and reactor coolant pump.

c. Reactor coolant loop C and its associated steam generator and reactor coolant pump.

2. At least one of the above reactor coolant loops shall be in operation.*

3. With less than the above required reactor coolant loops operable, restore the required loops to operable status within 72 hours or be in Hot Shutdown within the next 12 hours.

4. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the reactor coolant system and immediately initiate corrective action to return the required reactor coolant loop to operation.

F. During Mode 4, the following specifications shall apply: 80 10/4/84

1. At least two of the reactor coolant loops/residual heat removal (RHR) trains listed below shall be operable:

   a. Reactor coolant loop A and its associated steam generator and reactor coolant pump.

   b. Reactor coolant loop B and its associated steam generator and reactor coolant pump.

   c. Reactor coolant loop C and its associated steam generator and reactor coolant pump.

   d. Residual heat removal (RHR) pump G-14A and one associated RHR train.

   e. Residual heat removal (RHR) pump G-14B and one associated RHR train.

2. At least one of the above loops/trains shall be in operation.**

* All reactor coolant pumps may be de-energized for up to one hour provided (a) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (b) core outlet temperature is maintained at least 40°F below saturation temperature.

** All reactor coolant pumps and residual heat removal pumps may be de-energized for up to one hour provided (a) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (b) core outlet temperature is maintained at least 40°F below saturation temperature.

3. With less than the above required loops/trains operable immediately initiate corrective action to return the required loops/trains to operable status as soon as possible; if the remaining operable loop/train is an RHR train, be in Cold Shutdown within 24 hours.

4. With no loop or train in operation, suspend all operations involving a reduction in boron concentration of the reactor coolant system and immediately initiate corrective action to return one required loop or train to operation.

G. During Mode 5 with reactor coolant loops filled, the following specifications shall apply:

1. At least one residual heat removal (RHR) train shall be operable and in operation*, and either

a. One additional RHR train shall be operable,** or

b. The secondary side water level of at least two steam generators shall be greater than or equal to 256 inches of narrow range on cold calibrated scale. 80 10/4/84

2. With less than the above required loops/trains operable, or with less than the required steam generator level, immediately initiate corrective action to return the required loops/trains to operable status or to restore the required level as soon as possible.

3. With no RHR train in operation, suspend all operations involving a reduction in boron concentration of the reactor coolant system and immediately initiate corrective action to return the required RHR train to operation.

H. During Mode 5 with reactor coolant loops not filled, the following specifications shall apply:

* The RHR pump may be de-energized for up to one hour provided (a) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (b) core outlet temperature is maintained at least $40^{o}F$ below saturation temperature.

** One RHR train may be inoperable for up to 2 hours for surveillance testing, provided the other RHR train is operable and in operation.

3.5.6 ACCIDENT MONITORING INSTRUMENTATION

APPLICABILITY: MODES 1, 2 and 3. 83 11/2/84

OBJECTIVE: To ensure reliability of the accident monitoring instrumentation. 64 12/16/81

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

ACTION:

A. 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.

B. With the number of OPERABLE accident monitoring instrumentation channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 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. 83 11/2/84

C. The provisions of Specification 3.0.4 are not applicable.

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 Lessons Learned Task Force Status Report and Short-Term Recommendations." 64 12/16/81

References:

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

(2) NRC letter dated November 1, 1983, from D. G. Eisenhut to all Pressurized Water Reactor Licensees, NUREG-0737 Technical Specification (Generic Letter No. 83-37).

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/steam generator	1/steam generator
Reactor Coolant System Subcooling Margin Monitor	2	1
PORV Position Indicator (Limit Switch)	1/valve	1/valve
PORV Block Valve Position Indicator (Limit Switch)	1/valve	1/valve
Safety Valve Position Indicator (Limit Switch)	1/valve	1/valve
Containment Pressure (Wide Range)	2	1
Steam Generator Water Level (Narrow Range)	1/steam generator	1/steam generator
Refueling Water Storage Tank Level	1	1
Containment Sump Water Level (Narrow Range)**	2	1
Containment Water Level (Wide Range)	2	1
Neutron Flux (Wide Range)	2	1

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

** Operation may continue up to 30 days with one less than the total number of channels OPERABLE.

TABLE 3.5.7-2

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
a. Manual Actuation	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. 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

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11/7/84

TABLE 4.1.2
MINIMUM EQUIPMENT CHECK AND SAMPLING FREQUENCY

		Check	Frequency	
1a. Reactor Coolant Samples	1.	Gross Activity Determination	At least once per 72 hours. Required during Modes 1, 2, 3 and 4.	96 3/5/87
	2.	Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days. Required only during Mode 1.	
	3.	Spectroscopic for E(1) Determination	1 per 6 months(2) Required only during Mode 1.	74 12/6/84
	4.	Isotopic Analysis for Iodine Including I-131, I-133, and I-135.	a) Once per 4 hours,(3) whenever the specific activity exceeds 1.0 μCi/gram DOSE EQUIVALENT I-131 or 100/ E (1) μCi/gram.	
			b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	38 12/20/77
	5.	Boron concentration	Twice/Week	

(1) E is defined in Section 1.0. 117 12/13/88

(2) Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

(3) Until the specific activity of the reactor coolant system is restored within its limits.

TABLE 4.1.2 (continued)

		Check	Frequency	
1.b	Secondary Coolant Samples	1. Gross Activity Determination	At least once per 72 hours. Required only during Modes 1, 2, 3 and 4.	
		2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit. Required only during Modes 1, 2, 3 and 4.	74 12/6/83
			b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit. Required only during Modes 1, 2, 3, and 4.	

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TABLE 4.1.2 (continued)

			Check	Frequency	
2.	Safety Injection Water Samples	a.	Boron Concentration	Monthly when the reactor is critical and prior to return of criticality when a period of subcriticality extends the test beyond 1 month	12 9/17/73
3.	Control Rod Drop	a.	Verify that all rods move from full out to full in, in less than 2.44 seconds	At each refueling shutdown	101 4/26/88
4.	(Deleted)				61 6/11/81
5.	Pressurizer Safety Valves	a.	Pressure Setpoint	At each refueling shutdown	
6.	Main Steam Safety Valves	a.	Pressure Setpoint	At each refueling shutdown	
7.	Main Steam Power Operated Relief Valves	a.	Test for Operability	At each refueling shutdown	
8.	Trisodium Phosphate Additive	a.	Check for system availability as delineated in Technical Specification 4.2	At each refueling shutdown	
9.	Hydrazine Tank Water Samples	a.	Hydrazine concentration	Once every six months when the reactor is critical and prior to return of criticality when a period of subcriticality extends the test interval beyond six months	34 4/1/77
10.	Transfer Switch No. 7	a.	Verify that the fuse block for breaker 8-1181 to MCC 1 is removed	Monthly, when the reactor is critical and prior to returning reactor to critical when period of subcriticality extended the test interval beyond one month	

TABLE 4.1.2 (continued)

		Check	Frequency	
11.	MOV-LCV-1100 C Transfer Switch	a. Verify that the fuse block for either breaker 8-1198 to MCC 1 or breaker 42-12A76 to MCC 2A is removed.	Same as Item 10 above	
12.	Emergency Siren Transfer Switch	a. Verify that the fuse block for either breaker 8-1145 to MCC 1 or breaker 8-1293A to MCC 2 is removed	Same as Item 10 above	34 4/1/77
13.	Communication Power Panel Transfer Switch	a. Verify that the fuse block for either breaker 8-1195 to MCC 1 or breaker 8-1293B to MCC 2 is removed	Same as Item 10 above	
14a.	Spent Fuel Pool Water Level	Verify water level per Technical Specification 3.8	a. Once every seven days when spent fuel is being stored in the pool.	
b.	Refueling Pool Water Level		b. Within two hours prior to start of and at least once per 24 hours thereafter during movement of fuel assemblies or RCC's.	43 9/25/78
15.	Reactor Coolant Loops/ Residual Heat Removal Loops	a. Per Technical Specifications 3.1.2.C and 3.1.2.D, in Mode 1 and Mode 2 and in Mode 3 with reactor trip breakers closed, verify that all required reactor coolant loops are in operation and circulating reactor coolant. b. Per Technical Specification 3.1.2.E, in Mode 3 with the reactor trip breakers open, verify	a. Once per 12 hours	104 6/9/88

TABLE 4.1.2 (Continued)

	Check	Frequency	
	1. At least two required reactor coolant pumps are operable with correct breaker alignments and indicated power availability.	1. Once per 7 days	
	2. The steam generators associated with the two required reactor coolant pumps are operable with secondary side water level $\geq$ 256 inches of narrow range on cold calibrated scale.	2. Once per 12 hours	
	3. At least one reactor coolant loop is in operation and circulating reactor coolant.	3. Once per 12 hours	
c.	Per Technical Specification 3.1.2.F, in Mode 4 verify		80 10/4/84
	1. At least two required (RC or RHR) pumps are operable with correct breaker alignments and indicated power availability.	1. Once per 7 days	
	2. The required steam generators are operable with secondary side water level $\geq$ 256 inches of narrow range on cold calibrated scale.	2. Once per 12 hours	
	3. At least one reactor coolant loop/RHR train is in operation and circulating reactor coolant.	3. Once per 12 hours	
d.	Per Technical Specifications 3.1.2.G and 3.1.2.H, in Mode 5 verify, as applicable:		

TABLE 4.1.2 (Continued)

	Check	Frequency	
	1. At least one RHR train is in operation and circulating reactor coolant.	1. Once per 12 hours	
	2. When required, one additional RHR train is operable with correct pump breaker alignments and indicated power availability.	2. Once per 7 days	
	3. When required, the secondary side water level of at least two steam generators is $\geq$ 256 inches of narrow range on cold calibrated scale.	3. Once per 12 hours	
e.	Per Technical Specification 3.8.A.3, in Mode 6, with water level in refueling pool greater than elevation 40 feet 3 inches, verify that at least one method of decay heat removal is in operation and circulating reactor coolant at a flow rate of at least 400 gpm.	e. Once per 12 hours	80 10/4/84
f.	Per Technical Specification 3.8.A.4, in Mode 6, with water level in refueling pool less than elevation 40 feet 3 inches, verify		
	1. At least one decay heat removal method is in operation and circulating reactor coolant.	1. Once per 12 hours	
	2. One additional decay heat removal method is operable with correct pump breaker alignments and indicated power availability.	2. Once per 7 days	

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	
Containment Pressure (Wide Range)	M	R	
Steam Generator Water Level (Narrow Range)	M	R	
Refueling Water Storage Tank Water Level	M	R	
Containment Sump Water Level (Narrow Range)	M	R	
Containment Water level (Wide Range)	M	R	
Neutron Flux (Wide Range)	M	R**	117 12/13/88

* See footnote of Table 3.5.6-1.
** Neutron detectors may be excluded from CHANNEL CALIBRATION. 117 12/13/88