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

SACRAMENTO MUNICIPAL UTILITY DISTRICT

DOCKET NO. 50-312

RANCHO SECO NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 93
License No. DPR-54

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

   A. The application for amendment by Sacramento Municipal Utility District (the licensee) dated December 5, 1986, as supplemented March 26, July 31 and November 6, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-54 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 93, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The changes in Technical Specifications are to become effective within 30 days of issuance of the amendment. In the period between issuance of the amendment and the effective date of the new Technical Specifications, the licensee shall adhere to the Technical Specifications existing at the time. The period of time during change over shall be minimized.

4. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

H. Rood for

George W. Knighton, Director
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 5, 1988

January 5, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 93

FACILITY OPERATING LICENSE NO. DPR-54

DOCKET NO. 50-312

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
iii	*iii
1-2	1-2
1-7	1-7
3-1	3-1
3-2	3-2
3-2a	3-2a
3-15a	**3-15a
3-23	3-23
-	3-23a
3-24	3-24
-	3-24a
3-25	3-25
-	3-25a
3-26	3-26
3-26a	3-26a
-	3-26b
3-30a	3-30a
-	3-30b
-	3-30c
3-34	3-34
3-38a	**3-38d
3-38b	**3-38e
3-38c	**3-38f
3-37d	**3-38a
3-37e	**3-38b
3-38f	**3-38c
-	3-38g
-	3-38h
4-7b	4-7b
4-7c	4-7c
-	4-7d
-	4-7e
4-8	4-8
4-8a	4-8a
4-29	**4-29
4-39	4-39
4-39a	4-39a

* Pages issued to revise content and correct previously issued amendments.

** Pages issued to correct previously issued amendments.

TABLE OF CONTENTS (Continued)

140 F. Pressure is defined by Specification 3.1.2. A refueling shutdown refers to a shutdown to replace or rearrange all or a portion of the fuel assemblies and/or control rods.

1.2.7 Refueling Operation

An operation involving a change in core geometry by manipulation of fuel or control rods when the reactor vessel head is removed.

1.2.8 Refueling Interval*

18 months.

1.2.9 Startup

The reactor shall be considered in the startup mode when the shutdown margin is reduced with the intent of going critical.

1.2.10 Remain Critical

A technical specification that requires that the reactor shall not remain critical shall mean that an uninterrupted normal hot shutdown procedure will be completed within 12 hours unless otherwise specified.

1.2.11 T_{avg}

At operating conditions T_{avg} is defined as the arithmetic average of the coolant temperatures in the hot and cold legs of the loop with the greater number of reactor coolant pumps operating, if such a distinction of loops can be made.

1.2.12 Heatup - Cooldown Mode

The heatup-cooldown mode is the range of reactor coolant temperature greater than 200 F and less than 525 F.

1.3 OPERABLE

A component or system is operable when it is capable of performing its intended function within the required range. The component or system shall be considered to have this capability when: (1) it satisfies the limiting conditions for operation defined in Specification 3, (2) it has been tested periodically in accordance with Specification 4, and has met its performance requirements, (3) the system has available its normal and emergency sources of power, and (4) its required auxiliaries are capable of performing their intended function. When a system or component is determined to be inoperable solely because its normal power source is inoperable or its emergency power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation provided its redundant system or component is OPERABLE with an OPERABLE normal and emergency power source.

*See page 1-2b

1.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)

An OFFSITE DOSE CALCULATION MANUAL (ODCM) shall be a manual containing the methodology and parameters to be used in the calculation of offsite dose due to radiioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints and specific details of the environmental radiological monitoring program.

1.16 RESTRICTED AREA

That portion of the site property, the access to which is controlled by security fencing, equipment and personnel.

1.17 SITE BOUNDARY

The boundary of the SMUD owned property.

1.18 DOSE EQUIVALENT I-131

The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites".

1.19 MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all individuals who by virtue of their occupational status have no formal association with the plant. This category shall include non-employees of the licensee who are permitted to use portions of the site for recreational, occupational, or other purposes not associated with plant functions. This category shall <u>not</u> include non-employees such as vending machine servicemen or postmen who, as part of their formal job function, occasionally enter an area that is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

1.20 VECTOR LOGIC

A set of circuitry in each channel of the EFIC system which once AFW has been initiated determines whether AFW to a steam generator should be allowed or terminated and the signal output for each EFIC channel to the AFW valves associated with that channel.

3. LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To specify those limiting conditions for operation of the reactor coolant system which must be met to ensure safe reactor operations.

3.1.1 OPERATIONAL COMPONENTS

Specification

3.1.1.1 Reactor Coolant Pumps

A. Pump combinations permissible for given power levels shall be as shown in specification Table 2.3-1.

B. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant.

C. Operation at power with two pumps shall be limited to 24 hours in any 30 day period.

3.1.1.2 Steam Generators

A. Two steam generators shall be operable whenever the reactor coolant average temperature is above 280 F, except as described in 3.1.1.2.B.

B. With one or more steam generator(s) inoperable due to excessive leakage per 3.1.6.9, bring the reactor to cold shutdown conditions within 48 hours.

C. With one or more steam generator(s) inoperable due to steam generator defective tube(s), restore the inoperable generator(s) to operable status prior to increasing reactor coolant average temperature above 200°F.

3.1.1.3 Pressurizer Safety Valves

A. The reactor shall not remain critical unless both Pressurizer Coolant System code safety valves are operable.

B. When the reactor is subcritical, at least one Pressurizer code safety valve shall be operable if all reactor coolant system openings are closed, except for hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III.

3.1.1.4 Pressurizer Electromatic Relief Valve

A. The nominal setpoint of the pressurizer electromatic relief valve shall be 2450 psig ± 10 psig except when required for cold overpressure protection.

3.1.1.5 Decay Heat Removal

A. At least two of the coolant loops listed below shall be operable when the coolant average temperature is below 280°F. except during fuel loading and refueling.

1. Reactor Coolant Loop (A) and its associated steam generator and at least one associated reactor coolant pump,

2. Reactor Coolant Loop (B) and its associated steam generator and at least one associated reactor coolant pump,

3. Decay Heat Removal Loop (A)

4. Decay Heat Removal Loop (B)

With less than the above required coolant loops OPERABLE, immediately initiate corrective action to return the required coolant loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

3.1.1.6 Reactor Coolant System High Point Vents

A. The vent path on Loop A and vent path on Loop B shall be operable and closed during power operation.

B. The vent path on the pressurizer shall be operable and closed during power operation.

C. With one of the above reactor coolant system vent paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days. If the status is not restored to operable in 30 days, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

D. With two or more of the above reactor coolant system vent paths inoperable; maintain the inoperable vent paths closed with power removed from the valve actuators of all the valves in the inoperable vent paths, and restore at least (two) of the vent paths to OPERABLE status within 72 hours. If the status is not restored to operable in 72 hours, be in HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours.

Bases

A reactor coolant pump or decay heat removal pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One decay heat removal pump will circulate the equivalent of the reactor coolant system volume in one half hour or less. (1)

The decay heat removal system suction piping is designed for 300 F and 300 psig; thus, the system can remove decay heat when the reactor coolant system is below this temperature. (2) (3)

One Pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. (4) Both Pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for rod withdrawal accidents. (5) The Pressurizer code safety valve lift set point shall be set at 2500 psig ± 1 percent allowance for error and each valve shall be capable of relieving 345,000 lb/hr of saturated steam at a pressure not greater than 3 percent above the set pressure.

The electromatic relief valve setpoint was established to prevent operation of the Safety Valves during transients.

Two pump operation is limited until further ECCS analysis is performed.

When the reactor is not critical but TAV is above 280° F, one steam generator provides sufficient heat removal capability for removing decay heat. However, single failure considerations require that both steam generators be operable.

When TAV is below 280°F, a single reactor coolant loop or DHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two DHR loops to be OPERABLE.

The purpose of the high point vents is to vent noncondensible gases from the RCS which may inhibit core cooling during natural circulation. In compliance with 10CFR50 Appendix R the power to all the valve actuators in the vent path has been removed.

REFERENCES

(1) USAR Tables 9.5-2, 4.2-1, 4.2-2, 4.2-4, 4.2-5, 4.2-6

(2) USAR paragraph 9.5.2.2 and 10.2.2

(3) USAR paragraph 4.2.5

(4) USAR paragraph 4.3.8.4 and 4.2.4

(5) USAR paragraph 4.3.6 and 14.1.2.2.3

3.1.7 MODERATOR TEMPERATURE COEFFICIENT OF REACTIVITY (continued)

as power increases. Adjust the moderator coefficient at 15 percent power to the coefficient at any power level above 15 percent.

5. Dissolved boron concentration - This correction is for any difference in boron concentration between zero and full power. Since the moderator coefficient is more positive for greater amounts of dissolved boron, the sign of the correction depends on whether boron is added or removed.

6. Control rod insertion - This correction is for the difference in moderator coefficients between an unrodded and rodded core.

7. Isothermal to distributed temperatures - The correction for spatially distributed moderator temperature effects has been found to be insignificant. Therefore, correction for distributed effects is not required.

REFERENCES

(1) USAR, subsections 14.1 and 14.2

(2) USAR, paragraph 3.2.2.1.5.D

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the operability of the turbine cycle during normal operation and for the removal of decay heat.

Objective

To specify minimum conditions of the turbine cycle equipment necessary to assure the required steam relief capacity during normal operation and the capability to remove decay heat from the reactor core.

Specification

3.4.1 The reactor coolant system shall not be brought or remain above 280F with irradiated fuel in the pressure vessel unless the following conditions are met:

A. Capability to remove decay heat by use of two steam generators as specified in 3.1.1.2.A.

B. One atmospheric dump valve per steam generator shall be operable.

C. A minimum of 250,000 gallons of water shall be available in the condensate storage tank.

D. Two main steam system safety valves are operable per steam generator.

E. Both auxiliary feedwater trains (i.e., pumps and their flow paths) are operable.

F. Both trains of main feedwater isolation on each main feedwater line are operable.

G. Four independent backup instrument air bottle supply systems for ADVs and MFW, SFW, and AFW valves are operable.

With less than the above required components operable, be on decay heat cooling within 72 hours.

3.4.2 The reactor shall not be brought or remain critical unless the following conditions are met:

A. Capability to remove decay heat by use of two steam generators as specified in 3.1.1.2.

B. One atmospheric dump valve per steam generator shall be operable except that: (1) with only one atmospheric dump valve operable, restore an inoperable valve for the other steam generator within 72 hours or be in hot shutdown within 6 hours and on decay heat cooling within the next 12 hours; (2) with no atmospheric dump valves operable, restore at least one inoperable valve within 4 hours or be in hot shutdown within the next 6 hours and on decay heat cooling within the next 12 hours

C. A minimum of 250,000 gallons of water shall be available in the condensate storage tank except that with less than the minimum volume, restore the minimum volume within 4 hours or be in hot shutdown within the next 6 hours and on decay heat cooling within the next 12 hours.

D. Seventeen of the eighteen main steam safety valves are operable except that with less than the minimum number of valves, restore the inoperable valve(s) within 4 hours or be in hot shutdown within the next 6 hours and on decay heat cooling within the next 12 hours.

E. Four turbine throttle stop valves are operable except that with less than the minimum number of valves, restore the inoperable valve(s) within 4 hours or be in hot shutdown within the next 6 hours and on decay heat cooling within the next 12 hours.

F. Both auxiliary feedwater trains (i.e., pump and their flow path) are operable except that:

(1) With one auxiliary feedwater train inoperable, restore the train to operable status within 72 hours or be in hot shutdown within the next 6 hours and on decay heat cooling within the next 12 hours.

(2) With both auxiliary feedwater trains inoperable, the reactor shall be made subcritical within four hours and the reactor shall be on decay heat cooling within the next 12 hours.

3.4.2 G. Both trains of main feedwater isolation on each main feedwater line are operable except that:

(1) With one main feedwater isolation train inoperable, restore the train to operable status within 72 hours or be in hot shutdown within the next 6 hours and on decay heat cooling within the next 12 hours.

(2) With both main feedwater isolation trains inoperable, the reactor shall be made subcritical within four hours and the reactor shall be on decay heat cooling within the next 12 hours.

H. Two independent backup instrument air bottle supply systems (one per steamline) for ADVs are operable except that:

(1) With one system inoperable, restore the system to operable status within 7 days or be in hot shutdown within the next 6 hours and on decay heat cooling within the next 12 hours.

(2) With two systems inoperable, restore at least one system within 24 hours or be in hot shutdown within the next 6 hours. With one system restored to operable status within 24 hours, follow 3.4.2.H.(1).

I. Two independent backup instrument air bottle supply systems (one per feed water line) for MFW, SFW, and AFW control valves are operable except that with either one or both system(s) inoperable, restore the inoperable system(s) within 7 days or be in hot shutdown within the next 6 hours and on decay heat cooling within the next 12 hours.

Bases

The feedwater system and the turbine bypass system are normally used for decay heat removal and cooldown above 280 F. Main feedwater is supplied by operation of a condensate pump and main feedwater pump. If neither main feed pump is available, feedwater can be supplied to the steam generators by an auxiliary feedwater pump. Steam relief capability is provided the system's atmospheric dump valves.

The auxiliary feedwater system is designed to provide sufficient flow on loss of main feedwater to match decay heat plus Reactor Coolant Pump heat input to the Reactor Coolant System before solid pressurizer operation could occur.(4)

The 250,000 gallons of water in the condensate storage tank is sufficient to remove decay heat (plus Reactor Coolant pump heat for two pumps) for approximately 13 hours. This volume provides sufficient water to remove the decay heat for approximately 5.5 hours and to subsequently cool the plant to the DHR system pressure at a cooldown rate of 50°F/hr (1).

The minimum relief capacity of seventeen steam system safety valves is 13,329,163 lb/hr.[2] This is sufficient capacity to protect the steam system under the design overpower condition of 112 percent.[3]

Both trains of main feedwater isolation on each main feedwater line are required to be operable. Train A of main feedwater isolation is comprised of main feedwater control valves, main feedwater block valves and startup control valves. Train B of main feedwater isolation is comprised of the main feedwater isolation valves.

Four independent Class 1 backup air supply systems are provided to assure power available to certain air operated valves in the event of the loss of normal air supply. One system supplies power for the MFW, Startup Feedwater (SFW) and AFW control valves feeding the "A" OTSG; another system supplies power for same valves feeding the "B" OTSG. Two systems supply power for ADVs with one for the ADVs on the "A" main steam line and one for the ADVs on the "B" main steam line. Each system is sized to provide at least two hours of air supply.

REFERENCES

(1) B and W Document 32-1141727-00, "Heat Removal Capability of SMUD CST," March 1984.

(2) USAR paragraph 10.3.4

(3) USAR Appendix 3A, Answer to Question 3A.5

(4) B and W Calculation 86-1167930, "Rancho Seco: AFW Minimum Flow Analysis," (SMUD Calculation No. Z-FWS I0150)

3.5 INSTRUMENTATION SYSTEMS

3.5.1 OPERATIONAL SAFETY INSTRUMENTATION

Applicability

Applies to unit instrumentation and control systems.

Objective

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

3.5.1.1 Startup and operation are not permitted unless the requirements of Table 3.5.1-1, Columns A and B are met.

3.5.1.2 In the event the number of protection or EFIC System channels operable falls below the limit given under Table 3.5.1-1, Columns A and B, operation shall be limited as specified in Column C.

In the event the number of operable Process Instrumentation channels is less than the Total Number of Channel(s), restore the inoperable channels to operable status within 7 days, or be in at least hot shutdown within the next 12 hours. If the number of operable channels is less than the minimum channels operable, either restore the inoperable channels to operable within 48 hours or be in at least hot shutdown within the next 12 hours. If the number of operable channels is two less than the minimum channels operable, the reactor shall be made subcritical within four hours and on decay heat cooling within the next 12 hours.

3.5.1.3 For on-line testing or in the event of a protection instrument or channel failure, a key operated channel bypass switch associated with each reactor protection channel will be used to lock the channel trip relay in the untripped state as indicated by a light. Only one channel shall be locked in this untripped state at any one time.

3.5.1.4 The key operated shutdown bypass switch associated with each reactor protection channel shall not be used during reactor power operation.

3.5.1.5 During startup when the intermediate range instrument comes on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall be maintained in the source range until the one decade overlap is achieved.

3.5.1.6 In the event that one of the trip devices in either of the sources supplying power to the control rod drive mechanisms fails in the untripped state, the power supplied to the rod drive mechanisms through the failed trip device shall be manually removed within 30 minutes. The condition will be corrected and the remaining trip devices shall be tested within eight hours. If the condition is not corrected and the remaining trip devices are not tested within the eight-hour period, the reactor shall be placed in the hot shutdown condition within an additional four hours.

3.5.1.7 For calibration or maintenance of an Emergency Feedwater Initiation and Control (EFIC) channel, a key operated "maintenance bypass" switch associated with each channel will be used which will prevent the initiate signal from being transmitted to the Channel A and B trip logic. Only one channel shall be locked into "maintenance bypass" at any one time.

3.5.1.8 If a channel of the RPS is in bypass, it is permissible to bypass only the corresponding channel of EFIC.

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. A startup is not permitted unless three power range neutron instrument channels and two channels each of the following are operable: four reactor coolant temperature instrument channels, four reactor coolant flow instrument channels, four reactor coolant pressure instrument channels, four pressure-temperature instrument channels, four flux-imbalance flow instrument channels, four power-number of pumps instrument channels, and four high reactor building pressure instrument channels. The safety features actuation system must have two analog channels functioning correctly prior to startup. EFIC system instrumentation as required by Table 3.5.1-1 must be operable.

Operation at rated power is permitted as long as the systems have at least the redundancy requirements of Column B (Table 3.5.1-1). This is in agreement with redundancy and single failure criteria of IEEE 279 as described in FSAR section 7.

The four reactor protection channels were provided with key operated maintenance bypass switches interlocked to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided alarm and lights to indicate when that channel is bypassed.

Bases (Continued)

Each reactor protection channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypass switch is being used. There are four shutdown bypass keys in the control room under the administrative control of the shift supervisor. The keys will not be used during reactor power operation.

There are four reactor protection channels. Normal trip logic is two out of four. Required trip logic for the power range instrumentation channels is two out of three. The EFIC trip logic is two times one-out-of-two taken twice. Minimum trip logic on other instrumentation channels is one out of two.

The EFIC system is designed to automatically initiate AFW when:

1. all four RC pumps are tripped,
2. RPS has tripped the reactor on anticipatory trip indicating loss of main feedwater,
3. the level of either steam generator is low,
4. either steam generator pressure is low, or
5. SFAS ECCS actuation (high RB pressure or low RCS pressure).

The EFIC system will isolate main feedwater to any steam generator when the pressure goes below 600 psig.

The EFIC system is also designed to isolate or feed AFW according to the following logic:

- If both SGs are above 600 psig, supply AFW to both SGs
- If one SG is below 600 psig, supply AFW to the other SG
- If both SGs are below 600 psig but the pressure difference between the two SGs exceeds 100 psig, supply AFW only to the SG with the higher pressure
- If both SGs are below 600 psig and the pressure difference is less than 100 psig, supply AFW to both SGs

At cold shutdown conditions all EFIC initiate and isolate functions are manually or automatically bypassed. When pressure in both steam generators is greater than 750 psig, the following bypassed initiation signals will have been automatically reset: 1) Loss of 4 RC pumps, 2) low steam generator pressure, 3) low steam generator level.

Since the EFIC receives signals from the RPS it is important that only corresponding channels be placed in "maintenance bypass." If a channel of RPS is in maintenance bypass, only the corresponding channel of EFIC can be bypassed. An interlock feature also prevents bypassing more than one EFIC channel at a time. These interlocking features allow the EFIC system to take a single failure in addition to having one channel in maintenance bypass.

Various RPS test features can inhibit initiate signals to the EFIC system and degrade the EFIC system below acceptable limits if the RPS channel is not in bypass. Therefore, no testing should be performed on a RPS instrument string which supplies an output to EFIC without placing that RPS channel in bypass.

The EFIC system is designed to allow testing during power operation. The EFIC system can be tested from its input terminals to the actuated device controllers without placing the channel in key locked "maintenance bypass." A test of the EFIC trip logic will actuate one of two relays in the controllers. The two relays are tested individually to prevent automatic actuation of the component.

Each EFIC channel key operated maintenance bypass switch is provided with alarm and lights to indicate when the maintenance bypass switch is being used.

The source range and intermediate range nuclear flux instrumentation scales overlap by one decade. This decade overlap will be achieved at 10^{-10} amps on the intermediate range scale.

Power is normally supplied to the control rod drive mechanisms from two separate parallel 480 volt sources. Redundant trip devices are employed in each of these sources. If any one of these trip devices fails in the untripped state on-line repairs to the failed device, when practical, will be made, and the remaining trip devices will be tested. Eight hours is ample time to test the remaining trip devices and in many cases make on-line repairs.

The OPERABILITY of the SFAS instrumentation systems and bypasses ensure that 1) the associated SFAS 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 for SFAS purposes 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.

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."

REFERENCE

USAR, Subsection 7.1

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Total Number of Channels	(B) Minimum Channels Operable	(C) Operator Action if Conditions of Columns A and B Cannot be Met
9. Reactor Building Purge Isolation on high radiation	2	1	Operation may continue provided the purge inlet and outlet valves of the inoperable channel(s) are closed and their respective breakers de-energized or comply with 3.5.1.2. At cold shutdown or refueling, each of the purge inlet and outlet valves will be closed.
Emergency Feedwater Initiation and Control (EFIC) System			
1. AFW Initiation			
a. Manual	2 (Note 1)	2 (Note 1)	See Actions 3 and 4.
b. Low Level, SGA or B (Note 2)	4/SG (Note 1)	3/SG	See Actions 1, 2 and 3. May be bypassed below 750 psig OTSG pressure.
c. Low Pressure, SGA or B	4/SG (Note 1)	3/SG	See Actions 1, 2 and 3. May be bypassed below 750 psig OTSG pressure.
d. Loss of MFW Anticipatory Reactor Trip	4 (Note 1)	3	See Actions 1, 2 and 3. Loss of MFW Anticipatory Reactor Trip is effectively bypassed in RPS below 20 percent power.
e. Loss of 4 RC Pumps	4 (Note 1)	3	See Actions 1, 2 and 3. May be bypassed below 750 psig OTSG pressure.
f. Automatic Trip Logic	2 (Note 1)	2 (Note 1)	See Actions 3 and 4.
2. SG-A Main Feedwater Isolation			
a. Manual	2 (Note 1)	2 (Note 1)	See Actions 3 and 4.
b. Low SGA Pressure (Note 3)	4 (Note 1)	3	See Actions 1, 2 and 3. May be bypassed below 750 psig OTSG pressure.
c. Automatic Trip Logic	2 (Note 1)	2 (Note 1)	See Actions 3 and 4.

Note 1 For channel testing, calibration, or maintenance the Total Number of Channels and/or the Minimum Channels Operable may be reduced by one for a maximum of 6 hours providing the remaining channels are OPERABLE.
Note 2 Low level AFW Initiation has a maximum of a 10.0 second delay.
Note 3 Low pressure AFW Initiation has a maximum of a 3.0 second delay.

Table 3.5.1-1 (Continued)

INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Total Number of Channels	(B) Minimum Channels Operable	(C) Operator Action if Conditions of Columns A and B Cannot be Met
3. SG-B Main Feedwater Isolation			
a. Manual	2 (Note 1)	2 (Note 1)	See Actions 3 and 4.
b. Low SGB Pressure (Note 3)	4 (Note 1)	3	See Actions 1, 2 and 3. May be bypassed below 750 psig OTSG pressure.
c. Automatic Trip Logic	2 (Note 1)	2 (Note 1)	See Actions 3 and 4.
4. AFW Valve Commands (Vector)			
a. Vector Enable	2 (Note 1)	2 (Note 1)	See Actions 3 and 4.
b. Vector Module (Note 4)	4 (Note 1)	3	See Actions 1 and 5.
c. Control Enable	2 (Note 1)	2 (Note 1)	See Actions 1 and 3.
d. Control Module	2 (Note 1)	2 (Note 1)	See Actions 1 and 3.

Note 1 For channel testing, calibration, or maintenance the Total Number of Channels and/or the Minimum Channels Operable may be reduced by one for a maximum of 6 hours providing the remaining channels are OPERABLE.
Note 3 Low pressure AFW Initiation has a maximum of a 3.0 second delay.
Note 4 SG Pressure Difference AFW Valve Command (Vector) has a maximum of a 10.0 second delay.

Table 3.5.1-1 (Continued)

INSTRUMENTS OPERATING CONDITIONS

Action 1 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE within 7 days, or be in at least hot shutdown within the next 12 hours.

Action 2 - With the number of OPERABLE channels one less than the Minimum Channels Operable then put one of the inoperable channels in trip, and restore at least one of the inoperable channels to OPERABLE within 48 hours or be in at least hot shutdown within the next 12 hours.

Action 3 - With the number of OPERABLE channels two less than the Minimum Channels Operable, be in at least hot shutdown within 4 hours and in cold shutdown within the following 12 hours.

Action 4 - With the number of OPERABLE channels one less than the Total Number of Channels OPERABLE, restore the inoperable channel to OPERABLE within 48 hours, or be in at least hot shutdown within the next 12 hours.

Action 5 - With the number of OPERABLE channels one less than the Minimum Channels Operable, restore one inoperable channel to OPERABLE within 4 hours or be in at least hot shutdown within the next 12 hours.

3.5.3 SAFETY FEATURES ACTUATION SYSTEM SETPOINTS

Applicability

This specification applies to the safety features actuation system actuation setpoints.

Objective

To provide for automatic initiation of the safety features actuation system in the event of a breach of reactor coolant system integrity.

Specification

The safety features actuation setpoints and permissible bypasses shall be as follows:

Functional Unit	Action	Setpoint
High Reactor Building pressure*	Reactor Building spray valves***	≤30 psig
	Reactor Building spray pumps***	≤30 psig
	High pressure injection, and start of Reactor Building cooling and Reactor Building isolation.	≤4 psig
	Low pressure injection, EFIC AFW initiate	≤4 psig
Low reactor coolant system pressure**	High pressure injection, and start of Reactor Building cooling and Reactor Building Isolation	≥1600 psig
	Low pressure injection, EFIC AFW initiate	≥1600 psig
Automatic Actuation Logic	All above	Not Applicable
Manual	All above	Not Applicable

*May be bypassed during Reactor Building leak rate test.

**May be bypassed below 1850 psig and is automatically reinstated above 1850 psig

***Five-minute time delay.

Figure 3.5.4-1 Incore Instrumentation Specification
Axial Imbalance Indication

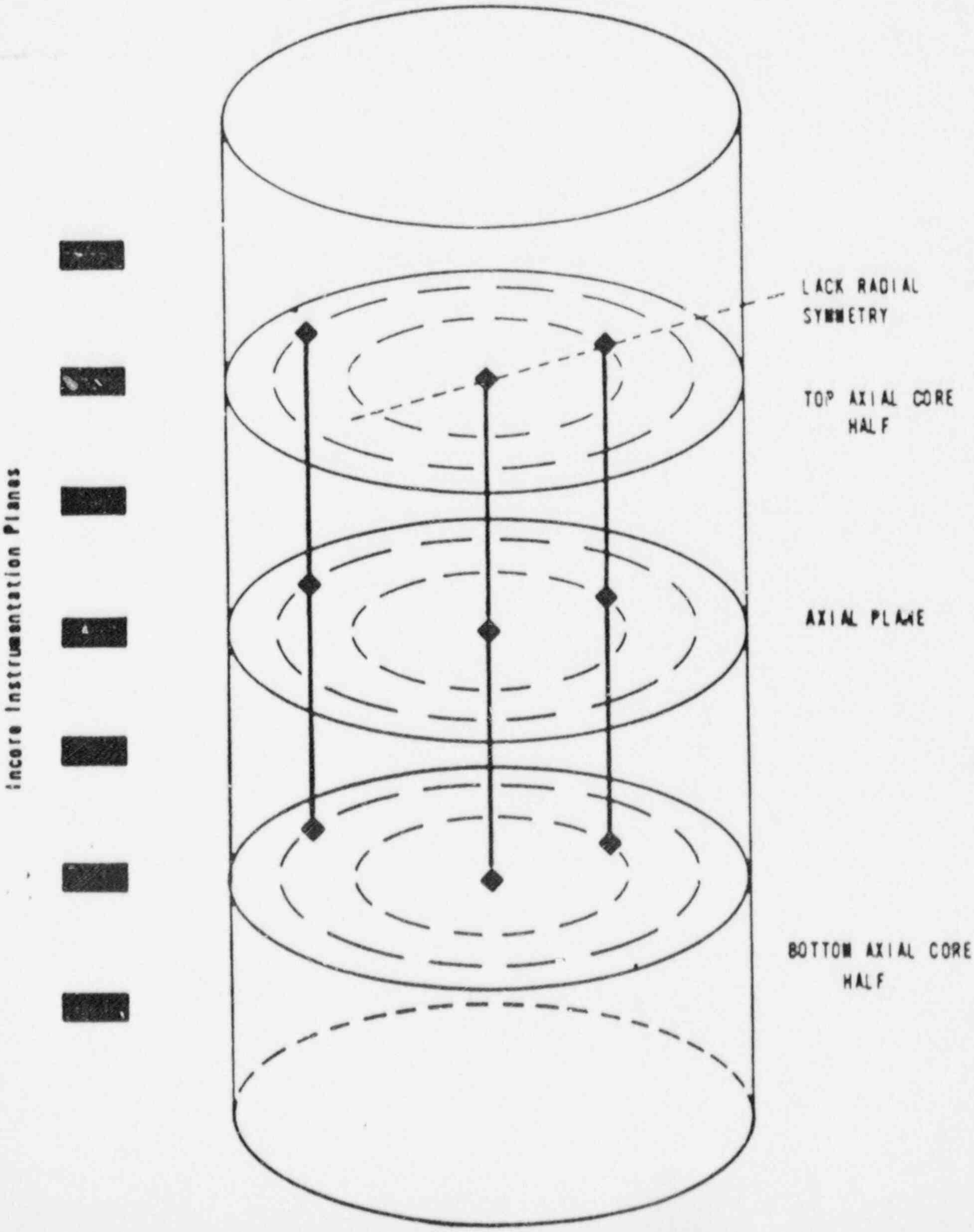


Figure 3.5.4-2 Incore Instrumentation Specification
Radial Flux Tilt Indication

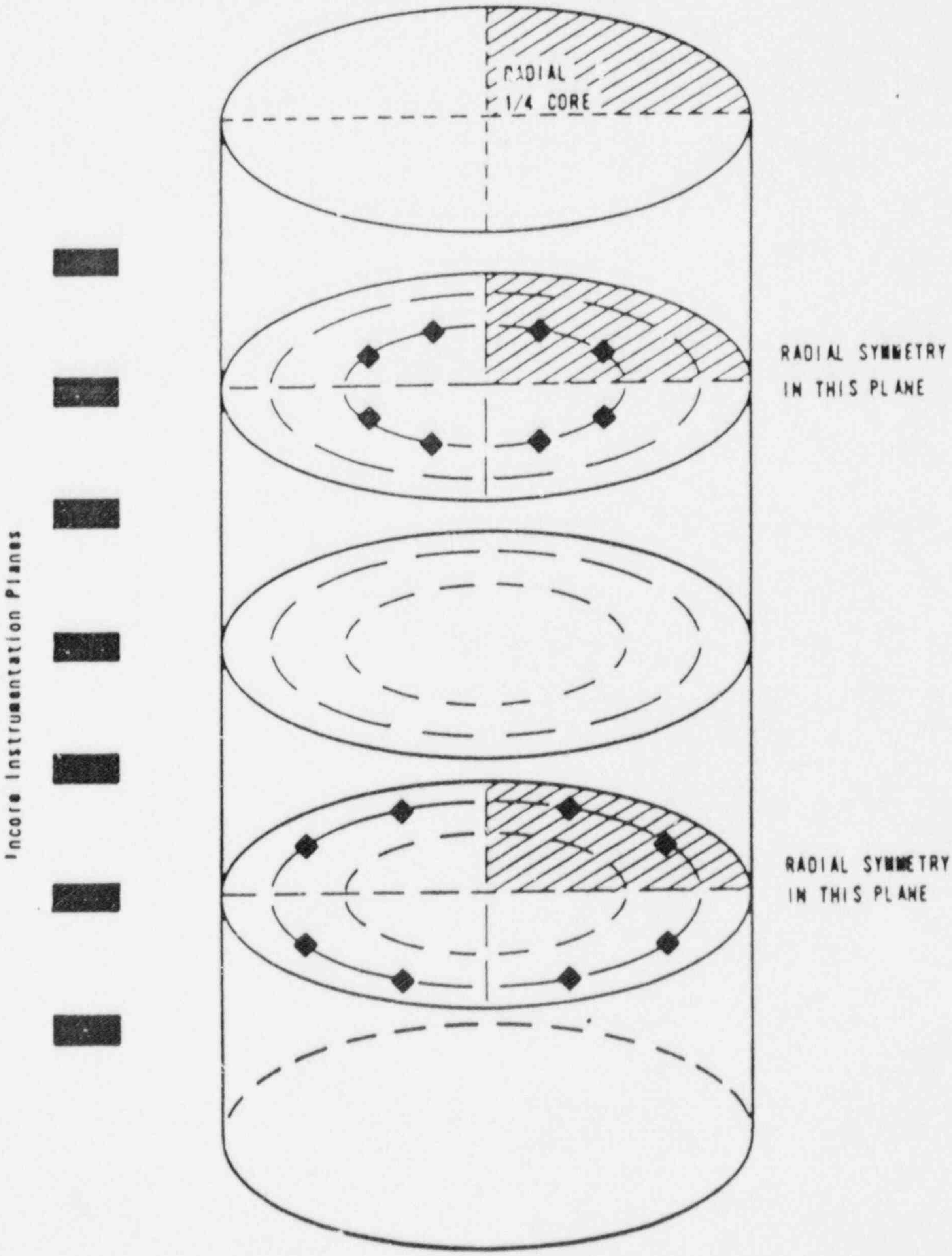
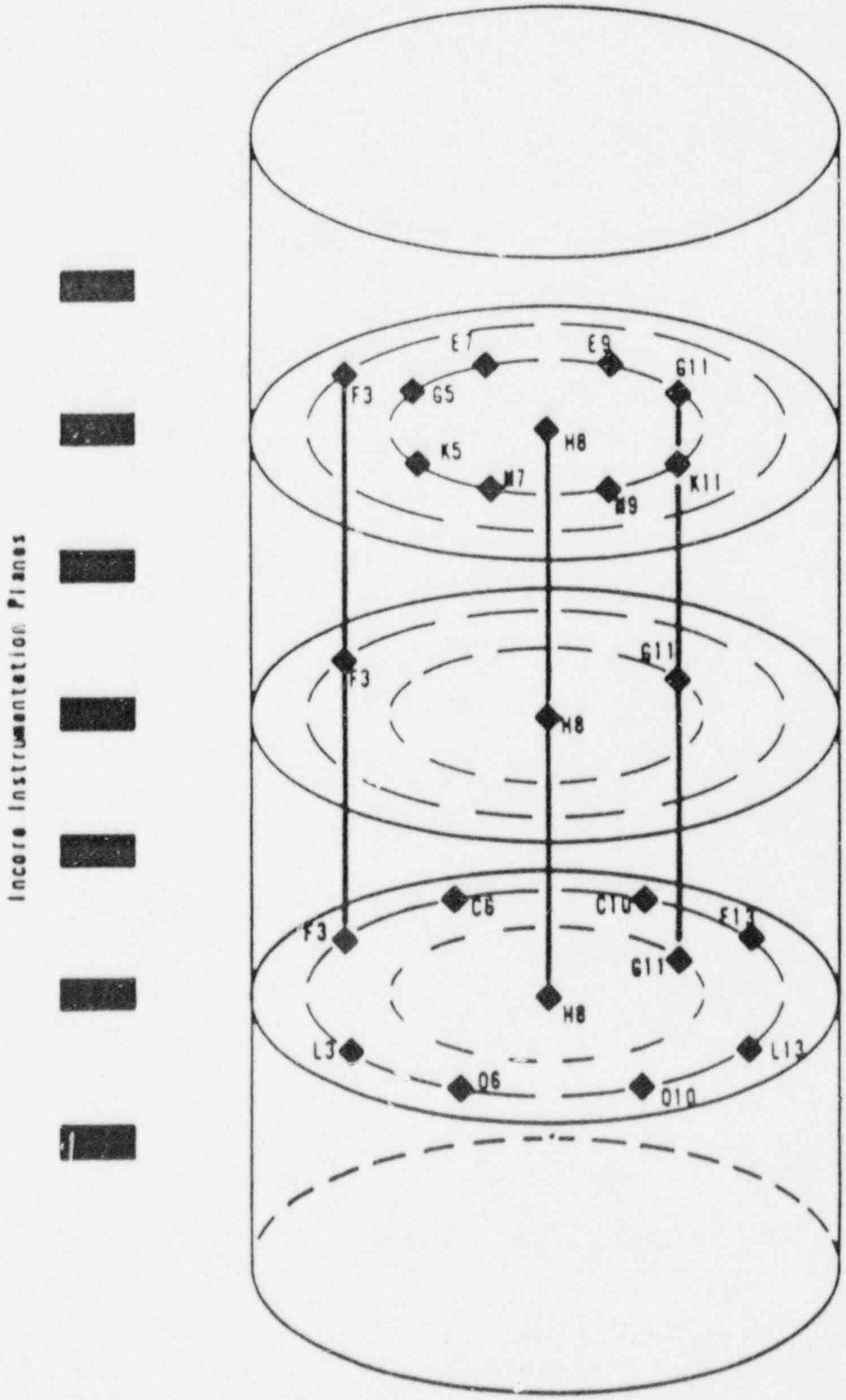


Figure 3.5.4-3 Incore Instrumentation Specification



3.5.5 ACCIDENT MONITORING INSTRUMENTATION

Accident monitoring instrumentation channels shown in Table 3.5.5-1 shall be OPERABLE with their alarm/trip setpoints as shown.

Applicability

As shown in Table 3.5.5-1.

Action

A. With an accident monitoring instrument channel less conservative than the setpoints provided in Table 3.5.5-1, declare the channel inoperable.

B. With less than the minimum number of operable channels, take the ACTION shown in Table 3.5.5-1.

Bases

Table 3.5.5-1 lists the operability requirements for the various types of accident monitoring instrumentation that were installed in response to NUREG 0737, items II.F.1 and II.F.2. This new set of equipment meets or exceeds the amount of coverage outlined in Generic Letter No. 83-37, "NUREG-0737 Technical Specifications." Most of the instrument parameters in Table 3.5.5-1 are monitored by redundant equipment. However, a failure of any one of the radiation monitors described in items 1, 6, and 7 would place that item in an LCO position and would require action number I. If the inoperable channel cannot be repaired within 72 hours Action Statement Number I requires that a pre-planned alternate method of monitoring be initiated. Operating procedures will be used to control the use of backup radiation equipment.

TABLE 3.5.5-1

ACCIDENT MONITORING INSTRUMENTATION OPERABILITY REQUIREMENTS(1)

Instrument	Total Number of Channels	Minimum Number of Channels Operable	Alarm/Trip Setpoint	Action
1. Containment Area High Range Radiation Monitor	2	2	$\leq$2 rad/hr	I
2. Wide Range Containment Water Level	2	1	N/A (Range 0-10 ft)	II
3. Containment Hydrogen Analyzer	2	1	$\leq$4 Percent H_2 Conc	II
4. Emergency Sump Level	2	1	$\leq$4 Ft. (High Alarm on Computer)	III
5. Containment Wide Range Pressure Monitor/Recorder	2	1	N/A (Range -5 to 180 psig)	II
6. High Range Noble Gas Effluent Monitors			N/A(2) (Range 10^{-7} - 10^{5} μc/cc)	I
a) RB Exhaust Stack(3)	1	1		
b) Aux Building Stack	1	1		
c) Radwaste Vent(4)	1	1		
7. Main Steam Lines Radiation Monitors	2	2	$\leq$10 mr/hr	I
8. Subcooling Margin Monitor	2	1	No alarms. Procedural controls in place	II
9. Incore Thermocouples	4/core quadrant	2/core quadrant	(Range 200 - 2300°F)	III

(1) This Table applies at all times except during cold shutdown or refueling.

(2) Alarm limits are set according to the Offsite Dose Calculation Manual.

(3) Monitoring of the RB Exhaust Stack is not required when the purge and/or equalizing valves are closed.

(4) Monitoring of the Radwaste Vent is not required when the unit is not operating.

Table 3.5.5-1 (Continued)

Action

I. With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the pre-planned alternate method of monitoring, and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.5.D, within 30 days following the event, outlining the action taken, the cause of the inoperability, and the corrective action and schedule for implementation.

II. a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels, restore the inoperable channel(s) to OPERABLE status within 30 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 Number of Channels Operable, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

III. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Number of Channels Operable, restore the inoperable channel(s) to OPERABLE status within 30 days, or be in at least HOT SHUTDOWN within the next 12 hours.

3.5.6 EMERGENCY FEEDWATER INITIATION AND CONTROL SETPOINTS

Applicability

This specification applies to the emergency feedwater initiation and control (EFIC) setpoints.

Objective

To provide for automatic initiation and control of auxiliary feedwater and automatic isolation of main feedwater.

Specification

The emergency feedwater initiation and control setpoints and bypasses shall be as follows:

Functional Unit	Action	Setpoint
a. Low SG Level	Initiates AFW	$\geq$9 inches
b. Low SG Pressure	Initiates AFW and Isolates MFW	$\geq$575 psig
c. Loss of All RCP	Initiates AFW	N/A
d. SFAS Actuation	Initiates AFW	N/A (1)
e. RPS Actuation on Loss Of MFW	Initiates AFW	N/A (2)
f. Vector Logic	Isolates Faulted SG	Various (3)
g. Shutdown Bypass	Bypass Permissive	<750 psig

(1) Refer to Specification 3.5.3 for SFAS setpoint
(2) Refer to Table 2.3-1 for RPS setpoint
(3) Refer to Bases below for description of vector setpoints

3.5.6 (continued)

Bases

The EFIC system is designed to automatically initiate AFW when:

1. all four RC pumps are tripped, or
2. RPS has tripped the reactor on anticipatory trip indicating loss of main feedwater, or
3. the level of either steam generator is low, or
4. either steam generator pressure is low, or
5. SFAS ECCS actuation (high RB pressure or low RCS pressure).

The EFIC system will initiate main feedwater isolation to any steam generator as the pressure goes and stays below a minimum set point of 575 psig.

The EFIC system is also designed to isolate or feed AFW according to the following vector logic. Setpoints are nominal and subject to instrument inaccuracies:

- If both SGs are above 600 psig, supply AFW to both SGs
- If one SG is below 600 psig, supply AFW to the other SG
- If both SGs are below 600 psig but the pressure difference between the two SGs exceeds 100 psig, supply AFW only to the SG with the higher pressure
- If both SGs are below 600 psig and the pressure difference is less than 100 psig, supply AFW to both SGs

At cold shutdown conditions all EFIC automatic initiate and isolate functions are manually or automatically bypassed. Prior to a pressure of greater than 750 psig in both steam generators, the following bypassed initiation signals automatically reset: 1) Loss of 4 RC pumps, 2) low steam generator pressure, 3) low steam generator level.

Bypassing of automatic AFW initiation on Loss of MFW Anticipatory Trip or SFAS actuation is controlled by bypass permissive logic within the RPS and SFAS, respectively.

Table 4.1-1 (Continued)
INSTRUMENT SURVEILLANCE REQUIREMENTS

	Channel Description	Check	Test	Calibrate	Remarks
42.	Reactor Building drain accumulation tank level	NA	NA	R	
43.	Incore neutron detectors	M(1)	NA	NA	(1) Check functioning, including functioning of computer readout and/or recorder readout.
44.	a. Process and area radiation monitoring system	W	M	Q	
	b. Containment Area Monitors	W	NA	R	
45.	Emergency plant radiation Instruments	M(1)	NA	R	(1) Battery check
46.	Environmental air monitors	M(1)	NA	R	(1) Check functioning
47.	Strong motion accelerometer	Q(1)	NA	R	(1) Battery check
48.	Deleted				
49.	Pressurizer Water Level	M	NA	R	
50.	Auxiliary Feedwater Flow Rate	M	NA	R	
51.	Reactor Coolant System Subcooling Margin Monitor	M	NA	R	
52.	EMOV Power Position Indicator (Primary Detector)	M	NA	R	
53.	EMOV Position Indicator (Backup Detector) T/C or Acoustic	M	NA	R	
54.	EMOV Block Valve Position Indicator	M	NA	R	
55.	Safety Valve Position Indicator (Primary Detector) T/C	M	NA	R	

TABLE 4.1-1 (Continued)
INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate	Remarks
56. Safety Valve Position Indicator (Backup Detector) Acoustic	M	N/A	R	
57. Voltage Protection	S(1)			(1) Compare voltmeter readings
a. Undervoltage		M	R	
b. Overvoltage		M	R	
c. Time Delay		M	R	
58. Containment Area High Range Monitor	S	M(1)	R	(1) Test using installed source
59. Wide Range Containment Water Level	M	N/A	R	
60. Containment Hydrogen Analyzer	S	M	Q	
61. Emergency Sump Level	M	N/A	R	
62. Containment Wide range Pressure Monitor/Recorder	M	N/A	R	
63. High Range Noble Gas Effluent Monitors - RB Exhaust Stack - Aux. Building Stack - Radwaste Vent	S	M	R	
64. Main Steam Line Radiation Monitors	S	M(1)	R	(1) Test using installed source
65. Subcooling Margin Monitors	M	N/A	R	
66. Incore Thermocouples	M	N/A	R	
67. Low Temperature Over-Pressure Protection (EMOV)	N/A	(1)	R	(1) Prior to cooldown.

TABLE 4.1-1 (Continued)
INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate	Remarks
68. AFW Initiation				
a. Manual	N/A	M	N/A	
b. Low Level SGA or B	S	M (1)	R (1)	(1) Include time delay module.
c. Low Pressure SGA or B	S	M (1)	R (1)	(1) Include time delay module.
d. Loss of MFW Anticipatory Reactor Trip	S	M	N/A	
e. Loss of 4 RC Pumps	S	M	N/A	
f. SFAS Actuation	S	R	N/A	
g. Automatic Trip Logic	S	M	N/A	
h. Bypasses	S	M	R	
69. SGA Main Feedwater Line Isolation				
a. Manual	N/A	M	N/A	
b. Automatic Trip Logic	S	M	N/A	
70. SGB Main Feedwater Line Isolation				
a. Manual	N/A	M	N/A	
b. Automatic Trip Logic	S	M	N/A	

TABLE 4.1-1 (Continued)
INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate	Remarks
71. AFW Valve Commands (Vector)				
a. Vector Enable	S	M	N/A	
b. SGA Pressure Low	S	M	R	
c. SGB Pressure Low	S	M	R	
d. SG Pressure Difference				
SGA Pressure > SGB Pressure	S	M (1)	R (1)	(1) Include time delay module.
SGB Pressure > SGA Pressure	S	M (1)	R (1)	(1) Include time delay module.
72. AFW Control Valve Control				
a. Manual/Auto in Manual	N/A	M	N/A	
73. SG Level Control				
a. Setpoint Selection	N/A	M	N/A	
b. Control Enable	N/A	M	N/A	
c. Module Response	N/A	M (1)	R	(1) Confirm External Controller Settings
74. ADV Control Valve Control				
a. Manual/Auto in Manual	N/A	M	N/A	
75. SG Pressure Control				
a. Module Response	N/A	M (1)	R	(1) Confirm External Controller Settings
76. Backup Instrument Air Supply System				
a. Pressure	D	N/A	N/A	

Table Notations

S - Each shift
D - Daily
W - Weekly
M - Monthly
Q - Quarterly
SY - Semiannual
P - Prior to each startup if not done previous week
R - Once during the refueling interval

TABLE 4.1-2

MINIMUM EQUIPMENT TEST FREQUENCY

Item	Test	Frequency
1. Control rods	Rod drop times of all full length rods	Each refueling shutdown
2. Control rod movement	Movement of each rod	Every two weeks
3. Pressurizer code safety valves	Setpoint	Note 3
4. Main Steam safety valves	Setpoint	Note 3
5. Refueling system interlocks	Functional	Each refueling interval prior to handling fuel
6. Turbine throttle stop valves	Movement of each valve	Monthly
7. Reactor Coolant System	Leakage	Calculated inventory weekly Leakage check daily
8. Charcoal and high efficiency filters	Charcoal and HEPA filter for iodine and particulate removal efficiencies. DOP test on HEPA filters. Freon test on charcoal filter units.	Each refueling interval and at any time work on filters could alter their integrity
9. Fire pumps and power supplies	Functional	Monthly
10. Reactor Building isolation trip	Functional	Each refueling interval
11. Spent fuel cooling system	Functional	Each refueling interval prior to fuel handling
12. Turbine Overspeed Trips	Calibration	Each refueling interval
13. Internals Vent Valves	Manual Actuation,(1) Remote Visual inspection,(2) and verify that valve not stuck open.	Each refueling interval

TABLE 4.1-2 (Continued)

MINIMUM EQUIPMENT TEST FREQUENCY

Item	Test	Frequency
14. Reactor Coolant System High Point Vents	Functional test of each valve$^{(4)}$	Each refueling interval
15. Low Temperature Overpressure Protection (EMOV)	Functional$^{(5)}$	Prior to RCS temperature decreasing below 350°F
16. Main Feedwater Isolation Valves		
a. Main Feedwater Isolation Valves	Functional	Each refueling interval.
b. Main Feedwater Block Valves	Functional	Each refueling interval.
c. Startup Feedwater Control Valves	Functional	Each refueling interval.
d. Main Feedwater Control Valves	Functional	Each refueling interval.
17. Turbine Throttle Stop Valves	Cycle	Each refueling interval.
18. Backup Instrument Air Supply System	Functional	Each refueling interval

1. Verifying through manual actuation that the valve is fully open with a force of $\leq$ 400 lbs. (applied vertically upward).

2. Check visually accessible surfaces to evaluate observed surface irregularities.

3. 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).

4. Cycle each valve in the vent path through at least one complete cycle of full travel from the control room and verify the flow of gas through the system vent path. Verify all manual isolation valves in each vent path are locked in the open position.

5. EMOV block valve closed during test.

4.5.2 REACTOR BUILDING COOLING SYSTEMS

Applicability

Applies to testing of the Reactor Building cooling systems.

Objective

To verify that the Reactor Building cooling systems are operable.

Specification

4.5.2.1 System Tests

A. Reactor Building Spray System

1. During each refueling interval a system test shall be conducted to demonstrate proper operation of the system. A manual trip signal will be applied to demonstrate actuation of the Reactor Building spray system (except for Reactor Building motor-operated inlet valves which prevent water entering nozzles). Water will be circulated from the borated water storage tank through the Reactor Building spray pumps and returned through the test line to the borated water storage tank.

2. The test will be considered satisfactory if visual observation and control board indication verifies that all components have responded to the actuation signal and the appropriate pump breakers shall have opened and closed, and all valves shall have completed their travel except the blocked Reactor Building inlet valve.

3. Air will be introduced into the spray headers to verify the availability of the headers and spray nozzle at least every 10 years.

B. Reactor Building Emérgency Cooling System

1. During each refueling interval, a system test shall be conducted to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

A manual trip signal will be applied to actuate the Reactor Building emergency cooling system for Reactor Building cooling operation.

4.8 AUXILIARY FEEDWATER PUMP PERIODIC TESTING

Applicability

Applies to the periodic testing of the turbine and motor driven auxiliary feedwater pumps.

Objective

To verify that the auxiliary feedwater pump and associated valves are operable.

Specification

4.8.1 Monthly on a staggered test basis at a time when the average reactor coolant system temperature is $\geq$305°F, the turbine/motor driven and motor driven auxiliary feedwater pumps shall be operated on recirculation to the condenser to verify proper operation.

Separate tests will be performed in order to verify the turbine driven capability and the motor driven capability of auxiliary feedwater pump P-318.

The monthly test frequency requirement shall be brought current within 72 hours after the average reactor coolant system temperature is $\geq$305°F for the motor driven pumps. The turbine driven capability shall be brought current within 72 hours of obtaining 5 percent reactor power.

Acceptable performance will be indicated if the pump starts and operates for fifteen minutes at a flow rate sufficient to assure 475 gpm of flow to the Steam Generator at a discharge pressure sufficient to drive that flow through the most restrictive flow path to a single steam generator which is at a pressure of 1050 psig.

The monthly testing of the auxiliary feedwater pumps and valves shall be performed in accordance with the inservice inspection requirements of Specification 4.2.2.1.

4.8.2 At least once per 18 months:

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

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

4.8.3 All auxiliary feedwater system valves, including those that are locked, sealed, or otherwise secured in position, are to be inspected to verify they are in the proper position following surveillances performed pursuant to Specifications 4.8.1, 4.8.2 and 4.8.4.

4.8.4 Prior to startup following a refueling shutdown or any cold shutdown of longer than 30 days duration, conduct a test to demonstrate that the motor-driven AFW pumps can pump water from the CST to the steam generator.

Bases

The monthly test frequency will be sufficient to verify that the turbine/motor driven and motor driven auxiliary feedwater pumps are operable. Verification of correct operation will be made both from the control room instrumentation and direct visual observation of the pumps.

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 305°F from normal operating conditions in the event of a total loss of off-site power.

The electric driven auxiliary feedwater pumps are capable of delivering a total feedwater flow of 475 gpm at a pressure of 1050 psig to the entrance of the steam generators. The steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 475 gpm to the entrance of the steam generators over the steam generator operating range of 800 psig to 1050 psig. This capacity is utilized as analytical input to the Loss of Main Feedwater Analysis which is the design basis event for AFW flow requirements.

4.8.4 Prior to startup following a refueling shutdown or any cold shutdown of longer than 30 days duration, conduct a test to demonstrate that the motor-driven AFW pumps can pump water from the CST to the steam generator.

Bases

The monthly test frequency will be sufficient to verify that the turbine/motor driven and motor driven auxiliary feedwater pumps are operable. Verification of correct operation will be made both from the control room instrumentation and direct visual observation of the pumps.

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 305°F from normal operating conditions in the event of a total loss of off-site power.

The electric driven auxiliary feedwater pumps are capable of delivering a total feedwater flow of 475 gpm at a pressure of 1050 psig to the entrance of the steam generators. The steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 475 gpm to the entrance of the steam generators over the steam generator operating range of 800 psig to 1050 psig. This capacity is utilized as analytical input to the Loss of Main Feedwater Analysis which is the design basis event for AFW flow requirements.