ATTACHMENT 2

TO

AEP:NRC:00449

Proposed Technical
Specifications Changes
for Unit 1

8012180317

SURVEILLANCE REQUIREMENTS

4. Verifying that each automatic valve in the flow path is in the fully open position whenever the auxiliary feedwater system is placed in automatic control or when above 10% RATED THERMAL POWER.

b. At least once per 18 months during shutdown by:

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of the appropriate Engineered Safety Features actuation test signal required by Specification 3/4.3.2.

2. Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of the appropriate Engineered Safety Features actuation test signal required by Specification 3/4.3.2.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION, FEEDWATER ISOLATION AND MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13
c. Containment Pressure-High	3	2	2	1, 2, 3	14
d. Pressurizer Pressure - Low	3	2	2	1, 2, 3#	14
e. Differential Pressure Between Steam Lines - High				1, 2, 3##	
Four Loops Operating	3/steam line	2/steam line any steam line	2/steam line		14
Three Loops Operating	3/operating steam line	1###/steam line, any operating steam line	2/operating steam line		15

TABLE 3.3-3 (Cont'd.)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level -- Low-Low	3/Stm. Gen.	2/Stm. Gen. Any Stm. Gen.	2/Stm. Gen.	1, 2, 3	14*
b. 4 kv Bus Loss of Voltage	2/Bus	2/Bus	2/Bus	1, 2, 3	14*
c. Safety Injection	2	1	2	1, 2, 3	14*
d. Loss of Main Feedwater Pumps	2	2	2	1, 2, 3	14*
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS					
a. Steam Generator Water Level -- Low-Low	3/Stm. Gen.	2/Stm. Gen. Any 2 Stm. Gen.	2/Stm. Gen.	1, 2, 3	14*
b. Reactor Coolant Pump Bus Undervoltage	4-1/Bus	2	3	1, 2, 3	14*
8. LOSS OF POWER					
a. 4 kv Bus Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	14*
b. 4 kv Bus Degraded Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4	14*

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION, FEEDWATER ISOLATION AND MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS		
a. Manual Initiation	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--High	$\leq$ 1.1 psig	$\leq$ 1.2 psig
d. Pressurizer Pressure--Low	$\geq$ 1815 psig	$\geq$ 1805 psig
e. Differential Pressure Between Steam Lines--High	$\leq$ 100 psi	$\leq$ 112 psi
f. Steam Flow in Two Steam Lines--High Coincident with T_{avg} --Low-Low or Steam Line Pressure--Low	$\leq 1.42 \times 10^6$ lbs/hr from 0% load to 20% load. Linear from 1.42×10^6 lbs/hr at 20% load to 3.88×10^6 lbs/hr at 100% load $T_{avg} \geq 541°F$ $\geq$ 600 psig steam line pressure	$\leq 1.56 \times 10^6$ lbs/hr from 0% load to 20% load. Linear from 1.56×10^6 lbs/hr at 20% load to 3.93×10^6 lbs/hr at 100% load. $T_{avg} \geq 539°F$ $\geq$ 580 psig steam line pressure

TABLE 3.3-4 (Cont'd.)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS		
a. Steam Generator Water Level -- Low-Low	≥ 10% of narrow range instrument span each steam generator	≥ 9% of narrow range instrument span each steam generator
b. 4 kv Bus Loss of Voltage	3196 volts with a 2-second delay	3196 ± 18 volts with a 2 ±.2 second delay
c. Safety Injection	Not Applicable	Not Applicable
d. Loss of Main Feedwater Pumps	Not Applicable	Not Applicable
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS		
a. Steam Generator Water Level -- Low-Low	≥ 10% of narrow range instrument span each steam generator	≥ 9% of narrow range instrument span each steam generator
b. Reactor Coolant Pump Bus Undervoltage	≥2750 Volts-each bus	≥2725 Volts-each bus
8. LOSS OF POWER		
a. 4 kv Bus Loss of Voltage	3196 volts with a 2-second delay	3196 ± 18 volts with a 2 ± .2 second delay
b. 4 kv Bus Degraded Voltage	3596 volts with a 2.0 min. time delay	3596 ± 18 volts with a 2.0 minute ± 6 second time delay

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	≤ 13.0#/23.0##
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation-Phase "A"	≤ 18.0#/28.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	≤ 14.0#/48.0##
h. Steam Line Isolation	≤ 8.0
7. Containment Pressure--High-High	
a. Containment Spray	≤ 45.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	≤ 7.0
d. Containment Air Recirculation Fan	≤ 660.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip-Reactor Trip	≤ 2.5
b. Feedwater Isolation	≤ 11.0
9. Steam Generator Water Level--Low-Low	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
b. Turbine Driven Auxiliary Feedwater Pumps	≤ 60.0
10. 4160 volt Emergency Bus Loss of Voltage	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
11. Loss Of Main Feedwater Pumps	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
12. Reactor Coolant Pump Bus Undervoltage	
a. Turbine Driven Auxiliary Feedwater Pumps	≤ 60.0

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION, FEEDWATER ISOLATION AND MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS				
a. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
c. Containment Pressure-High	S	R	M(3)	1, 2. 3
d. Pressurizer Pressure--Low	S	R	M	1, 2, 3
e. Differential Pressure Between Steam Lines--High	S	R	M	1, 2, 3
f. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low or Steam Line Pressure--Low	S	R	M	1, 2, 3
2. CONTAINMENT SPRAY				
a. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
c. Containment Pressure--High-High	S	R	M(3)	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION				
a. Manual	N.A.	N.A.	M(1)	1, 2, 3
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3
c. Containment Pressure--High-High	S	R	M(3)	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg} -- Low or Steam Line Pressure--Low	S	R	M	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION				
a. Steam Generator Water Level--High-High	S	R	M	1, 2, 3
6. MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS				
a. Steam Generator Water Level--Low-Low	S	R	M	1, 2, 3
b. 4 kv Bus Loss of Voltage	S	R	M	1, 2, 3
c. Safety Injection	N.A.	N.A.	M(2)	1, 2, 3
d. Loss of Main Feed Pumps	N.A.	N.A.	M	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS				
a. Steam Generator Water Level--Low-Low	S	R	M	1, 2, 3
b. Reactor Coolant Pump Bus Undervoltage	N.A.	R	M	1, 2, 3
8. LOSS OF POWER				
a. 4 kv Bus Loss of Voltage	S	R	M	1, 2, 3, 4
b. 4 kv Bus Degraded Voltage	S	R	M	1, 2, 3, 4

POST-ACCIDENT INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The post-accident monitoring instrumentation channels shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE post-accident monitoring channels less than required by Table 3.3-11, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8 Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

TABLE 3.3-11

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE
1. Containment Pressure	2
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2
4. Reactor Coolant Pressure - Wide Range	2
5. Pressurizer Water Level	2
6. Steam Line Pressure	2/Steam Generator
7. Steam Generator Water Level - Narrow Range	1/Steam Generator
8. Refueling Water Storage Tank Water Level	2
9. Boric Acid Tank Solution Level	1
10. Auxiliary Feedwater Flow Rate	1/Steam Generator*
11. Reactor Coolant System Subcooling Margin Monitor	1**
12. PORV Position Indicator - Limit Switches***	1/Valve
13. PORV Block Valve Position Indicator - Limit Switches	1/Valve
14. Safety Valve Position Indicator - Acoustic Monitor	1/Valve

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* Steam Generator Water Level Channels can be used as a substitute for the corresponding auxiliary feedwater flow rate channel instrument.

** PRODAC 250 subcooling margin readout can be used as a substitute for the subcooling monitor instrument.

*** Acoustic monitoring of PORV position (1 channel per three valves - headered discharge) can be used as a substitute for the PORV Position Indicator - Limit Switches instruments.

TABLE 4.3-7

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R
4. Reactor Coolant Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Line Pressure	M	R
7. Steam Generator Water Level - Narrow Range	M	R
8. RWST Water Level	M	R
9. Boric Acid Tank Solution Level	M	R
10. Auxiliary Feedwater Flow Rate	M	R
11. Reactor Coolant System Subcooling Margin Monitor	M	R
12. PORV Position Indicator - Limit Switches	M	R
13. PORV Block Valve Position Indicator - Limit Switches	M	R
14. Safety Valve Position Indicator - Acoustic Monitor	M	R

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a water volume less than or equal to 62% of span and at least 150 kW of pressurizer heaters.

APPLICABILITY: MODES 1 and 2

ACTION:

With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 Not applicable.

4.4.4.2 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by transferring power from the normal to the emergency power supply and energizing the required capacity of heaters.

RELIEF VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.11 Three Power Operated Relief Valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3

ACTION:

a. With one PORV inoperable, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specifications 6.9.1.9, 3.0.3 and 3.0.4 are not applicable.

b. With two or more PORVs inoperable, within 1 hour either restore the PORVs to OPERABLE status or close the associated block valves and remove power from the block valves; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one block valve inoperable, within 1 hour either (1) restore the block valve to OPERABLE status or (2) close the block valve and remove power from the block valve or (3) close the associated PORV and remove power from its associated Solenoid valve;otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specifications 6.9.1.9, 3.0.3 and 3.0.4 are not applicable.

d. With two or more block valves inoperable, within 1 hour either (1) restore the block valves to OPERABLE status or (2) close the block valves and remove power from the block valves or (3) close the associated PORVs and remove power from their associated Solenoid valves; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11.1 Each of the three PORVs shall be demonstrated OPERABLE:

a. At least once per 31 days by performance of a CHANNEL FUNCTIONAL TEST, excluding valve operation, and

b. At least once per 18 months by performance of a CHANNEL CALIBRATION.

4.4.11.2 Each of the three block valves shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel. The provisions of Specification 4.0.4 are not applicable when Actions 3.4.11.a or 3.4.11.c are applied.

4.4.11.3 The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by operating the valves through a complete cycle of full travel while the emergency buses are energized by the on-site diesel generators and on-site plant batteries. This testing can be performed in conjunction with the requirements of Specifications 4.8.1.1.2.b and 4.8.2.3.2.c.

SURVEILLANCE REQUIREMENTS (Continued)

b. A total maximum combined interval time for any 3 consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval.

4.0.3 Performance of a Surveillance Requirement within the specified time interval shall constitute compliance with OPERABILITY requirements for a Limiting Condition for Operation and associated ACTION statements unless otherwise required by the specification. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified applicability condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified.

The provisions of Specification 4.0.4 are not applicable to the performance of surveillance activities associated with fire protection technical specifications, 4.3.3.7, 4.7.9 and 4.7.10, until the completion of the initial surveillance interval associated with each specification.

TABLE 3.6-1 (Continued)

VALVE NUMBER		FUNCTION	TESTABLE DURING PLANT OPERATION	ISOLATION TIME IN SECONDS
A. PHASE "A" ISOLATION (Continued)				
57.	NCR-107	PRZ Liquid Sample	Yes	10
58.	NCR-108	PRZ Liquid Sample	Yes	10
59.	NCR-109	PRZ Steam Sample	Yes	10
60.	NCR-110	PRZ Steam Sample	Yes	10
61.	NCR-252	Primary Water to Pressurizer Relief Tank	Yes	10
62.	QCM-250	RCP Seal Water Discharge	No	15
63.	QCM-350	RCP Seal Water Discharge	No	15
64.	QCR-300	Letdown to Letdown Hx.	No	10
65.	QCR-301	Letdown to Letdown Hx.	No	10
66.	RCR-100	PRZ Relief Tank to Gas Anal.	Yes	10
67.	RCR-101	PRZ Relief Tank to Gas Anal.	Yes	10
68.	VCR-10	Glycol Supply to Fan Cooler	Yes	10
69.	VCR-11	Glycol Supply to Fan Cooler	Yes	10
70.	VCR-20	Glycol Supply from Fan Cooler	Yes	10
71.	VCR-21	Glycol Supply from Fan Cooler	Yes	10
72.	XCR-100	Control Air to Containment	No	10
73.	XCR-101	Control Air to Containment Isolation	No	10
74.	XCR-102	Control Air to Containment Isolation	No	10
75.	XCR-103	Control Air to Containment	No	10
B. PHASE "B" ISOLATION				
1.	CCM-451	CCW from RCP Oil Coolers	No	60
2.	CCM-452	CCW from RCP Oil Coolers	No	60
3.	CCM-453	CCW from RCP Thermal Barrier	No	30
4.	CCM-454	CCW from RCP Thermal Barrier	No	30
5.	CCM-458	CCW to RCP Oil Coolers & Thermal Barrier	No	60
6.	CCM-459	CCW to RCP Oil Coolers & Thermal Barrier	No	60
7.	ECR-31	Containment Air Particle Radio Gas Detector	No	10
8.	ECR-32	Containment Air Particle Radio Gas Detector	No	10

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

LICENSE CATEGORY	APPLICABLE MODES	
	1, 2, 3 & 4	5 & 6
SOL	1	1*
OL	2	1
NON-Licensed	2	1
Shift Technical Advisor	1**	None Required

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE ALTERATIONS after the initial fuel loading.

#Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1.

**Shared With D.C. COOK - UNIT 2.

6.3 FACILITY STAFF QUALIFICATIONS

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

6.4 TRAINING

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

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Plant Manager and shall meet or exceed the requirements of Section 27 of the NFPA Code-1976.

6.5 REVIEW AND AUDIT

6.5.1 PLANT NUCLEAR SAFETY REVIEW COMMITTEE (PNSRC)

FUNCTION

6.5.1.1 The PNSRC shall function to advise the Plant Manager on all matters related to nuclear safety.

*Full compliance by January 1,1981.

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.8 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valves against water relief. The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. The requirement that 150 KW of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation conditions.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those parameter limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an

the ASME Boiler and Pressure Vessel Code "Inservice Inspection of Nuclear Reactor Coolant Systems", 1971 Edition and Addenda through Winter 1972.

All areas scheduled for volumetric examination have been pre-service mapped using equipment, techniques and procedures anticipated for use during post-operation examinations. To assure that consideration is given to the use of new or improved inspection equipment, techniques and procedures, the Inservice Inspection Program will be periodically reviewed on a 5 year basis.

The use of conventional nondestructive, direct visual and remote visual test techniques can be applied to the inspection of most reactor coolant loop components except the reactor vessel. The reactor vessel requires special consideration because of the radiation levels and the requirement for remote underwater accessibility.

The techniques anticipated for inservice inspection include visual inspections, ultrasonic, radiographic, magnetic particle and dye penetrant testing of selected parts.

The nondestructive testing for repairs on components greater than 2 inches diameter gives a high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. Repairs on components 2 inches in diameter or smaller receive a surface examination which assures a similar standard of integrity. In each case, the leak test will ensure leak tightness during normal operation.

For normal opening and reclosing, the structural integrity of the Reactor Coolant System is unchanged. Therefore, satisfactory performance of a system leak test at 2235 psig following each opening and subsequent reclosing is acceptable demonstration of the system's structural integrity. These leak tests will be conducted within the pressure-temperature limitations for Inservice Leak and Hydrostatic Testing and Figure 3.4-1.

3/4.4.11 RELIEF VALVES

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