ATTACHMENT 3

TO

AEP:NRC:00449

Proposed Technical
Specifications Changes
for Unit 2

~~8404180321~~

20 pp

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 (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[illegible] (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 Line Pressure--Low	
a. Safety Injection (ECCS)	≤ 12.0#/24.0##
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation-Phase "A"	≤ 18.0#/28.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
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	≤ 600.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip-Reactor Trip	Not Applicable
b. Feedwater Isolation	Not Applicable
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 (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

TABLE 3.3-10

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 4.3-10

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

REACTOR COOLANT SYSTEM

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, 2, and 3

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 SHUTDOWN with the reactor trip breakers open within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

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.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME IN SECONDS
A. PHASE "A" ISOLATION (Continued)		
61. NCR-252	Primary Water to Pressurizer Relief Tank	$\leq$ 10
62. QCM-250	RCP Seal Water Discharge	$\leq$ 15
63. QCM-350	RCP Seal Water Discharge	$\leq$ 15
64. QCR-300	Letdown to Letdown Hx.	$\leq$ 10
65. QCR-301	Letdown to Letdown Hx.	$\leq$ 10
66. RCR-100	PRZ Relief Tank to Gas Anal.	$\leq$ 10
67. RCR-101	PRZ Relief Tank to Gas Anal.	$\leq$ 10
68. VCR-10	Glycol Supply to Fan Cooler	$\leq$ 10
69. VCR-11	Glycol Supply to Fan Cooler	$\leq$ 10
70. VCR-20	Glycol Supply from Fan Cooler	$\leq$ 10
71. VCR-21	Glycol Supply from Fan Cooler	$\leq$ 10
72. XCR-100	Control Air to Containment	$\leq$ 10
73. XCR-101	Control Air to Containment Isolation	$\leq$ 10

BASES

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

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 at HOT STANDBY.

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 chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these 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 adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	ISOLATION TIME IN SECONDS
A.	PHASE "A" ISOLATION (Continued)		
74.	XCR-102	Control Air to Containment Isolation	≤ 10
75.	XCR-103	Control Air to Containment	≤ 10
B.	Phase "B" ISOLATION		
1.	CCM-451	CCW from RCP Oil Coolers	≤ 60
2.	CCM-452	CCW from RCP Oil Coolers	≤ 60
3.	CCM-453	CCW from RCP Thermal Barrier	≤ 30
4.	CCM-454	CCW from RCP Thermal Barrier	≤ 30
5.	CCM-458	CCW to RCP Oil Coolers & Thermal Barrier	≤ 60
6.	CCM-459	CCW to RCP Oil Coolers & Thermal Barrier	≤ 60
7.	ECR-31	Containment Air Particle Radio Gas Detector	≤ 10
8.	ECR-32	Containment Air Particle Radio Gas Detector	≤ 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.

#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 1.

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 Radiatic 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 response 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.

COMPOSITION

6.5.1.2 The PNSRC shall be composed of the:

Chairman:	Plant Manager or designated alternate
Member:	Asst. Plant Manager
Member:	Operations Supervisor
Member:	Technical Supervisor
Member:	Maintenance Supervisor
Member:	Instrument and Control Engineer
Member:	Nuclear Engineer
Member:	Chemical Supervisor
Member:	Performance Supervising Engineer
Member:	Radiation Protection Supervisor

*Full compliance by January 1,1981

3/4.3.3.6 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.

3/4.3.3.7 AXIAL POWER DISTRIBUTION MONITIORING SYSTEM (APDMS)

OPERABILITY of the APDMS ensures that sufficient capability is available for the measurement of the neutron flux spatial distribution within the reactor core. This capability is required to 1) monitor the core flux patterns that are representative of the peak core power density and 2) limit the core average axial power profile such that the total power peaking factor F_Q is maintained within acceptable limits.

3/4.3.3.8 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 returned to service.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

BASES

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

3/4.4.10 STRUCTURAL INTEGRITY

The inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

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.

ATTACHMENT 4 TO AEP:NRC:00449

DONALD C. COOK NUCLEAR PLANT UNIT NOS. 1 AND 2
DOCKET NOS. 50-315 AND 50-316
LICENSE NOS. DPR-58 AND DPR-74

PROPOSED LICENSE CONDITIONS
FOR SYSTEMS INTEGRITY AND
IODINE MONITORING

System Integrity

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

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and
2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

Iodine Monitoring

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

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.